10 STEAM AND POWER CONVERSION SYSTEM

10.1 Summary Description

The steam and power conversion system is designed to remove heat energy from the reactor and to generate electric power in the turbine generator. After the steam passes through the high- and low-pressure turbines, the main condensers (MCs) will condense and deaerate the low-pressure turbine exhaust and transfer the rejected heat to the circulating water system, which, in turn, will reject the heat to the power cycle heat sink (PCHS). The condensate will be reheated and returned as feedwater to the reactor. The entire system is designed for the maximum expected energy from the nuclear steam supply system. GE states in SSAR Section 10.1 that nothing in the ABWR standard plant design is meant to preclude the use of a once-through cooling system and a single pressure condenser nor will such changes affect the nuclear island.

A turbine steam bypass system is designed to discharge at least 33-percent of the reactor's design steam flow directly to the condenser during certain transient conditions. GE states that although the ABWR standard plant design is for 33-percent bypass, this capability could be increased to a full-load reject capability without affecting the nuclear island.

10.2 Turbine Generator

10.2.1 Turbine Generator System

The staff reviewed the turbine generator system (TGS) in accordance with Standard Review Plan (SRP) Section 10.2. The design of a TGS is acceptable if the integrated design meets the requirement of General Design Criteria (GDC) 4 as to protect structures, systems, and components (SSC) important to safety from the effects of turbine missiles by providing a turbine overspeed protection system to minimize the probability of generating turbine missiles.

The 188.5 rad/s (1800-rpm) turbine generator unit will be a compound-type unit with one double-flow high-pressure turbine and three double-flow low-pressure turbines, in tandem, coupled directly to a generator with a nominal rating of approximately 1400 Mwe. Each low-pressure turbine will exhaust to a multi-pressure three-shell, single-pass surface condenser.

The turbine generator is equipped with an electrohydraulic control system that will perform two basic functions: (1) turbine speed control for a variety of operating load conditions for which a digital control and monitoring system will be used and (2) turbine overspeed protection. The design functions of the turbine speed control system are (1) to control turbine speed throughout the normal range of load conditions and ensure that a full-load turbine trip will not cause the turbine to overspeed beyond its design overspeed and (2) to provide turbine overspeed protection to minimize the probability of the generation of turbine missiles, in accordance with GDC 4. The turbine control system is, therefore, important to the overall safe operation of the plant.

The turbine is equipped with four turbine stop valves, four turbine control valves, and six combined intermediate valves collectively referred to as turbine steam admission valves. The turbine stop valves and turbine control valves are located upstream of the high pressure turbine steam inlet. The combined intermediate valves are located between the moisture separators and the steam inlets to the three low-pressure turbines. The combined intermediate valves consist of an intermediate stop valve and an intercept valve in a single casing; each will have separate operating mechanisms and controls. The turbine stop valves and the intermediate stop valves will be in the full-open position during normal operation. The control valves are designed to modulate with load on the turbine generator. The intercept valves will modulate, as required, to control turbine speed following a load rejection. All of these valves will be capable of closing in approximately 0.2 second.

The speed control unit is designed to provide speed error signals to a load control unit. These signals, in turn, will operate to open or close the valve, as required, to maintain desired turbine steam flow. In the case of a generator load rejection up to and including full load followed by an increase in turbine speed, the speed-control unit will close both the control and intercept valves to limit turbine overspeed as follows: (1) the control and intercept valves will start to close at approximately 101 percent of rated speed and (2) the control and intercept valves will be fully closed by the time the turbine reaches approximately 104 percent of rated speed.

The turbine overspeed protection system will consist of mechanical and electrical overspeed control systems. At a predetermined speed (110 percent of rated speed), centrifugal force causes the loss of hydraulic pressure to the associated turbine steam admission valve actuators, thus closing the turbine steam admission valves. The electrical overspeed trip system will back up the mechanical overspeed trip. At a predetermined speed (111 percent of rated speed), solenoids will be deenergized. This, in turn, will actuate the electrical trip valve to release hydraulic pressure to the associated turbine steam admission valve actuators, thus closing the turbine steam admission valves.

Protection of safety-related SSC from turbine generator missiles is also assured by proper turbine-generator orientation. This is discussed in SSAR Section 3.5.1.1.3 and reviewed in Section 3.5.1.3 of this report.

A number of turbine generator electrical and mechanical parameters will be monitored during operation. An abnormal condition, as described in SSAR Section 10.2.2.5, in these monitored parameters will also cause a trip of turbine main stop and control valves, and combined intermediate valves by way of their disk/pump valves. These emergency trips will further reduce the possibility of a turbine missile by shutting down the turbine before overspeed or mechanical failures can occur. Some parameters that will be monitored include turbine shaft, vibration, various temperatures and fluid levels, condenser vacuum, EHC electrical power, lube oil and hydraulic pressure, generator trip, electrical and mechanical overspeed, and thrust bearing wear. All of the above trip signals except vibration (2 out of 2 per bearing) and manual trips use 2 out of 3 or 2 out of 4 coincident trip logic.

The turbine steam admission valves can be manually tripped and will automatically trip on loss of power to the hydraulic and control systems, or on loss of both speed control signals.

An inservice inspection (ISI) program for the turbine stop and control valves and combined intermediate valves will be provided and will include: (1) dismantling and inspecting at least one turbine stop valve, one turbine control valve, one stop valve, and one intercept valve at approximately 3 ⅓-year intervals during refueling or maintenance shutdowns coinciding with the ISI schedule and (2) testing the turbine stop valves, the control valves, the combined intermediate valves and the extraction steam nonreturn valves at least once a week. At least once per month, closure of each turbine stop valve, control valve, and combined intermediate valve will be verified by direct observation of the valve motion. GE has included preoperational and startup tests of the turbine generator in accordance with Regulatory Guide (RG) 1.68, "Initial Test Programs for Water-Cooled Power Plants," (Rev. 2). Testing is discussed in Section 10.2.3.6 of the SSAR. The adequacy of the test program is evaluated in Section 14.2 of this report.

GE has committed to provide turbine generator equipment shielding and access control for all areas of the turbine building (TB) that will meet the dose criteria required by 10 CFR Part 20 for operating personnel. This subject is evaluated in the discussion of the radiation protection design acceptance criteria (DAC) in Chapter 12 of this report.

The turbine generator system meets Branch Technical Positions Auxiliary Systems Branch (ASB) 3-1, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment," and Mechanical Engineering Branch (MEB) 3-1, "Postulated Break and Leakage Locations in Fluid Systems Outside Containment." Evaluation of protection against dynamic effects associated with a postulated pipe failure is covered in Section 3.6 of this report.

GE submitted the design description and the inspections, tests, analyses, and acceptance criteria relating to the TGS. This was identified as draft final safety evaluation report (DFSER) Open Item 10.2.1-1. GE provided a revised set of design descriptions and ITAAC. The adequacy and acceptability of the final certified advanced boiling water reactor (ABWR) design description and the ITAAC are evaluated in Section 14.3 of this report. On the basis of this evaluation, this item is resolved.

The TGS includes all components and equipment, including the turbine stop and control valves and the combined intermediate valves. The scope of the review of the TGS for the ABWR included layout drawings, piping and instrumentation diagrams (P&IDs), and descriptive information for the system and for control and supporting systems that are essential to its operation.

The basis for acceptance of the TGS was conformance of the design, design criteria, and design bases to the Commission's regulations as set forth in the GDC of Appendix A to 10 CFR Part 50. The staff concludes that the design is acceptable and meets the requirements of GDC 4 with respect to the protection of SSC important to safety from the effects of turbine missiles. GE has met this requirement by providing a turbine overspeed protection system to control the turbine action under all operating conditions and to ensure that a full-load turbine trip will not cause the turbine to overspeed beyond acceptable limits and will not result in turbine missiles.

The staff concludes that the TGS can perform its design function, meets the guidelines of SRP Section 10.2, and is acceptable.

10.2.2 Turbine Rotor Integrity

GDC 4 requires that structures, systems, and components important to safety shall be appropriately protected against environmental and dynamic effects, including the effects of missiles, that may result from equipment failure. Because turbine rotors have large masses and rotate at relatively high speeds during normal reactor operation, failure of a rotor may result in the generation of high energy missiles and cause excessive vibration of the turbine rotor assembly. The staff reviewed the measures taken by the applicant to assure turbine rotor integrity and reduce the probability of turbine rotor failure.

The staff utilized the guidelines of SRP Section 10.2.3 to review and evaluate the information submitted by the applicant to assure rotor integrity and low probability of turbine rotor failure with the generation of missiles.

As discussed in SSAR Section 10.2.3, turbine rotors and parts will be made from vacuum-melted or vacuum-degassed Ni-Cr-Mo-V alloy steel by processes that minimize flaw occurrence and provide adequate fracture toughness. The fracture appearance transition temperature (FATT) (50 percent FATT), as obtained from Charpy tests, will be no higher than -17.8 °C (0 °F) for low-pressure turbine rotors. The Charpy V-notch energy at the minimum operating temperature of low-pressure rotors in the tangential direction will be at least 8.3 kgm (60 ft-lbs).

The ratio of fracture toughness (K_{IC}) of the rotor material to the maximum tangential stress at speeds from normal to 115 percent of rated speed will be at least 10 square root millimeters (2 ksi square root in.). However, K_{IC} will be used only for materials that exhibit a well-defined Charpy energy and FATT curve and are strain-rate insensitive.

In the DFSER the staff stated that the applicant or a licensee referencing the ABWR standard plant design should submit the turbine rotor test data and the calculated toughness curve to the NRC staff for review. This requirement should be included in the turbine ITAAC and was identified as DSFER Open Item 10.2.2-1. The staff, upon further consideration, reclassified this as a COL action item. GE addressed this item in the SSAR which states the COL applicant will provide turbine inservice test and inspection requirements to the staff for review. This is acceptable to the staff and therefore, Open Item 10.2.2-1 is resolved.

Sufficient warmup time and adequate metal temperature is to be specified by the COL applicant in the turbine operating instruction to ensure that toughness will be adequate to prevent brittle fracture during startup. This was identified as DFSER COL Action Item 10.2.2-1. GE addressed this item in SSAR Section 10.2.5, which states that the COL applicant will provide the turbine material property data and assure sufficient turbine warmup time as required by Subsection 10.2.3.2 of the ABWR SSAR.

The combined stresses of low low-pressure turbine rotor at design overspeed resulting from centrifugal forces, interference fit, and thermal gradients will not exceed 75 percent of the minimum specified yield strength of the material. The design overspeed of the turbine will be 5 percent above the highest anticipated speed resulting from a loss of load. In the DFSER, the staff stated that the applicant or licensee referencing the ABWR standard plant design should provide the basis for the turbine design overspeed to the NRC. This requirement should be included in the turbine ITAAC and was identified as DFSER Open Item 10.2.2-2. The staff, upon further consideration, reclassified this as a COL action item. GE addressed this item in the SSAR Section 10.2.5, which states that the COL applicant will provide the basis for the turbine overspeed design to the staff for review. This is acceptable to the staff and therefore, Open Item 10.2.2-2 is resolved.

The ABWR ISI for the turbine assembly will include the high- and low-pressure turbine rotor, low-pressure turbine buckets, turbine shafts, couplings, and coupling bolts. During plant shutdown, coinciding with the ISI schedule for ASME Code, Section III components, turbine inspection will be performed in sections so that a complete turbine inspection will be performed at least once every 10 years. The low-pressure turbines in currently operating nuclear plants are inspected on an average of once every 5 operating years. However, most of these turbines are of shrunk-on design, which is more susceptible to stress corrosion cracking than the forged monoblock rotor in the ABWR turbine. Thus, the extended inspection interval for the ABWR is acceptable.

The turbine rotor design will be a solid forged monoblock rotor rather than one with shrunk-on disks. The current practice employed by the turbine manufacturers is to bore the center of the monoblock rotor to remove metal impurity and permit internal inspection. A forged or welded rotor will not be as susceptible to stress corrosion cracking as experienced in the shrunk-on disks. However, the one-piece rotor design requires stringent part-machining inspections. Therefore, the applicant referencing the ABWR design must submit inspection requirements for a one-piece rotor to the NRC staff for review and approval before plant operation. Further, the applicant must submit an actual turbine inspection schedule following the third refueling outage. The actual turbine inspection schedule should be based on a probability calculation of turbine missile generation. The calculated probability for turbine missile generation is expected to be less than or equal to 1.0E-4 per year since GE specified that the turbine be favorably oriented. The NRC-approved methodology for calculating probability for turbine missiles is discussed in NUREG-1048 (Supplement 6, July 1986). This was identified as DFSER COL Action Item 10.2.2-1. GE addressed this item in the SSAR Subsection 3.5.1.1.1.3 to state that the COL applicant will submit for NRC approval, within 3 years of obtaining a COL, a turbine maintenance program including probability calculations of turbine missile generation based on methodology approved by the NRC. Further, during the preservice inspection of the turbine, each machined turbine rotor is subjected to 100-percent ultrasonic examination and surface visual examinations, using established acceptance criteria. The ISI program for the turbine assembly includes the disassembly of the turbine and complete inspection of all normally inaccessible parts, such as couplings, coupling bolts, turbine shafts, low-pressure turbine brackets, and low-pressure and high pressure rotors. This is acceptable to the staff.

The staff concludes that the integrity of the turbine rotor is acceptable and meets the relevant requirements of GDC 4 of 10 CFR Part 50. GE has met the requirements of GDC 4 of 10 CFR Part 50 with respect to the commitment to use material of acceptable fracture toughness and elevated temperature properties, adequate design, and the requirements for preservice and ISIs. GE has also described its program for ensuring the integrity of low-pressure turbine rotor through the use of suitable materials of adequate fracture toughness and conservative design practices. The GE program will provide reasonable assurance that the probability of failure with missile generation will be low during normal operation, including transients in which the turbine speed may reach its design overspeed.

10.3 Main Steam Supply System

10.3.1 System Description and Operation

The staff reviewed the main steam supply system (MSSS) in accordance with SRP Section 10.3. The design of the MSSS is acceptable if the design is in accordance with GDC 2 as it relates to safety-related portions of the system being capable of withstanding the effects of natural phenomena such as earthquakes, tornados, hurricanes, and floods, GDC 4 as it relates to safety-related portions of the system being capable of withstanding the effects of missiles, pipe whip, and jet impingement forces associated with pipe breaks, and GDC 5 as it relates to the capability of shared systems and components important to safety to perform required safety functions. Compliance with RG 1.115, "Protection Against Low-Trajectory Turbine Missiles," is evaluated in Section 3.5.1.3 of this report. The system design should adequately consider steam hammer and relief valve discharge loads to ensure that system safety functions can be achieved and should ensure that operating and maintenance procedures include adequate precautions to avoid steam hammer and relief valve discharge loads. The system design should also include protection against water entrainment.

The MSSS is designed to supply the required amount of steam at the required pressure and temperature to the turbine, reheaters, condenser evacuation system, turbine gland sealing system (TGSS), and offgas system.

The MSSS extends from the seismic interface restraint to the turbine stop valves and also includes connected piping up to and including the first shutoff valve on the connected lines. The safety-relief valves, which will be mounted on the main steamlines upstream of the containment isolation valves for the system, are evaluated separately in Section 5.2.2 of this report.

The steam generated in the reactor vessel will be routed to the turbine and power cycle auxiliary equipment via four 70-cm (28-in.) nominal diameter MSL. Each MSL will be equipped with a flow restrictor and two main steam isolation valves (MSIVs), thus ensuring MSL isolation in the event of a steamline break outside the containment and a concurrent failure of an MSIV. One MSIV is located immediately inside the drywell and the other immediately outside the drywell. The MSIVs are designed to provide positive isolation against steam flow associated with a MSL break. They will be pneumatic or spring-operated (to close), fast-closing (3-4.5 seconds), Y-pattern, globe valves. Operating fluid will be supplied to the valves from the nitrogen supply system. Nitrogen accumulators will supply backup operating nitrogen for the MSIVs in the event of a loss of the normal nitrogen supply system. Open Item 66 in the draft safety evaluation report (DSER) (SECY-91-235) required GE to clarify whether the backup nitrogen accumulators were seismic Category I. Subsequently, in a meeting with the staff on May 5, 1992, GE clarified that these accumulators are seismic Category I as shown on the nuclear boiler (NB) system P&Ds, ABWR Figure 5.1-3, page 3 of 11. Therefore, this item was resolved in the DFSER.

Downstream of the outboard MSIVs and upstream of the turbine stop valves, the MSLs that will be routed to the turbine contain no other shutoff valves. From the MSL header, in addition to the four steamlines to the turbine, two steamlines will supply steam to the power cycle auxiliary equipment. One of the branch steamlines will supply steam to the TGSS and to two reheaters. The other branch line will supply steam to the offgas system, the steam jet air ejectors (SJAE), the condenser sparger, and two other reheaters. Each of these steamlines contains a power-operated pneumatic gate shutoff valve. These valves are 41-cm (16-in.) diameter, Quality Group (QG) B, seismic Category I, shut within approximately 2 seconds following an MSIV closure signal, and are equipped with air operators and spring closure mechanisms. These valves fail closed on loss of electrical power to the valve actuating solenoid or on loss of pneumatic pressure and are analysed to demonstrate structural integrity under SSE loading conditions.

SSAR Section 5.4.9 and Table 3.2-1 state that the MSLs from the reactor vessel, out to and including the outboard MSIVs, are designed to QG A standards. From the outboard MSIVs to the turbine stop valves, the steamlines and associated equipment are designed to QG B standards. The MSLs that will extend from the reactor vessel up to and including the seismic interface restraint (which is downstream of the outboard MSIVs) are seismic Category I. Downstream of the seismic interface restraints, the main steam (MS) piping and equipment is classified as nonnuclear safety-related. This includes the shutoff valves on the two branch steamlines (inside the TB) that will supply steam to the power cycle auxiliary equipment (QG D).

SSAR Section 3.2.5.3, however, states that the MSLs from the containment outboard isolation valves, up to and including the turbine stop valves and all branch lines 6.3 cm (2.5 in) and larger and up to and including the first valve and its supports, are designed using an appropriate dynamic seismic system analysis to withstand the safe shutdown earthquake (SSE) design loads in combination with appropriate loads within the limits specified. In the DSER (SECY-91-235), the staff stated that the design for the MSLs downstream of the outboard MSIVs up to the turbine stop valves, the connecting lines up to the turbine bypass valves, and all other connecting lines up to and including the first shutoff valves, did not comply with the staff's requirement for seismic and QG classifications as stated in SRP Section 10.3, Criterion III.3.b, which requires that the subject portions of the MSSS be designed to seismic Category I and QG B requirements. Therefore, the staff identified the lack of seismic Category I classification of the subject portions as DSER Open Item 3 (SECY-91-153). Although the subject piping portions will not be seismic Category I, by letter dated April 1, 1992, GE committed to apply the quality assurance criteria of 10 CFR Part 50, Appendix B, to these portions. The combination of appropriate dynamic seismic system analysis, QG B classification, and application of the Appendix B criteria to these portions are adequate and acceptable. Therefore, this open item was resolved in the DFSER (this issue is further discussed in Section 3.2.1 of this report).

In addition to meeting the acceptance criteria of SRP Section 10.3, the MS system for the ABWR must also be capable of mitigating the radiological consequences of an accident that could result in potential offsite exposures comparable to the dose reference values specified in 10 CFR Part 100. Most of the currently licensed BWRs rely on the MSIV leakage control system to mitigate the radiological consequences of MSIV leakage following a design-basis loss-of-coolant accident (LOCA) and to stay within 10 CFR Part 100 limits if the MSIV leakage rate exceeded the technical specification limit of 0.3 m^3/hr (11.5 ft^3/hr). The ABWR will not have an MSIV leakage control system and, therefore, will rely on the MS system coupled with the main condenser (MC) to contain MSIV leakage, relying on plateout and holdup of radioactive iodine and to limit the radiological consequences to within 10 CFR Part 100 limits.

In the DFSER, the staff stated that in order to take credit for the MSSS and MC for containment and holdup of MSIV leakage, the MSSS and the MC and connections from the MSLs to the condenser must be capable of maintaining their integrity during and following an SSE. Subsequently, GE added SSAR Subsection 3.2.5.3 which clarified the seismic requirements for the MSL leakage paths. Section 3.2 of this report contains a detailed evaluation of the seismic analysis requirements for the MSSS and the MC. To process the MSIV leakage through the MC, a leakage path must be ensured either through the MS drain line to the condenser or through the turbine bypass system (TBS) to the condenser. In the DFSER, the staff stated that, whichever of these two paths is chosen, reliable power sources must be available so that a control operator can establish the flow path assuming a single active failure. The staff stated they would review this issue on a case-by-case basis for each ABWR COL applicant. GE subsequently stated that the MS drain lines have parallel motor operated and air operated valves to ensure a leakage path to the MC. The motor operated valves are powered from their respective Class 1E bus and the air operated valves fail open on loss of pneumatic pressure or on loss of power to the actuating solenoid. Additionally, the turbine bypass valves are hydraulically operated and powered by redundant uninterruptable non-Class 1E power supplies. Therefore, the staff concludes that reliable power sources are available to establish the flow path.

In the DFSER, the staff stated that the amount of allowable MSIV leakage will also be reviewed for each ABWR COL applicant. This was identified as DFSER COL Action Item 10.3.1-1. Subsequently, the staff reviewed the final SSAR and concluded that the modifications to SSAR Subsections 10.3.2 and 10.3.7 will ensure that the COL applicant referencing the ABWR design will provide the amount of allowable MSIV leakage to the staff for review. This is acceptable to the staff.

The steam lines in the reactor building (including the containment and some portion of the steam tunnel) and in the steam tunnel portion in the control building, are located in seismic Category I, flood-protected and tornado-protected structures. Thus, these portions of the MSSS meet the requirements of GDC 2 and the guidelines of RG 1.29, "Seismic Design Classification," (Rev. 3) Positions C.1 and C.2.

In the DFSER, the staff explained that SSAR Appendix 3F stated that since the safety-related portions of the condensate and feedwater system (CFS) and the MSSS (from the reactor up to and including the seismic interface restraints) are qualified for the leak-before-break (LBB) criterion, a high-energy pipe break does not need to be considered in those portions of the above systems solely for considering the local dynamic effects associated with such breaks. In SSAR Appendix 3F, GE provided generic LBB evaluation procedures and methodology for the systems to support this claim. In Section 3.6.3 of this report, the staff states that a LBB analysis should use plant-specific data such as piping geometry, materials, fabrication procedures, loads, degradation mechanisms, and pipe support locations. Therefore, the staff will evaluate the acceptability of the LBB methodology on a plant-specific basis to determine if the essential portion of the MSSS will be adequately protected against dynamic effects associated with high-energy pipe breaks. In a meeting with the staff on May 5, 1992, GE committed to remove references to LBB from the SSAR. This was identified as DFSER Confirmatory Item 10.3.1-1. Subsequently, the staff reviewed the final SSAR and found that GE had deleted Appendix 3F from the SSAR as agreed. GE has also removed this information from the SSAR. Therefore, Confirmatory Item 10.3.1-1 is resolved. GE has provided, in SSAR Section 6.2.3, an analysis of a MSL and main feedwater line pipe failure inside the MS tunnel. The results of the staff's review of this analysis are contained in Section 6.2.1.7 of this report. Features to protect the MSIVs from the effects of a pipe failure inside the main steam tunnel (MST) are evaluated in Section 3.6.1 of this report.

Regarding the other aspect of GDC 4, which deals with the environmental design basis for SSCs important to safety, GE states in SSAR Appendix 3I that the essential equipment of the system is environmentally qualified to function following a postulated high-energy pipe break. Specifically, this means that the MSIVs will be required to function to ensure MSL isolation and will be qualified to function in the expected steam environment resulting from a steamline break. Further, GE identified an interface requirement for the COL applicant to provide any additional protective features (e.g., shields and other barriers) that may be needed to protect the MSIV functional capability against the effects of postulated pipe failures. On further evaluation, the staff determined that this requirement can be accomplished by identifying a COL action item in the SSAR requiring the applicant to provide this information. This was identified as DFSER Confirmatory Item 10.3.1-2. GE has included this information in SSAR Section 3.6.5.1. Therefore, Confirmatory Item 10.3.1-2 is resolved.

GE addressed the issue of steam hammer and relief valve discharge loads in a submittal dated February 28, 1990, and stating that the system design accommodates steam hammer and relief valve discharge loads. In the DFSER, the staff stated that the staff would require the COL applicant to have operating and maintenance procedures that include adequate precautions to avoid steam hammer and relief valve discharge loads. This was identified as DFSER COL Action Item 10.3.1-2. Subsequently, the staff reviewed the final SSAR and concluded that the modifications to SSAR Subsections 10.3.3 and 10.3.7 provided adequate assurance that the applicant referencing the ABWR design will provide the necessary procedures to assure that steam hammer and relief valve discharge loads will be minimized. This is acceptable to the staff.

The system design includes drains to protect against water entrainment. The essential equipment of the system is located in tornado-missile-protected structures (as stated above) and is protected from the effects of internally generated missiles. The appendix to RG 1.117, "Tornado Design Classification," (Rev. 1), specifies SSCs of light-water-cooled reactors that should be protected against tornados. On this basis, the staff finds that the safety-related portion of the system meets the requirements of GDC 4 and the guidelines of RG 1.117 (Rev. 1), Appendix Position 4.

Because the ABWR is designed as a single-unit facility, the requirements of GDC 5 do not apply.

GE submitted the design description and the ITAAC relating to the MSSS. This was identified as DFSER Open Item 10.3.1-1. GE provided a revised set of design descriptions and ITAAC. The adequacy and acceptability of the final certified ABWR design description and the ITAAC are evaluated in Section 14.3 of this report. On the basis of this evaluation, this item is resolved.

The MSSS includes all components and piping from the outermost containment isolation valve up to and including the turbine stop valves. The safety-related portions of the system are designed to QG B from the outermost containment isolation valve and connecting piping up to and including the first normally closed valve. Those portions of the MSSS necessary to mitigate the consequences of an accident are designed to the quality standards commensurate with the importance to its safety function. The scope of the review included layout drawings, P&IDs, and descriptive information for the system.

Based on the above discussion, the staff concludes that the MSSS for the ABWR from the reactor to the TB satisfies the requirements of GDC 2, 4, and 5 and the guidelines of RG 1.29 (Rev. 3), Positions C.1 and C.2, and RG 1.117 (Rev. 1), Appendix Position 4; meets SRP Section 10.3 acceptance criteria; and is acceptable.

10.3.2 Steam and Feedwater System Materials

GDC 1 requires that systems important to safety shall be designed to quality standards commensurate with the importance to safety of the functions to be performed. GDC 35 requires suitable inter-connections, leak detection, isolation, and contaminant capabilities be provided to assure that the safety system function (i.e., emergency core cooling) can be accomplished, assumed a single failure.

The staff reviewed the steam and feedwater system materials in accordance with SRP Section 10.3.6. The steam and feedwater system materials are acceptable if they satisfy the requirements of the ASME Code, Section III and GDC 1 and 35.

The Class 2 materials specified in the SSAR for the MS and feedwater system satisfy the requirements specified in Appendix I to Section III of the ASME Code, and Parts A, B, and C of Section II of the Code. The fracture toughness properties of the materials meet the requirements of NC-2300 of ASME Code Section III and RG 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive Waste-Containing Components of Nuclear Power Plants" (Rev. 3).

The materials selection and fabrication follow RG 1.71, "Welder Qualification for Areas of Limit Accessibility" (original), RG 1.85, "Materials Code Case Acceptability-ASME Section III, Division 1" (Rev. 28), RG 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants" (original) and ANSI N45.2.1. The non-destructive examination of the steam and feedwater piping meets the acceptance criteria in NC-2550 through NC-2570 of ASME Code, Section III.

Compliance with the requirements of the ASME Code, ANSI standard, and regulatory guides satisfy the applicable requirements of GDC 1 and 35 and Appendix B to 10 CFR Part 50. The staff concludes that the MS and feedwater system materials are acceptable and meet the relevant requirements of 10 CFR 50.55a, GDC 1 and 35, and Appendix B to 10 CFR Part 50.

10.4 Other Features

10.4.1 Main Condenser

The staff reviewed the MC in accordance with SRP Section 10.4.1. The acceptability of the system design is based on its meeting the requirements of GDC 60 as it relates to failure of the system not result in excessive releases of radioactivity to the environment, not cause unacceptable condensate quality, and not flood areas housing safety-related equipment.

The MC is designed to function as a steam cycle heat sink. The MC will receive, condense, and deaerate the turbine exhaust steam and the turbine bypass steam. The MC will also collect miscellaneous steam cycle drains and vents as well as transfer heat to the circulating water system (CWS), which, in turn, will reject the heat to the PCHS. GE states in SSAR Section 10.4.1.2.2 that nothing precludes the use of a single pressure condenser and a parallel (instead of series) circulating water system since these will have no affect on the nuclear TS conditions.

The MC will not be required to serve or support any reactor safety function. However, because there is no MSIV leakage control system, the MSLs and condenser will be used to collect MSIV leakage following a LOCA. Therefore, the MC must be capable of maintaining its integrity following a SSE. The condenser supports and anchorages will be seismically analyzed to demonstrate that they are capable of sustaining the SSE loading conditions without failure (Section 3.2 of this report contains additional discussion and evaluation of the capability of the MC to meet this requirement.)

The MC consists of three multi-pressure, at least two-tube bundle, single-pass shells. Each of the shells is located under its respective low-pressure turbine. The MC hotwell is sized and designed to retain all condensate for 4 minutes from the time it enters the hotwell until it is removed by the condensate pumps. Condensate will be retained in the condenser for a minimum of 2 minutes to permit radioactive decay (primarily of nitrogen-16) before it enters the condensate system. Offgas from the MC will be processed in the gaseous waste management system, which is described in Section 11.3 of this report. The MC is designed to (1) deaerate the condensate, (2) remove air and noncondensable gases, and (3) remove hydrogen and oxygen formed in the steam.

Circulating water on the tube side of the MC will be treated with chemicals to limit algae growth and to minimize long-term corrosion of the tubes. Corrosion on the shell side of the condenser will be controlled by adhering to strict water quality. The construction materials used for the MC will be chosen so that corrosion as a result of galvanic and other effects can be kept to a minimum.

Condenser leakage will be in-leakage since the MC will normally be operated under vacuum. Tube leakage will be monitored by measuring the conductivity of water samples taken beneath the tube bundles. Additionally, since the condensate will be monitored at the condensate pump discharge, any tube leakage will also be detected at this monitoring point. Conductivity of the condensate will be continuously monitored at selected locations in the condenser. Condenser vacuum will also be monitored. The loss of the MC vacuum will cause a turbine trip and MSIV closure. An alarm actuates at -81 kPa at 0 °C (24 in. Hg vacuum at 32 °F), and a high condenser pressure turbine trip will occur at -75 kPa at 0 °C (22 in. Hg vacuum at 32 °F). Additionally, MSIV closure occurs at -24 to -34 kPa at 0 °C (7 to 10 in. Hg vacuum at 32 °F) while turbine bypass valve closure will take place at -41 kPa at 0 °C (12 in. Hg vacuum at 32 °F).

The MC is designed to condense at least 33 percent of the full-rated turbine steam flow as bypass steam.

During the initial phase of startup, the mechanical vacuum pump establishes a vacuum in the MC. The discharge from the vacuum pump is routed to the Turbine Building Compartment Exhaust System. Radiation monitors in the TBCE and plant vent alarm in the MCR if abnormal radioactivity is detected. In addition, radiation monitors are provided on the main steam lines which trip the vacuum pump if abnormal radioactivity is detected in the steam being supplied to the condenser. This is discussed in Section 10.4.2 of the SSAR and evaluated in Sections 10.4.2 and 11.5 of this report.

The low-pressure turbine exhaust and the MC will be connected by a stainless steel expansion joint. Since no safety-related equipment is located in the condenser area, failure of the joint will have no adverse effect on safety-related equipment. Protection of safety-related equipment from flooding in the TB is reviewed in Section 3.4.1 of this report.

Based on this information, the staff concludes that the main condenser design includes provisions which assure that failures of the system will not result in excessive releases of radioactivity to the environment, do not cause unacceptable condensate quality, or result in flooding of areas housing safety-related equipment and therefore meets the requirements of GDC 60.

GE submitted the design description and the ITAAC relating to the MC system. This was identified as DFSER Open Item 10.4.1-1. GE provided a revised set of design descriptions and ITAAC. The adequacy and acceptability of the final certified ABWR design description and the ITAAC are evaluated in Section 14.3 of this report. On the basis of this evaluation, this item is resolved.

The MC includes all components and equipment from the turbine exhaust to the connections with the CFS and other systems. The scope of the review of the MC system included layout drawings, P&IDs, and descriptive information for the MC system and supporting systems that are essential to its operation.

The basis for acceptance of the MC system was conformance of the design, design criteria, and design bases to the Commission's regulations as set forth in GDC 60. The staff concludes that the MC system design is acceptable and meets the requirements of GDC 60 with respect to failures not resulting in excessive releases of radioactivity to the environment, not causing unacceptable condensate quality, and not flooding areas housing safety-related equipment.

The staff concludes that the design of the MC is in conformance with SRP Section 10.4.1, can perform its design function, and is acceptable.

10.4.2 Main Condenser Evacuation System

The staff reviewed the main condenser evacuation system (MCES) in accordance with the acceptance criteria in SRP Section 10.4.2, and guidelines contained in RG 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," (Rev. 3); RG 1.33, "Quality Assurance Program Requirements (Operation)," (Rev. 2); and in the Heat Exchanger Institute's "Standards for Steam Surface Condensers," 6th Edition (1970). To be acceptable, the MCES must meet the requirements of GDC 60 for controlling releases of radioactive materials to the environment and the requirements of GDC 64 for monitoring the release of radioactive material to the environment.

The MCES is designed to establish and maintain condenser vacuum by removing noncondensable gases from the MC and directing them to the offgas system for processing before release through the plant stack during normal plant operation. The MCES will not perform or support any safety function.

The MCES is designed to QG D standards and consists of a mechanical vacuum pump for use during startup, and two 100-percent capacity, double stage, steam jet air ejector (SJAE) units (complete with intercondenser) for normal operating conditions. During startup, the mechanical vacuum pump will be used to establish a vacuum in the MC and the exhaust gas will be vented to the turbine building (TB) compartment exhaust system. High radioactivity in the MSL will trip the mechanical vacuum pump. The TB compartment exhaust will pass through a medium efficiency filter and be monitored for radioactivity before discharge to the plant vent. After the mechanical vacuum pump has created an absolute pressure of about -34 to -50 kPa at 0 °C (10 to 15 in. Hg at 32 °F) in the MC and adequate nuclear steam pressure is available, one of the two SJAEs will be put in service to remove noncondensable gases from the condenser.

Steam supply to the second stage of the SJAE will be kept at a minimum predetermined flow to help ensure adequate dilution of hydrogen (below 4 percent by volume) to prevent the hydrogen in the offgas system from reaching a flammable concentration. Low flow of the dilution steam will result in automatic isolation of the MC from the offgas system.

GE submitted the design description and the ITAAC relating to the MCES. This was identified as DFSER Open Item 10.4.2-1. GE provided a revised set of design descriptions and ITAAC. The adequacy and acceptability of the design description and the ITAAC are evaluated in Section 14.3 of this report. On the basis of this evaluation, this item is resolved.

The MCES includes equipment and instruments to establish and maintain condenser vacuum and to prevent an uncontrolled release of radioactive material to the environment. The scope of the review included the system capability to transfer radioactive gases to the offgas system and the design provisions incorporated to monitor and control releases of radioactive materials in effluents. The staff has reviewed the applicant's system descriptions, P&IDs, and design criteria for the components of the MCES.

The staff concludes that the MCES design is acceptable and meets the requirements of GDC 60 and 64 and the guidelines of SRP Section 10.4.2 for controlling and monitoring releases of radioactive material to the environment.

10.4.3 Turbine Gland Sealing System

The staff reviewed the TGSS in accordance with SRP Section 10.4.3. Acceptance is based on TGSS meeting the requirements of GDC 60 for controlling the releases of radioactive materials to the environment and the requirements of GDC 64 for monitoring the release of radioactive material to the environment.

The TGSS is designed to prevent release to the TB of radioactive steam from the turbine shaft/casing penetrations and valve stems and to prevent air leakage into the steam cycle via the subatmospheric turbine glands. TGSS will be accomplished by providing a continuous supply of relatively clean (i.e., practically free of radioactivity) sealing steam to the turbine shaft seals and the steam packings of the stop valves, control valves, and combined intermediate and bypass valves. The TGSS will not perform or support any safety function. The TGSS will consist of a sealing steam pressure regulator, a sealing steam header, a gland steam condenser, and two 100-percent capacity, motor-driven blowers. The system is designed to QG D standards.

The annular space between the turbine shaft and the casing will be sealed with sealing steam supplied to the shaft seals. At all gland seals, the vent annulus will be kept under a slight vacuum condition and also will receive outside air as in-leakage. The steam mixture from the vent annulus then will be pulled to the gland steam condenser, which will be operated under a slight vacuum condition created by one of the two exhaust blowers. The steam mixture will be condensed in the gland steam condenser and the condensate will be returned to the MC. The blower is designed to discharge the air in-leakage and the noncondensable gases to the TB compartment exhaust system, which eventually will discharge to the plant vent. As mentioned above, the TGSS is also designed to provide sealing steam to the turbine stop and control valves and combined intermediate valve packings. The staff stated Open Item 67 in DSER (SECY-91-235) that SSAR Table 11.5-2 did not indicate any separate process radiation monitoring solely for TGSS exhaust. GE addressed this issue by including monitoring and sampling provisions for the turbine gland seal exhausts in SSAR Tables 11.5-1 and 11.5-7. The staff reviewed these submittals and found them acceptable. This item was resolved in the DFSER.

During startup, sealing steam will be provided from the MSL or the plant auxiliary steam header. The use of MS as sealing steam will not pose a significant long-term average release of radioactive material to the environment because the startup time will be relatively short and plant startup radioactivity is relatively low. If the MS sealing steam supply has an abnormally high radioactivity content, the sealing steam supply can be switched to the plant auxiliary steam header, which contains clean steam (i.e., radioactivity free) from a conventional auxiliary boiler.

During normal operation (above approximately 50-percent load), process steam will be used for the sealing steam supply. The high-pressure heater drain tank vent header, which is designed to provide relatively clean steam for turbine gland sealing, will provide the process steam. Again, if this normal source of sealing steam is observed to have high radioactivity content, the source for the sealing system will be switched to the plant auxiliary steam system, which provides 100 percent backup capability. Thus, the long-term average amount of radioactive material released to the environment should be minimal. In the DSER (SECY-91-235), the staff agreed with this approach for providing sealing steam, subject to the identification of an interface requirement to provide procedures for the switchover to the plant auxiliary steam. In response to Open Item 67 in the DSER (SECY-91-235), GE included in SSAR Section 10.4.10 that COL applicants will provide the necessary procedures for switchover to the auxiliary steam system if the monitored radiation level in the gland sealing system exhaust exceeded an acceptable preset level. The staff agreed with GE's approach for providing sealing steam for the turbine gland seals. However, upon further evaluation, the staff determined that this requirement could be accomplished by identifying a COL action item in the SSAR requiring the applicant to provide the necessary procedures. This was identified as DFSER COL Action Item 10.4.3-1. GE has included this action item in the SSAR. This is acceptable to the staff.

GE submitted the design description and the ITAAC relating to the TGSS. This was identified as DFSER Open Item 10.4.3-1. GE provided a revised set of design descriptions and ITAAC. The adequacy and acceptability of the final certified ABWR design description and the ITAAC are evaluated in Section 14.3 of this report. On the basis of this evaluation, this item is resolved.

The TGSS includes the equipment and instruments to provide a source of sealing steam to the annulus space where the turbine and large steam valve shafts penetrate their casings. The scope of the review included the source of sealing steam and the provisions incorporated to monitor and control releases of radioactive material in effluents.

The staff concludes that the TGSS is acceptable because it meets the requirements of GDC 60 and 64 for controlling and monitoring releases of radioactive material to the environment. Therefore, the system meets the guidelines of SRP 10.4.3 and is acceptable.

10.4.4 Turbine Bypass System

The staff reviewed the TBS in accordance with SRP Section 10.4.4. The design is acceptable if, in accordance with GDC 4, failure of the system (due to a pipe break or system malfunction) does not adversely affect safety-related systems or components and if, in accordance with GDC 34, the system can shut down the plant during normal operations. Use of this system will eliminate the need to rely solely on safety systems.

The TBS is designed to bypass at least 33 percent of the rated MS flow to the MC. It is also designed to bypass steam to the MC during plant startup and to permit a normal manual cooldown of the reactor coolant system (RCS) from hot shutdown to the point at which the residual heat removal function can be placed in service. In addition, during a power operation transient (i.e., when steam produced by the reactor cannot be entirely used by the turbine), the TBS, in conjunction with the RCS, will allow a step load reduction up to 40 percent of the turbine generator rated electrical load without causing a reactor trip. TBS will also allow a turbine trip or a full load rejection from 100 percent power with reactor trip, without lifting the MS relief and safety valves. Thus, the TBS minimizes step-load reduction transient effects as well as turbine trip effects on the RCS.

The TBS consists of three control valves that are housed in a common valve chest connected to the MSLs upstream of the turbine stop valves and three dump lines that separately connect each regulating valve outlet to one condenser shell. Each bypass valve is operated by hydraulic fluid pressure with spring action to close. The valve chest assembly will include hydraulic supply and drain piping, hydraulic accumulators, servo valves, fast-acting servo valves, and position transmitters. Each bypass valve is operated by the turbine hydraulic fluid power unit or it may be provided with a separate hydraulic fluid power unit.

The bypass valves are designed to open whenever the actual steam pressure exceeds the preset steam pressure. Fast-acting servo valves will be used to allow the bypass valves to open rapidly in case a turbine trip or generator load rejection occurs. The turbine bypass valves are designed to trip closed on loss of MC vacuum, loss of electrical power, or loss of hydraulic system pressure.

The TBS is designed to be tested during operation. Periodic inspections will be performed on a rotating basis within a preventive maintenance program recommended by the manufacturer. As stated in GE's submittal dated February 28, 1990, the detailed design of the bypass valves will follow standard industry practice and reduce the bypassed steam pressure sequentially through orifices before the steam enters the condenser.

The TBS will not serve or support any safety function. No safety-related equipment is located inside the TB. All high-energy lines associated with the TBS are located in the TB. Therefore, failure of the TBS are any safety-related equipment or hamper the capability for safe shutdown of the plant.

Although the TBS will not be required to serve or support any reactor safety function, it will have a post-LOCA function for the ABWR. In the absence of an MSIV leakage control system, the MSLs and condenser will be used to collect MSIV leakage following a LOCA. Therefore, the TBS must be capable of maintaining its integrity following a SSE. The turbine bypass line from the bypass valve to the condenser will be seismically analyzed to demonstrate that it is capable of sustaining the SSE loading conditions without failure (Section 3.2 of this report contains additional discussion and evaluation of the capability of the turbine bypass piping to meet this requirement.)

GE submitted the design description and the ITAAC relating to the TBS. This was identified as DFSER Open Item 10.4.4-1. GE provided a revised set of design descriptions and ITAAC. The adequacy and acceptability of the final certified ABWR design description and the ITAAC are evaluated in Section 14.3 of this report. On the basis of this evaluation, this item is resolved.

The TBS includes all components and piping from the branch connection at the MS to the MC. The scope of the review included layout drawings, P&IDs and descriptive information for the TBS.

The basis for acceptance was conformance of the design, design criteria, and design bases to the Commission's regulations as set forth in GDC 4 and 34. The TBS has met the requirements of GDC 4 with respect to the system being designed such that a safe shutdown will not be precluded as a result of a TBS failure. The system has also met the requirements GDC 34 with respect to the ability to use the TBS for shutting down the plant during normal operations.

The staff concludes that the design of the TBS meets the guidelines of SRP Section 10.4.4 and is acceptable.

10.4.5 Circulating Water System

The staff reviewed the CWS in accordance with SRP Section 10.4.5. The design of the CWS is acceptable if, in accordance with GDC 4, the system can accommodate the effects water that may be discharged because a component or piping in the CWS fails.

The CWS is partially within the ABWR scope. The in-scope portion of the system includes all piping, valves, instrumentation, and controls within the TB. All other equipment outside the TB, including the CWS pumps is outside the ABWR scope and is the responsibility of the COL applicant.

GE has provided a conceptual design and interface requirements for that portion of the CWS outside the scope of the ABWR design, as required by 10 CFR Part 52.

The CWS is designed to remove the power cycle water heat from the MC and transfer this heat to the PCHS. The CWS will not be required to maintain the reactor in a safe shutdown condition or support any safety-related systems or components. The system is nonseismic and QG D.

The CWS consists of at least three fixed-speed, motor-driven pumps for circulating water throughout the system, screenhouse and intake screens, condenser water boxes, piping and valves, water box fill and drain subsystem, and general support facilities. A chemical addition subsystem minimizes biological buildup and chemical deposits within the system.

The CWS pumps will be vertical, wet pit-type, capable of delivering approximately 45,430 m^3/hr (200,000 gpm) per pump. The discharge line of each pump will be equipped with a butterfly valve to allow isolation and maintenance of any one pump while the others are in operation. The CWS pumps will be tripped and the pump and the condenser isolation valves closed on a high-high level condenser pit signal. A condenser pit high-level alarm will actuate in the control room.

The PCHS will be designed to maintain the temperature of the water entering the CWS within the range of 0 °C to 38 °C (32 °F to 100 °F). The CWS is designed to deliver water to the MC within a temperature range of 4 °C to 38 °C (39 °F to 100 °F). The 4 °C (39 °F) minimum temperature will be maintained by recirculating warm water from the discharge side of the condenser back to the screenhouse.

In the DFSER, the staff stated that GE had not included an analysis of flooding in the TB and could affect safety-related equipment. The staff requested GE to provide a flood analysis which characterized the nature of the hazards and the design features to protect safety-related equipment from flooding in the TB. This was identified as DFSER Open Item 3.4.1-1. Subsequently, GE provided a flood analysis for the TB. The major flood hazard in the TB is from a failure in the CWS, which is an open-cycle system. Leak detectors in the condenser pit will alert the control room and automatically isolate the CWS on indication of building flooding. A postulated failure of the isolation function can result in flooding of the TB up to grade level. A non-watertight truck door at grade level will allow release of the flood water onto the ground. As stated in Section 3.4.1 of this report, the below-plant-grade tunnel connecting the radwaste, reactor, and TB is sealed to prevent water from entering the reactor building. The staff finds that the design adequately protects safety-related SSCs from the effects of flooding as a result of pipe failures in the CWS. GE has also included this information in the SSAR. Open Item 3.4.1-1 is resolved.

CWS performance will be monitored by temperature and pressure indicators in the main control room. CWS-related valve positions also will be indicated in the control room.

GE submitted the design description and the ITAAC relating to the CWS. This was identified as DFSER Open Item 10.4.5-1. GE provided a revised set of design descriptions and ITAAC. The adequacy and acceptability of the final certified ABWR design description and the ITAAC are evaluated in Section 14.3 of this report. On the basis of this evaluation, Open Item 10.4.5-1 is resolved.

The CWS includes all components and equipment necessary to provide the MC with a continuous supply of cooling water. The system is designed to nonnuclear safety and QG D requirements. Based on the review of the applicant's proposed design criteria and bases for the CWS, the staff concludes that the design is acceptable and meets the requirements of GDC 4 and the guidelines of SRP Section 10.4.5.

10.4.6 Condensate Purification System

The condensate cleanup system (CCS) will remove dissolved and suspended solids from the condensate to maintain a high quality of feedwater to the reactor under all normal plant conditions (startup, shutdown, hot standby, and power operation). The CCS will accomplish this task by directing the full flow of condensate to five of the six polishing vessels, which will be piped in parallel. The sixth polisher will be on standby or in the process of being cleaned, emptied, or refilled. The six polishing vessels will contain mixed bed ion exchange resin with a strainer installed downstream of each vessel. The strainers will be used to prevent gross resin leakage into the feed system in the event of vessel underdrain failure and to minimize resin fine leakage. The CCS will include all components and equipment needed to remove dissolved and suspended impurities present in the condensate.

The staff has reviewed the design of the sampling equipment, sampling locations, and instrumentation to monitor and control the CCS parameters and finds the design acceptable. However, in the DSER (SECY-91-355), the staff identified that SSAR Section 10.4.6 contained insufficient information for the staff to evaluate conformance with SRP Section 10.4.6 in the following areas:

- Under SRP Section 3.6.1, the effects of high and moderate energy piping failures to assure that other safety-related systems are not rendered inoperable must be evaluated.
- Under SRP Section 12.2, the adequacy of the shielding design of the CCS polisher vessels must be evaluated.
- Although the SSAR indicates conformance with RG 1.56, "Maintenance of Water Purity in Boiling Water Reactors," to meet the requirements of GDC 14 and to mitigate the potential of intergranular stress corrosion cracking, GE should state conformance with EPRI NP-4947-SR, "BWR Hydrogen Water Chemistry Guidelines," (1987 Revision, October 1988).
- GE should state that the CCS will remove condensate system corrosion products and impurities from condenser leakage in addition to radioactive material, activated corrosion products, and fission products carried over from the reactor.

The staff concluded in the DSER that the adequacy of the CCS was Open Item 105 in DSER (SECY-91-355).

The effects of high and moderate energy piping failures on safety-related equipment are discussed in Section 3.6.1 of this report. However, as previously stated, the TB contains no safety-related equipment and, therefore, safety-related systems should not be affected by a CCS piping failure. This item is resolved.

The adequacy of the shielding design of the CCS polisher vessels will be evaluated as part of the radiation protection DAC discussed in Chapter 12 of this report. This item is resolved.

GE responded to the last two parts of this open item in its letter of March 11, 1992. GE revised SSAR Section 10.4.6.3 to state that condensate system corrosion products and impurities from condenser leakage will also be removed and that the CCS will comply with EPRI NP-4947-SR. The NRC staff finds this response acceptable; therefore, this DSER (SECY-91-355) issue number 105 is resolved. The staff concludes that the design of the CCS and its supporting systems conforms to staff guidelines and is acceptable.

10.4.7 Condensate and Feedwater System

The staff reviewed the condensate and feedwater system (CFS) in accordance with SRP Section 10.4.7. Acceptance of the system is based on the system's meeting the requirements of GDC 2 that the system can withstand the effects of earthquakes, the requirements of GDC 4 that the system be protected against dynamic effects associated with fluid flow instabilities during normal operation as well as during upset or accident conditions, the requirements of GDC 5 that shared systems and components important to safety can perform required safety functions, the requirements of GDC 44 that the system can reliably transfer heat loads from the reactor to a heat sink during both normal and accident conditions and can isolate components, subsystems, or piping if necessary to maintain the system safety function, and GDC 45 and 46 as they relate to ISI and testing, respectively.

The CFS consists of the piping, valves, pumps, heat exchangers, and associated controls and instrumentation that extends from the MC outlet to the nuclear boiling (NB) system at the seismic interface restraint and to the heater drain system. The system is designed to receive condensate from the MC hotwell; supply condensate to the condensate purification cleanup system; supply cooling water to the gland steam exhauster, SJAE, and offgas recombiner coolers; and deliver high-purity feedwater to the reactor at the required flow rate, pressure, and temperature. The major equipment in the CFS includes: (1) four identical fixed-speed, motordriven condensate pumps, of which three are normally operating and one is on standby; (2) three identical and independent, 33-65 percent capacity variable speed motor-driven reactor feed pumps; (3) three parallel and independent trains of four closed, low-pressure feedwater heaters; (4) two parallel and independent trains of two high-pressure feedwater heaters; (5) two heater drain tanks; and (6) two independent, motor-driven heater drain pumps that take suction from a heater drain tank and discharge into the suction side of the feedwater pumps. The CFS is described in SSAR Sections 5.4.9 and 10.4.7.

The CFS flow begins at the MC hotwell. Three normally operated condensate pumps take suction from the hotwell and pump condensate through the condensate filters and demineralizers. The condensate is discharged into a common header that feeds five parallel auxiliary condenser coolers (one gland steam exhauster condenser, two SJAE condensers, and two offgas recombiner coolers). The condensate then flows to three parallel trains of low-pressure feedwater heaters that discharge into a common header routed to three reactor feedwater pumps arranged in parallel. The reactor feedwater pumps then discharge into two parallel high-pressure feedwater heater trains. Downstream of the high-pressure feedwater heaters, the feedwater is combined into a common header that discharges into the reactor through two parallel 56-cm (22-in.) nominal diameter feedwater lines, as stated in SSAR Section 5.4.9.3.

On each of the feedwater lines from the common feedwater header to the reactor, there is a seismic interface restraint. A remote manual motor-operated gate valve powered by a non-safety-grade bus serves as a feedwater shutoff valve. Downstream of this motor-operated gate valve, there is a spring-closing check valve that is held open by air and serves as the outboard containment isolation valve. On the other side of the containment, a check valve serves as the inboard containment isolation valve, and downstream of this check valve is a manual maintenance valve. However, the staff stated Open Item 69 in the DSER (SECY-91-235) that the provision of a non-safety-grade power source for the remote manual shutoff gate valve was inappropriate because the valve and the portion of piping in which it is located are designed as seismic Category I and the valve serves as a long-term isolation for the containment. In a meeting with the staff on May 5, 1992, GE stated that insights from the probabilistic risk assessment (Chapter 19 of the SSAR) indicate that the ability to open the valve using diverse non-safety-grade on-site power instead of safety-related power to initiate feedwater flow reduces the risk. Furthermore, this valve is not relied upon as a long-term leakage barrier. Instead, GE has provided high reliability check valves. Additionally, the spring-closing check valves are testable and provide a positive means of isolation. Therefore, the use of diverse non-safety-grade power for the manual shutoff gate valve was acceptable subject to GE providing documentation of the information discussed in the May 5, 1992, meeting. This information was identified as DFSER Confirmatory Item 10.4.7-1. Subsequently, the staff reviewed the final SSAR and concluded that the modification to Subsection 5.4.9.3 regarding the power sources for the shutoff valves was acceptable. This is acceptable to the staff, and therefore, Confirmatory Item 10.4.7-1 is resolved.

As indicated in SSAR Section 5.4.9 and Table 3.2-1, the feedwater piping is QG A from the reactor pressure vessel out to and including the outboard isolation valve, QG B from the outboard isolation valve up to and including the shutoff valve and QG D beyond the shutoff valve. The feedwater piping and all connected piping of 6.5 cm (2-½ in.) or larger nominal size, is seismic Category I from the reactor pressure vessel out to and including the seismic interface restraint. The safety-related equipment is physically separated and protected against the effects of internally generated missiles. The staff concludes that the design of the CFS meets the requirements of GDC 2 and the guidelines of RG 1.29, Positions C.1 and C.2.

In a meeting with the staff on May 5, 1992, GE committed to remove references to LBB from the SSAR. This commitment was identified as DFSER Confirmatory Item 10.3.1-1. In SSAR Section 6.2.3, GE has analyzed a MSL and main feedwater line pipe failure inside the MST and has removed Appendix 3F from the SSAR. The results of the staff's review of this analysis are found in Section 6.2.1.7 of this report. This is acceptable to the staff, and therefore, Confirmatory Item 10.3.1-1 is resolved.

Regarding the other aspect of GDC 4, that deals with the environmental design basis for SSC important to safety, SSAR Appendix 3I states that the essential equipment of the system is environmentally qualified to function following a postulated high-energy pipe break. GE has also addressed the issue of water-hammer loads as a result of hydraulic transients that can occur when feedwater control valves rapidly interrupt feedwater flow (submittal dated February 28, 1990). The ABWR design uses a modified CFS design that has only a low feedwater flow control valve specifically designed to minimize cycling in feedwater nozzles. The valve is used only at low-power operating conditions. During normal power operations, feedwater flow is varied as needed by using adjustable speed, motor-driven feed pumps, thus eliminating the need for any flow control valve. As discussed in Section 10.3.1 of this report, the staff will require COL applicants to provide operating and maintenance procedures to ensure that water hammer and its effects are avoided or minimized. Finally, the system piping is analyzed for loads from anticipated flow transients. On this basis, the staff finds that the CFS design complies with GDC 4. Protection of safety-related equipment from flooding in the TB is discussed in Section 3.4.1 of this report.

Because the ABWR is designed as a single-unit facility, the requirements of GDC 5 do not apply.

The system is sized to provide adequate flow to the reactor. The system contains parallel trains that will allow for isolation, inspection, and testing during normal operation. Therefore, GDC 44, 45, and 46 are met.

GE submitted the design description and the ITAAC relating to the CFS. This was identified in the DFSER as Open Item 10.4.7-1.

GE provided a revised set of design descriptions and ITAAC. The adequacy and acceptability of the final certified ABWR design description and the ITAAC are evaluated in Section 14.3 of this report. On the basis of this evaluation, this item is resolved.

The CFS includes all components and equipment from the condenser outlet to the connection with the NB system and heater drain system. Based on the review of the applicant's proposed design criteria, design bases, and safety classification for the system, the staff concludes that the design of the CFS and supporting systems conforms with GDC 2, 4, 5, 44, 45, and 46 and with the guidelines of SRP Section 10.4.7 and is, therefore, acceptable.

10.4.8 Power Cycle Heat Sink

The staff reviewed the conceptual design and the design interface requirements for the PCHS. The PCHS was included in SSAR Section 10.4.5.8 and was reviewed in accordance with SRP Section 9.2.5. Acceptance of the design is based on meeting GDC 2 as it relates to structures housing the system and the system itself being capable of withstanding the effects of natural phenomena, GDC 5 as it relates to shared systems between units, GDC 44 as it relates to the capability to transfer heat loads from safety-related SSCs to the heat sink under normal and accident conditions as well as providing suitable redundancy of components to ensure adequate safety function given a single active component failure and the capability to isolate parts of the system so that the safety function is not compromised, and GDC 45 and 46 as they relate to ISI and operational functional testing, respectively, of safety-related systems and components.

GE designated the PCHS as being outside the scope of the ABWR design. GE has provided a conceptual design and interface criteria, as required by 10 CFR Part 52, to allow an applicant referencing the ABWR to provide a plant-specific PCHS design capable of dissipating turbine plant heat. The PCHS will be designed to accept the heat loads of the turbine service water (TSW) system (Section 9.2.16 of this report) and the CWS (Section 10.4.5 of this report). The TSW in turn accepts the heat loads of the turbine building cooling water (TCW) system (Section 9.2.14 of this report) while the CWS accepts the heat load from the MC (Section 10.4.1 of this report).

GE has provided a conceptual design for the PCHS. The PCHS is a non-safety-related, non-seismic Category I system and will consist of a natural draft cooling tower from which the TSW and CWS systems will receive cooling water. Water circulation, makeup, chemical control, and inventory blowdown are part of the CWS.

The PCHS is designed to maintain the temperature of the water entering the CWS within the range of 0 °C to 38 °C (32 °F to 100 °F).

The PCHS is designed to ensure that its failure will not adversely affect safety-related equipment. A flooding analysis of the TB was performed using the CWS and postulating that a complete rupture of a single expansion joint thus introducing PCHS water into the TB. In this situation, high-level sensors in the condenser pit will isolate the CWS on sensing a high water level in the pit. Should this isolation fail, excess flood water will rise to grade level and exit on site. As discussed in Section 10.1 of the SSAR, safety-related instrumentation is provided in the TB which detects the oil pressure of the main turbine control valves and the turbine first-stage pressure and main condenser pressure. As discussed in SSAR Section 3.4.1 and Subsection 10.4.5.6 and as reviewed in Section 3.4.1 of this report, this equipment is protected from both internal and external flooding. Based on this, the staff concludes that failure of the PCHS will not adversely affect safety-related equipment and that the PCHS meets the requirements of GDC 2.The plant design is for a single-unit site and, therefore, the requirements of GDC 5 regarding the sharing of SSCs between units do not apply.

Because the PCHS is a non-safety-related system and is not required to remove reactor heat, the requirements of GDC 44 do not apply.All active and passive system components will be accessible for inspection, maintenance, and testing during normal power operation. Therefore, the inspection and testing requirements of GDC 45 and 46 are met.

GE submitted the interface requirements relating to the PCHS. The adequacy and acceptability of the interface requirements are evaluated in Section 14.3 of this report.

The conceptual design and interface requirements provided for the PCHS provide adequate guidelines to ensure that the plant-specific design can meet the requirements of GDC 2, 5, and 44 with respect to protection against natural phenomena, shared systems, and heat removal. The system will be designed to allow periodic inspections and tests and will, therefore, meet the requirements of GDC 45 and 46. Use of the interface criteria will allow an applicant referencing the ABWR to design a PCHS that will meet all applicable regulatory requirements. Therefore, the staff concludes that the design of the PCHS system is acceptable.

11 RADIOACTIVE WASTE MANAGEMENT

The advanced boiling water reactor (ABWR) design has three radioactive waste management systems: the liquid waste management system, the gaseous waste management system, and the solid waste management system. The systems are designed to provide for the controlled handling and treatment of liquid, gaseous, and solid wastes. The liquid radioactive waste system will collect and process liquid wastes from equipment and floor drains; sampling, decontamination, and laboratory wastes; reactor water cleanup decant wastes; chemical wastes; and detergent wastes. The gaseous waste system consists of (1) catalytic recombiners to reduce the volume of offgases from the main condenser air ejector, (2) charcoal delay beds to allow decay of short-lived noble gases from the main condenser air ejector and to adsorb radioiodines, and (3) high-efficiency particulate air (HEPA) filters to retain particulates in the offgas stream. Thus, the system will control the release of gaseous radioactive effluents to the site environs so as to keep the exposure of persons in unrestricted areas as low as reasonably achievable in accordance with 10 CFR Part 20 and Appendix I to 10 CFR Part 50. The solid waste system will package spent resins and backwash slurries, solidify concentrator bottoms, incinerate and package combustible dry radioactive materials, compact and package noncombustible materials, and store processed solid wastes before they are shipped off site to a licensed facility for burial. Radioactive waste management also includes monitoring and sampling systems to detect and measure radioactive materials in plant process and effluent streams.

The staff reviewed the applicant's design, design criteria, and design bases for the radioactive waste management systems for the ABWR design. The acceptance criteria that the staff used for this evaluation are set forth in Section II of Standard Review Plan (SRP) Sections 11.1, 11.2, 11.3, 11.4, and 11.5, which include 10 CFR 50.34(a) as it relates to the technical information contents in safety analysis reports, 10 CFR 20.106 as it relates to radioactivity in effluents to unrestricted areas and 10 CFR Part 71, as it relates to packaging of processed solid wastes. In lieu of 10 CFR 20.106, the staff used 10 CFR 20.1302, which is the current requirement. The staff also used compliance with 10 CFR 50.34(f)(2)(xvii) as it relates to instrumentation for monitoring noble gases and continuous sampling of radioiodines and particu-lates in gaseous effluents during an accident and onsite capability to analyze and measure these samples, as acceptance criteria for the gaseous effluent monitoring and sampling systems. Additionally, the staff used compliance with 10 CFR 61.56 as it relates to waste characteristics for the waste products that result from solid waste processing. The above SRP sections, additionally, include the following general design criteria (GDC) of 10 CFR Part 50, Appendix A, as acceptance criteria for radioactive waste management systems and liquid and gaseous process and effluent monitoring and sampling systems: GDC 3 as it relates to providing protection to gaseous waste handling and treatment systems from the effects of an explosive mixture of hydrogen and oxygen; GDC 60 as it relates to the radioactive waste management systems being designed to control releases of radioactive materials to the environment; GDC 61 as it relates to the liquid and gaseous radioactive waste management systems and ventilation systems for the fuel storage and handling areas being designed to assure adequate safety under normal and postulated accident conditions; and GDC 63 and 64 as they relate to solid radioactive waste management system and liquid and gaseous process and effluent monitoring and sampling systems being designed to monitor radiation leakages and radioactivity releases to the environment. The compliance of radioactive waste management systems and liquid and gaseous process and effluent monitoring and sampling systems with the above regulations are discussed in the sections that follow. Because specific compliance with Appendix I to 10 CFR Part 50 and the guidelines given in American National Standards Institute (ANSI) N13.1, "Guide to Sampling Airborne Radioactive Materials in Nuclear Facilities," Regulatory Guide (RG) 1.21, "Measuring and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants," and RG 4.15, "Quality Assurance for Radiological Monitoring Programs (Normal Operation)--Effluent Streams and the Environment," is not within the scope of ABWR design, the staff will review individual combined license (COL) applications for the ABWR design to ensure their conformance with these documents. Therefore, this was identified as draft final safety evaluation report (DFSER) COL Action Item 11.0-1. By amended standard safety analysis report (SSAR) GE included COL License Information (Section 11.2.5.1, Item 1) which states Chapter 11 that the COL applicant will show compliance with 10 CFR Part 50, Appendix I and the guidelines given in ANSI N13.1 and the RGs 1.21 and 4.15. This approach by GE is acceptable. GE has also included this action item in the final certified SSAR.

Certified Design Material

Tier 1 design information and inspections, tests, analyses, and acceptance criteria (ITAAC) are required for the radwaste system. GE submitted the radwaste system ITAAC for staff review. This was identified as DFSER Open Item 11.0-1. GE provided a revised set of design descriptions and ITAAC. The adequacy and acceptability of the design description and the ITAAC are evaluated in Section 14.3 of this report. On the basis of this evaluation, this item is resolved.

11.1 Source Terms

The staff calculated the expected releases of radioactive materials via gaseous effluents using the boiling water reactor (BWR) GALE Code methodology described in NUREG-0016, Revision 1, January 1979. The calculations in the code for estimating the liquid and gaseous effluents during normal plant operation, including anticipated operational occurrences, are based on (1) data from operating reactors, (2) field and laboratory tests, (3) standardized coolant activities derived from American Nuclear Society 18.1 working group recommendations, (4) release and transport mechanisms that result in the appearance of radioactive material in liquid streams, and (5) the plant's radwaste system design features used to reduce the quantities of radioactive materials ultimately released to the environs. The principal parameters used in these calculations based on data given in ABWR SSAR Sections 11.1, 11.2, and 11.3 tables and the gaseous source terms are given in Tables 11.1 and 11.2, respectively, of this report. Capacities of the principal components of the liquid and gaseous waste management systems for a single-unit plant are listed in Table 11.3 of this report. The liquid effluent source terms will be reviewed on a plant-specific basis as discussed below.

11.2 Liquid Waste Management System

11.2.1 System Description and Review Discussion

The liquid radioactive waste management system will consist of process equipment and instrumentation necessary to collect, process, monitor, and recycle or discharge the processed radioactive liquid wastes. The processing equipment for the system will be located in the radwaste building. Treatment of liquid waste will depend on the source, activity, and composition of the particular liquid waste and on the intended disposal procedure. The liquid wastes generated during operation will be collected and processed in three liquid radwaste management subsystems. The three subsystems are (1) the low-conductivity waste (LCW) (high purity) subsystem, (2) the high-conductivity waste (HCW) (low purity) subsystem, and (3) the detergent waste subsystem. These systems are described in detail in SSAR Section 11.2. The LCW subsystem will use high efficiency filters that require less backwash water than the precoat filters used in current designs. The LCW subsystem will receive less wastes because the ABWR will not have recirculation pumps and associated valves and will not regenerate the deep bed condensate demineralizers or use ultrasonic resin cleaning as older BWR designs do. For these reasons, the staff expects less generation of waste in the LCW subsystem than in the systems of older BWRs.

The liquid radwaste treatment systems are designed to completely recycle of processed liquids from the LCW and HCW subsystems during normal operation. Processed liquids will be handled on a batch basis to permit optimum control and release of radioactive materials from the LCW, HCW, and detergent waste subsystems. Discharge of processed LCW or HCW water will be solely governed by the plant water balance considerations. Before being released, samples will be analyzed to determine the types and amounts of radioactivity present. On the basis of the results of the analyses, the waste from the HCW and LCW subsystems will be recycled for eventual reuse in the plant, retained for further processing, or released under controlled conditions to the environment through the liquid pathway. All detergent wastes are expected to be released. A common radiation monitor (RM) in the discharge line will automatically terminate liquid waste discharges to the discharge canal from the LCW, HCW, or the detergent waste subsystem if radiation measurements exceed a predetermined level set by the COL applicant in order to meet 10 CFR Part 20, Appendix B, Table 2, Column 2 effluent concentration limits for the applicable subsystem. The predetermined level will be based on the ratio of instantaneous radionuclide concentration in any unrestricted area to the effluent concentration limit for that radionuclide given in the above table summed over all the radionuclides present in the liquid effluent not exceeding 10. This was identified as DFSER COL Action Item 11.2.1-1. By amended SSAR Chapter 11, GE identified COL License Information Section 11.2.5.1, Item 5, which states the COL applicant will provide a RM in the liquid radwaste discharge line to the environment to perform the function stated above. This approach by GE is acceptable. GE has also included this action item in the final certified SSAR.

The LCW subsystem will collect and process clean wastes such as those from equipment drains (from the drywell, reactor, radwaste, and turbine buildings) and spent resin backwash transfer water. The wastes will be collected in one or two parallel LCW collector tanks, filtered in one or two parallel high efficiency filters (for the normal waste generation rate, one collector tank and one filter are used) for removal of insolubles and demineralized in a mixed-bed demineralizer and a backup polishing demineralizer. Conductivity instrumentation on the demineralizer discharge will route the effluent either to the LCW sample tanks or back to the LCW collector tanks for reprocessing. From the sample tanks, the liquid stream will normally be routed to the condensate storage tank for reuse. However, a small fraction of the processed waste may be discharged from one of the sample tanks should the plant's water balance considerations dictate such a discharge. The staff estimates that approximately 1 percent of the processed LCW will be discharged. The staff estimates that the

Table 11.1 Principal parameters used in the calculation of gaseous and liquid effluents from ABWR

Parameter	Value
Thermal Power, MWt	3926
Total steam flow rate, kg/h	7.63E6 (1.68E7 lb/h)
Reactor coolant mass, kg	3.06E5 (6.75E5 lb)
Steam/water concentration, reactor vessel:	
Halogens	0.015
Particulate	0.001
RWC deminieralizer flow rate, kg/h	1.52E5 (3.35E5 lb/h)
Fraction of FW through condensate demineralizer	0.67
Reactor bldg. iodine release fraction	1.0
Reactor bldg. particulate release fraction	1.0
Radwaste bldg. iodine release fraction	1.0
Radwaste bldg. particulate release fraction	1.0
Turbine bldg. iodine release fraction	1.0
Turbine bldg. particulate release fraction	1.0
Mechanical vacuum pump iodine release fraction	1.0
Charcoal delay system:	
Kr dynamic adsorption coefficient, cm^3/g	16.0
Xe dynamic adsorption coefficient, cm^3/g	260.0
Ar dynamic adsorption coefficient, cm^3/g	6.4
Mass of charcoal, t	113.4 (125 tons)
Liquid Waste inputs	
High purity (low conductivity subsystem)	
Waste collection rate, m^3/day	57.5 (15200 gpd)
Reactor coolant activity (RCA) fraction	0.23
Collection, and processtime, days	5.98, 0.96
DF for halogens; Cs and Rb; others*	1000; 100; 1000
Fraction discharged	0.01
Low purity (high conductivity subsystem)**	
Waste collection rate, m^3/day	23.8 (6300 gpd)
RCA fraction	0.0028
Collection, and process time, days	1.52, 0.67
IF for halogens; Cs and Rb; others*	10,000; 100,000; 100,000
Fraction discharged	0.1
Detergent Wastes	
DF for radionuclides	1.0
Fraction discharged	1.0

* Excludes dissolved noble gases and tritium
** Includes chemical wastes

Table 11.2 Calculated annual release of radioactive materials in gaseous effluents from single ABWR unit

Nuclide	Building Vents*		Gland Seal		Mechanical Vacuum Pump		Offgas System		Total	
	MBq/yr	Ci/yr	MBq/yr	Ci/yr	MBq/yr	Ci/yr	MBq/yr	Ci/yr	MBq/yr	Ci/yr
Ar-41	5.6E5	1.5E1	0.0	0.0**	0.0	0.0	3.7E4	1.0	5.9E5	1.6E1
KR-83M	0.0	0.0	1.5E5	4.0	0.0	0.0	0.0	0.0	1.5E5	4.0
KR-85M	1.1E6	2.9E1	2.6E5	7.0	0.0	0.0	3.1E6	8.5E1	4.4E6	1.2E2
KR-85	0.0	0.0	0.0	0.0	0.0	0.0	1.0E7	2.7E2	1.0E7	2.7E2
KR-87	2.3E6	6.3E1	8.9E5	2.4E1	0.0	0.0	0.0	0.0	3.2E6	8.7E1
KR-88	3.5E6	9.5E1	8.9E5	2.4E1	0.0	0.0	2.2E5	6.0	4.8E6	1.3E2
KR-89	2.3E7	6.1E2	4.8E6	1.3E2	0.0	0.0	0.0	0.0	2.7E7	7.4E2
XE-131M	0.0	0.0	0.0	0.0	0.0	0.0	1.3E6	3.6E1	1.3E6	3.6E1
XE-133M	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
XE-133	1.8E7	4.8E2	3.3E5	9.0	4.8E7	1.3E3	8.5E7	2.3E3	1.5E8	4.1E3
XE-135M	3.7E7	9.9E2	1.1E6	2.9E1	0.0	0.0	0.0	0.0	3.7E7	1.0E3
XE-135	2.7E7	7.4E2	9.6E5	2.6E1	1.9E7	5.0E2	0.0	0.0	4.8E7	1.3E3
XE-137	4.8E7	1.3E3	5.6E6	1.5E2	0.0	0.0	0.0	0.0	5.2E7	1.4E3
XE-138	3.7E7	1.0E3	3.5E6	9.5E1	0.0	0.0	0.0	0.0	4.1E7	1.1E3
I-131	5.9E3	1.6E-1	5.9E1	1.6E-3	3.3E3	8.8E-2	0.0	0.0	9.3E3	2.5E-1
I-133	8.5E4	2.3	2.1E2	5.8E-3	3.6E4	9.7E-1	0.0	0.0	1.2E5	3.3
C-14	0.0	0.0	0.0	0.0	0.0	0.0	3.5E5	9.5	3.5E5	9.5
H-3	2.2E6	5.9E1	0.0	0.0	0.0	0.0	0.0	0.0	2.2E6	5.9E1
CR-51	1.0E2	2.7E-3	0.0	0.0	3.7E-2	1.0E-6	0.0	0.0	1.0E2	2.7E-3
MN-54	2.2E2	6.0E-3	0.0	0.0	0.0	0.0	0.0	0.0	2.2E2	6.0E-3
CO-58	5.6E1	1.5E-3	0.0	0.0	0.0	0.0	0.0	0.0	5.6E1	1.5E-3
FE-59	2.9E1	7.9E-4	0.0	0.0	0.0	0.0	0.0	0.0	2.9E1	7.9E-4
CO-60	4.8E2	1.3E-2	0.0	0.0	2.1E-2	5.6E-7	0.0	0.0	4.8E2	1.3E-2
ZN-65	4.1E2	1.1E-2	0.0	0.0	1.3E-2	3.4E-7	0.0	0.0	4.1E2	1.1E-2
SR-89	2.3E2	6.1E-3	0.0	0.0	0.0	0.0	0.0	0.0	2.3E2	6.1E-3
SR-90	1.1	3.0E-5	0.0	0.0	0.0	0.0	0.0	0.0	1.1	3.0E-5
NB-95	3.7E2	1.0E-2	0.0	0.0	0.0	0.0	0.0	0.0	3.7E2	1.0E-2

* Does not include the HEPA filtered offgases resulting from incineration of certain types of dry solid wastes. The total release is expected to be 592 MBq/yr (0.016 Ci/yr) in particulate form.

** For noble gases and C-14, 0.0 means less than 37,000 MBq/yr (1Ci/yr). For others, it means that the release is a negligible fraction of the total release for the isotope.

Table 11.2 Calculated annual releases of radioactive materials in gaseous effluents from single ABWR unit (continued)

Nuclide	Building Vents*		Gland Seal		Mechanical Vacuum Pump		Offgas System		Total	
	MBq/yr	Ci/yr	MBq/yr	Ci/yr	MBq/yr	Ci/yr	MBq/yr	Ci/yr	MBq/yr	Ci/yr
ZR-95	6.7E1	1.8E-3	0.0	0.0	0.0	0.0	0.0	0.0	6.7E1	1.8E-3
MO-99	2.5E3	6.8E-2	0.0	0.0	0.0	0.0	0.0	0.0	2.5E3	6.8E-2
RU-103	1.6E2	4.3E-3	0.0	0.0	0.0	0.0	0.0	0.0	1.6E2	4.3E-3
Ag-110M	8.9E-2	2.4E-6	0.0	0.0	0.0	0.0	0.0	0.0	8.9E-2	2.4E-6
SB-124	8.1	2.2E-4	0.0	0.0	0.0	0.0	0.0	0.0	8.1	2.2E-4
CS-134	2.7E2	7.3E-3	0.0	0.0	1.2E-1	3.2E-6	0.0	0.0	2.7E2	7.3E-3
CS-136	2.2E1	6.0E-4	0.0	0.0	7.0E-2	1.9E-6	0.0	0.0	2.2E1	6.0E-4
CS-137	4.1E2	1.1E-2	0.0	0.0	3.3E-1	8.9E-6	0.0	0.0	4.1E2	1.1E-2
BA-140	1.2E3	3.2E-2	0.0	0.0	4.1E-1	1.1E-5	0.0	0.0	1.2E3	3.2E-2
CE-141	4.1E2	1.1E-2	0.0	0.0	0.0	0.0	0.0	0.0	4.1E2	1.1E-2

* Does not include the HEPA filtered offgases resulting from incineration of certain types of dry solid wastes. The total release is expected to be 592 MBq/yr (0.016 Ci/yr) in particulate form.

normal waste generation rate for the LCW system will be about 58 m^3/day (15,200 gpd); GE estimates 55 m^3/day (14,530 gpd). The capacity of the limiting processing equipment is 720 m^3/day (190,080 gpd). The difference between the expected normal waste generation rate and the design process flow rate provides adequate reserve for processing a surge in LCW generation rate.

The HCW subsystem will collect and process water of relatively high conductivity, such as the wastes from the floor drains (from drywell, reactor, radwaste, turbine, and service buildings). The HCW subsystem will also collect and process chemical wastes from chemical laboratories and laboratory drains. The wastes will be collected in one of two parallel HCW collector tanks, chemically adjusted to a suitable pH for evaporation, and concentrated in one of two parallel forced-circulation concentrators or evaporators to reduce the volume of water and decontaminate the distillate. The distillate will be demineralized by the HCW demineralizer and normally will be routed to the LCW system upstream of the polishing demineralizer. During normal processing, the LCW polishing demineralizer will be bypassed and the processed HCW will be directed to the LCW sample tanks. The processed stream will then be routed to the condensate storage tank for reuse. A small fraction of the processed HCW may be discharged from one of the LCW sample tanks should the plant's water balance dictate such a discharge. The staff estimates that approximately 10 percent of the processed HCW will be discharged. The distillate from the HCW demineralizer also can be routed to an HCW distillate tank to be reprocessed by the HCW demineralizer, if required. The concentrated waste from the evaporator will be routed to the concentrated waste storage tank for further processing by the solid waste system. The staff estimates that the normal waste generation rate for the HCW system will be approximately 24 m^3/day (6300 gpd); GE estimates 15 m^3/day (4000 gpd). The capacity of the limiting processing equipment in this system is 142 m^3/day (37,440 gpd), leaving adequate reserve for processing a surge in the HCW generation rate.

The detergent waste subsystem will collect and process detergent wastes from personnel showers and laundry operations. Detergent wastes will be collected in the single hot shower drain (HSD) receiver tank, processed through one or two HSD filters, and routed to the HSD sample tank before discharge. GE estimates that the normal generation of detergent waste will be approximately 11 m^3/day (3000 gpd); the staff's estimate is 4 m^3/day (1000 gpd). In a June 7, 1990, submittal, GE further stated that, if storm drains are included, the waste

Table 11.3 Design capacities of principal components in the liquid and gaseous radwaste treatment systems for ABWR single unit

Component	No.	Capacity or Flow Rate ▲	
Liquid Systems*			
High purity (low conductivity) subsystem			
Low conductivity collection tank	2	$430m^3$	(114,000 gal)
Waste high efficiency filter	2	$15\ m^3/h$	(66 gpm)
Waste demineralizer (mixed bed)	2	$30\ m^3/h$, $36\ m^3/h$	(130 gpm, 160 gpm)
Sample tank**	2	$430\ m^3$	(114,000 gal)
Low purity (high conductivity) subsystem			
High conductivity collection tank	2	$45\ m^3$	(12,000 gal)
Waste evaporator	2	$3\ m^3/h$	(13 gpm)
Distillate demineralizer	1	$6\ m^3/h$	(26 gpm)
Sample tank **	2	$430m^3$	(114,000 gal)
Distillate tank	1	$16\ m^3$	(4,200 gal)
Detergent Waste Subsystem			
Hot shower drain receiver tank	1	$33\ m^3$	(8,700 gal)
Hot shower drain sample tank	2	$210\ m^3$	(55,500 gal)
Detergent filter	2	$6m^3/h$	(26 gpm)
Gaseous Systems*			
Ambient temperature RECHAR system			
Catalytic recombiner	2	2413 kPa 232°C	(350psig)*** (450°F)†
Condenser	1	2413 kPa ***tube side 482°C	 (900°F)†
Charcoal adsorber beds	9	2413 kPa *** 4.4°C to 121°C	 (40°F to 250°F)†
Mass of activated charcoal		113.4t	(125 tons)

▲ For each component (e.g., for each tank, filter, demineralizer or evaporator)

* In accordance with RG 1.143

** Shared by high and low conductivity subsystems. Acts as a surge tank for both systems when condensate storage of the processed liquids is unavailable. In addition, serves as a sample tank for the high conductivity or low conductivity subsystem from where discharge to the environment can occur.

*** Design pressure

† Design temperature

generation for this subsystem will be approximately 31 m^3/day (8300 gpd). The storm drain water will be normally nonradioactive, but can become radioactive on contact with radioactive liquids. The storm drain water will be collected in one of the two HSD sample tanks and discharged after processing by the HSD filters, if needed. The staff estimates that all of the detergent wastes and storm drain water will be discharged. The capacity of the limiting processing equipment in this system is 284 m^3/day (75,000 gpd), which, together with the system's tanks will ensure adequate margin to collect and process any surge in the waste generation.

The liquid radwaste system has one discharge line to the environs for liquid waste. Radiation-monitoring equipment placed on this line will measure the activity discharged. At any one time, this line can be fed only by one HSD sample tank or one of the two LCW sample tanks. GE stated that administrative controls will limit the total plant release per year to 3,700 MBq (0.1 Ci), excluding tritium. The staff estimates a total annual release of about 7,400 MBq (0.2 Ci) for the liquid wastes, primarily as a result of 3330 MBq (0.09 Ci) of untreated detergent wastes and 3,700 MBq (0.1 Ci) because of adjustment for anticipated operational occurrences such as operator error, and 2.2 x 10^6 MBq (59 Ci) for tritium. Administrative controls for meeting GE's commitment to limit the liquid wastes to 3700 MBq/yr (0.1 Ci/yr) are not within the scope of the ABWR design, but will be the responsibility of the COL applicant. Therefore, this was identified as DFSER COL Action Item 11.2.1-2. By amended SSAR Chapter 11, GE identified COL License Information Section 11.2.5.1, Item 3 which states that the COL applicant will provide the specific administrative controls and liquid effluent source terms. This approach by GE is acceptable. GE has also included this action item in the final certified SSAR.

The tanks containing spent resin, filter/demineralizer, and filter sludges are part of the liquid radwaste management system. Separation of filter sludges and filter/demineralizer sludges from process and transfer water will take place in phase separator decant tanks. The liquid from the separator tanks will be routed to the LCW collector tanks. The spent resins from the condensate polishing system (i.e., part of the condensate/feedwater system) LCW and HCW demineralizers will be collected in a spent resin tank. The sludges from the separator tank, remaining after decant, and the spent resins from the spent resin tank will be treated either by a thin film dryer or by vendorsupplied mobile dewatering systems. The water will be routed to the LCW collector tank and the dewatered slurry will be loaded in high-integrity containers (HICs) for eventual shipment.

11.2.2 Conclusion

In evaluating of the liquid radioactive waste management system, the staff considered (1) the capability of the system to maintain releases below the limits in 10 CFR Part 20 during periods of fission-product leakage at design levels from the fuel, (2) the capability of the system to meet the processing demands of the station during anticipated operational occurrences, (3) the quality group and seismic design classification applied to the equipment, components, and structures housing the system, and (4) the design features that are incorporated to control the release of radioactive materials in accordance with GDC 60. The staff reviewed all applicable information provided in the amended SSAR Chapter 11 and GE's submittals dated June 7 and 29, and November 5, 1990, and August 2, 1991, in response to the staff's request for additional information.

The staff concludes that the liquid radwaste system includes the equipment necessary to control the releases of radioactive materials in liquid effluents in accordance with GDC 60 and radwaste system aspects of GDC 61 and that the design of the liquid waste management system is acceptable and meets the requirements of 10 CFR 20.1302 and GDC 60 and the applicable portion of GDC 61 for control of releases of radioactive material to the environment.

The staff further concludes that there is reasonable assurance that the COL applicant will be able to meet the Appendix I to 10 CFR Part 50 dose guidelines for radioactive materials released through liquid effluents with regard to the minimum discharge canal flow rate of 340 m^3/hr (1500 gpm) and the additional dilution credit of at least a factor of 10 between the point of release and the region in the unrestricted area where the water is used. The staff considers demonstrating compliance with 10 CFR Part 50, Appendix I, is the COL applicant's responsibility. So the staff will evaluate this compliance individually for each COL application. This was identified as DFSER COL Action Item 11.2.2-1. By amended SSAR Chapter 11, GE identified COL License Information (Section 11.2-5.1, Items 1 and 4) which requires the COL applicant to demonstrate the above compliance. This approach by GE is acceptable. GE has also included this action item in the final certified SSAR.

The staff's conclusions are based on the following findings:

(1) On the basis of ABWR parameters that govern reactor coolant system concentrations of radionuclides and design of the liquid radwaste treatment systems, as stated in Section 11.2.1 of

this report, the staff estimates that the total of radioactive wastes discharged via the liquid effluent during normal plant operation including anticipated operational occurrences will be no more than 7,400 MBq/yr (0.2 Ci/yr) (excluding tritium) and 2.2 x 10^6 MBq/yr (59 Ci/yr) for tritium. This finding, in conjunction with SSAR Section 11.2.3.2 assumed minimum dilution flow rate of 340 m^3/hr (1500 gpm) and at least an additional credit of a factor of 10 (as stated in SSAR Section 11.2.3.2) between the point of discharge and the region in the unrestricted area where water is used, provides reasonable assurance that the ABWR liquid waste management system will meet the applicable 10 CFR Part 50, Appendix I dose guidelines for liquid effluents.

(2) GE's ABWR design has met the requirements of 10 CFR Part 20 that will have the minimum discharge flow rate of 340 m^3/hr (1500 gpm) and for which liquid waste can only be discharged from either the HSD sample tank or the LCW sample tank. The staff has considered the potential consequences resulting from reactor operation and has determined that the concentrations of radioactive materials in liquid effluent averaged over a year, as permitted by 10 CFR 20.1302 for the above case, will be well below the limits in 10 CFR Part 20, Appendix B, Table 2, Column 2. Instantaneous discharge concentrations of the radionuclides in liquid effluents to an unrestricted area will also be within these limits because GE has stated in a submittal dated November 5, 1990, that the discharge rate via the single discharge line will be administratively controlled to conform to these limits. The staff will review the administrative controls to limit the instantaneous discharge concentrations of the radionuclides in liquid effluents to an unrestricted area to comply with the limits in 10 CFR Part 20, Appendix B, Table 2, Column 2, as explained in Section 11.2.1 of this report, on a plant-specific basis for the COL applicants. Therefore, this was identified as DFSER COL Action Item 11.2.2-2. By amended SSAR Chapter 11, GE included COL License Information Section 11.2.5.1, Item 5 which states the COL applicant will provide administrative controls to ensure the limits mentioned above. This approach by GE is acceptable. GE has also included this action item in the final certified SSAR.

(3) GE's ABWR design has met the requirements of GDC 60 and 61 with respect to system design for controlling releases of radioactive materials to the environment. The staff considered the capabilities of the proposed liquid radwaste treatment system to meet the demands of the plant resulting from anticipated operational occurrences and has concluded that the system's capacity and design flexibility are adequate to meet the anticipated needs of the plant as discussed in Section 11.2.1 of this report. The staff also reviewed GE's quality group classifications which are used for the system components and the seismic design applied to structures housing these systems. In the DFSER, the staff stated that quality assurance (QA) (Operation) provisions of the liquid radwaste systems will be reviewed individually for each COL application and identified this as DFSER COL Action Item 11.2.2-3. By amended SSAR Chapter 11, GE included COL License Information Section 11.2.5.1, Item 6 which states the COL applicant will provide QA (Operations) provisions of the liquid radwaste systems. This approach by GE is acceptable. GE has also included this action item in the final certified SSAR.

The design of the systems and structures housing these systems meets the applicable criteria given in RG 1.143, Revision 1. Specifically, the base mat and outside walls of the housing structures are seismic Category I to a height necessary to retain spilled liquids within the building. In the draft safety evaluation report (DSER) (SECY-91-235), the staff stated that the provisions incorporated in the ABWR design to control the release of radioactive materials in liquids resulting from inadvertent tank overflow were consistent with the criteria given in RG 1.143 except for the lack of a local alarm capability for the condensate storage tank (CST). This was identified as Open Item 70 in the DSER (SECY-91-235). GE responded in Amendment 20 to SSAR Section 9.2.9.2, Item (9), by stating: "Instrumentation shall be provided to indicate CST water level in the main control room. High water level will be alarmed both locally and in the main control room." This satisfies the staff concern regarding RG 1.143 requirements for a local high level alarm for CST which will be located in the radwaste building control room as stated in SSAR Section 11.2.1.2.1. Therefore, Open Item 70 is resolved.

IE Bulletin 80-05 identified an issue: cooling hot water in a low pressure tank could create a low vacuum condition and buckle the tank, releasing radioactive material or having other detrimental effects. In a fax dated May 21, 1992, GE stated that several low-pressure tanks that could contain primary system water have vents to prevent the development of a low vacuum condition. In the DFSER, the staff stated that the above information resolved the tank failure concern in IE Bulletin 80-05, subject to documentation of the information in the SSAR. Therefore, the staff identified the resolution of the issue as DFSER Confirmatory Item 11.2.2-1. By amended SSAR Section 11.2.4, GE included the above information by stating that the only tanks in the LWMs that can contain reactor water, diluted by other wastes are the LCW and HCW collector tanks and that these tanks are vented to preclude their vacuum collapse. This is acceptable to the staff, and therefore, this item is resolved.

The staff concludes that the liquid waste management system for the ABWR meets the acceptance criteria of SRP Section 11.2 and is, therefore, acceptable.

11.3 Gaseous Waste Management System

11.3.1 System Description and Review Discussion

The gaseous radioactive waste processing and plant ventilation systems are designed to collect, store, process, monitor, and discharge potentially radioactive gaseous wastes that will be generated during normal operation of the plant, including anticipated operational occurrences. The systems will consist of equipment and instrumentation necessary to reduce release of radioactive gases and particulates to the environment.

The principal sources of gaseous wastes in the plant will be the effluents from the offgas system, condenser mechanical vacuum pump, turbine gland seal system (TGSS), and ventilation exhausts from the radwaste building, containment purge, reactor building, and turbine building. All these effluents will be routed into the plant stack either directly or indirectly (TGSS and mechanical vacuum pump) and monitored continuously.

The major source of gaseous radwaste during normal plant operation before treatment will be the offgases from the main condenser air ejector. These will principally contain hydrogen and oxygen from the radiolytic decomposition of water, air from condenser in-leakage, fission and activation gases, and water vapor. To treat this effluent, the ABWR design uses an offgas processing system consisting of redundant catalytic hydrogen-oxygen recombiners, charcoal absorber delay beds, and a HEPA filter operating at ambient conditions.

The offgases will be diluted with sufficient steam in the last stage of the air ejector to reduce the hydrogen concentration to less than 4 percent by volume upstream of the recombiner. The offgases will be preheated in the first stage of the recombiners to approximately 177 °C (350 °F) to remove moisture before recombination and reduced in hydrogen concentration to less than 1 percent by volume by the recombiner(s). The recombiner effluent will subsequently be cooled to between 57 °C (135 °F) and 68 °C (154 °F) by the offgas condenser. The offgas condenser will also include baffles to reduce moisture entrainment. The offgas stream will be further cooled to 18 °C (65 °F) by the cooler condenser. The pressure boundary of the system will be detonation resistant, with a design pressure of 2413 kPa (350 psig). Redundant, nonigniting, detonation-resistant hydrogen analyzers will monitor hydrogen concentration downstream of the recombiners and alarm both locally and in the control room when appropriate.

Fission and activation gases will be held for decay in the charcoal absorber system downstream from the offgas condensers. Before entering the delay beds, these gases decay for 2.5 minutes during their transit from the main condenser to the delay beds. The charcoal absorber beds consist of one guard bed absorber followed by four parallel trains of two absorber beds in series. The total mass of charcoal will be 114,000 kg (250,000 lb). The offgas system is also designed to prevent, monitor, and suppress the potential ignition and combustion propagation of charcoal in the charcoal absorber tanks, with the necessary temperature elements in the charcoal tanks and connections for nitrogen purge and blanketing.

Before discharge, the offgas system effluent stream will be passed through an HEPA filter assembly to remove particulates. The holdup times in the ambient offgas treatment system charcoal beds at 38 °C (100 °F) with a dew point of 18 °C (65 °F) were calculated according to the methodology of NUREG-0016, Revision 1, and GE proprietary report NEDO-10751. The staff estimates these times to be approximately 30 days for xenon, 44 hours for krypton, and 18 hours for argon; GE estimates 42 days for xenon and 46 hours for krypton. These beds will also absorb iodines from the treatment system effluent. The offgas system is designed to withstand a hydrogen explosion.

The ventilation exhausts from all plant areas such as the reactor building (RB) (which includes the primary containment when it is vented or purged, the fuel handling area, the area housing the emergency core cooling system equipment, other areas such as the standby gas treatment system (SGTS) rooms, the fuel pool cooling system equipment rooms, and areas housing nonessential equipment), the service building controlled area, the radwaste building, and the turbine building, will be directed to the plant vent and monitored continuously before being released to the environs. The RB areas mentioned above will be serviced either directly or indirectly (for the primary containment purging or venting) by the secondary containment ventilation system during normal plant operation. However, ventilation exhausts from RB areas serviced by the RB safety-related electrical equipment heating, ventilation, and air conditioning (HVAC) system and the RB safety-related diesel generator HVAC system, the turbine island serviced by the electrical building ventilation system, and the control building areas serviced by the control building safety-related equipment area HVAC system will be discharged directly to the environs unmonitored. The lack of monitoring for the exhaust from these areas is evaluated in Section 11.5 of this report which concludes that this is acceptable. Although SSAR Section 9.4.8.1.2 states that the service building controlled area ventilation exhaust will be monitored before its release, SSAR Table 11.5.1 does not indicate any explicit monitoring provision for the subject exhaust. Therefore, the staff concluded in Open Item 71 in the DSER (SECY-91-235) that this discharge will not be monitored. In its response of December 19, 1991, to the issues raised in the DSER, GE stated: "The service building ventilation exhaust is sent to the plant release point where the offgases are monitored and sampled during release." By amended SSAR Section 11.5.2.2.4, GE further clarified the issue by stating that the exhaust from the service building controlled area will be discharged to the environs via the monitored plant vent. Therefore, the issue of service building ventilation exhaust monitoring is resolved. Open Item 71 in the DSER (SECY-91-235), is resolved.

On the basis of SSAR Section 9.4 and GE's responses dated May 23 and August 27, 1990, the staff stated in Open Item 72 in the DSER (SECY 91-235) that neither the mechanical vacuum pump exhaust nor the normal ventilation exhaust system of any building includes either charcoal adsorbers or HEPA filters to remove elemental and organic forms of iodine and particulates from the applicable effluent stream. In calculating the gaseous effluents, the staff, therefore, assumed that all the exhausts will be discharged to the environs untreated. In its response of December 19, 1991, GE confirmed the staff's assumption. The staff's assumption does not invalidate its conclusion that the gaseous waste management system discussed in Section 11.3.2 of this report is acceptable. Therefore, the issue of lack of charcoal absorbers and HEPA filters is resolved and Open Item 72 in the DSER (SECY-91-235) is resolved.

The calculated release values for iodines and particulate in Table 11.2 of this report reflect the above assumption. SSAR Sections 6.5.1, 9.4.5.1.3, and 9.4.5.1.5 state that, if high radiation is detected in the secondary containment exhaust or in the refueling floor atmosphere, the secondary containment normal ventilation system will be secured and the exhaust discharged through the safety-related SGTS, which consists of charcoal adsorbers and HEPA filters. The SGTS exhaust also will go through the monitored plant vent.

The plant stack and the major streams feeding the plant stack (offgas system and building ventilation systems) will be monitored to facilitate appropriate corrective action in a timely manner to prevent offsite release exceeding applicable limits. Additionally, the offgas treatment system includes an automatic control feature to terminate the post-treatment release if it exceeds a preset radiation level in the effluent. However, as part of Open Item 73 in the DSER, the staff questioned whether the monitors for the secondary containment exhaust will be sufficiently sensitive to detect a high-radiation level in the primary containment purge exhaust as specified by Branch Technical Position CSB 6-4, "Containment Purging During Normal Plant Operations." In SSAR Section 11.5.2.1.2, GE confirmed the sensitivity of these monitors to detect a high-radiation level in the primary containment purge exhaust. This part of Open Item 73 in the DSER (SECY-91-235) is resolved.

As the second part of Open Item 73 in the DSER (SECY-91-235), the staff noted that while SSAR Section 9.4.4.2.1.1 states that the turbine building ventilation exhaust will be monitored before its discharge to the monitored plant vent, SSAR Table 11.5-2 did not indicate any such process monitoring provision for that exhaust. GE amended SSAR Tables 11.5-1 and 11.5-2 to indicate that the turbine building ventilation exhaust will be monitored. This part of Open Item 73 is resolved. Both parts of Open Item 73, in DSER (SECY-91-235) are, therefore, resolved.

As stated above, all airborne radioactivity releases except offgases from the incinerator will be through the monitored plant vent. The plant vent is located on the reactor building at 76 m (249 ft) above grade and is the tallest point on the site. Incinerator offgases will also be monitored prior to their release to the atmosphere, through incinerator exhaust stack.

11.3.2 Conclusion

In evaluating the gaseous radwaste management system, the staff used the SRP criteria pertaining to (1) the capability of the system to maintain releases below the limits in 10 CFR Part 20 during periods of fission-product leakage at design levels from the fuel, (2) the capability of the system to meet the processing demands of the station during anticipated operational occurrences, (3) the quality group and seismic design classification applied to the equipment and to components and structures housing the system, (4) the design features that are incorporated to control the releases of radioactive materials in accordance with GDC 60, and (5) the potential for gaseous releases resulting from hydrogen explosions in the gaseous radwaste system. The staff also reviewed the capability of the offgas system to limit the whole-body dose to less than 10 percent of the 10 CFR Part 100 limits for an individual exposure of 2 hours at the nearest exclusion area boundary (EAB) as a result of radioactive releases from a postulated offgas system leak or failure as assumed in BTP ETSB 11-5, Revision 0, July 1981. The staff reviewed all the applicable information provided in the amended SSAR as well as GE's submittals dated May 23, June 29, August 22, September 14, and October 26, 1990, December 19, 1991, and June 23, 1993.

The staff concludes that the gaseous radioactive waste management system design for the ABWR meets the applicable requirements of 10 CFR 20.1302 and GDC 3, 60, and 61 with regard to radioactivity in gaseous effluents released to unrestricted areas, fire protection, control of releases of radioactive materials, and radioactivity control in the gaseous waste management system and ventilation system associated with fuel storage and handling areas.

The staff expects that the COL applicant will provide an operational demonstration that the system design complies with Appendix I to 10 CFR Part 50 numerical guidelines for offsite radiation doses as a result of gaseous or airborne radioactive effluents during normal plant operation, including anticipated operational occurrences. Therefore, this was identified as DFSER COL Action Item 11.3.2-1. By amended SSAR Chapter 11, GE included COL License Information in Section 11.3.11.1, which states that the COL applicant will demonstrate this above compliance. This approach by GE is acceptable. GE has also included this action item in the final certified SSAR.

Nonetheless, the staff has evaluated the ABWR design to determine if there is reasonable assurance that the COL applicant will be able to meet the Appendix I dose guidelines for design objectives. The ingestion, inhalation, and external irradiation of ground contamination pathway doses to applicable organs resulting from release of radioactive iodines, radioactive material in particulate form, and tritium and carbon-14 via airborne effluents depend on a number of site-dependent parameters. The population exposures (person-rem) and associated cost-benefit analysis are also site dependent. Therefore, the staff considered only if the standard design for the gaseous waste management system complies with Appendix I guidelines for external doses to any individual in an unrestricted area as a result of noble gas radionuclides in gaseous effluents. The staff concludes that there is reasonable assurance that ABWRs at sites that have an atmospheric dispersion factor (χ/Q) equal to or less than 9.8×10^{-6} sec/m^3 will meet the above dose guidelines (.05 mSv (5 mrem) per year to the total body).

Using the assumptions given in BTP ETSB 11-5 for analyzing a postulated leak or failure of a waste gas system and the EAB 0-2 hour χ/Q of 1.37×10^{-3} sec/m^3 (used in Chapter 15 of this report), the staff has determined that the wholebody dose at the EAB is less than 10 percent of the 10 CFR Part 100 limit. Therefore, the staff concludes that for all sites that have equal to or less than the above χ/Q at the EAB, the offgas system design will meet the above dose criterion and will be acceptable. This will be verified for each COL applicant.

These conclusions referred to above are based on the following findings:

(1) The ABWR design meets the requirements of GDC 60 and 61 by ensuring that the gaseous waste management system includes the equipment and instruments necessary to detect and control the release of radioactive materials in gaseous effluents.

(2) On the basis of expected radwaste inputs over the life of the plant, the staff has determined the releases of radioactive materials (noble gases, iodines, particulate, tritium and carbon-14) in gaseous effluents resulting from normal operation, including anticipated operational occurrences. The staff used the calculated releases for noble gases (Table 11.2 of this report) to determine the bounding value for χ/Q and assumed a 4-minute decay of the noble gas radionuclides during transit from the release point to the unrestricted area. The staff used the dose models and values for parameters given in RG 1.109 (Rev. 1) to evaluate compliance with Appendix I to 10 CFR Part 50. To calculate the external dose of noble gas radionuclides, the staff assumed a semi-infinite cloud model for the gaseous effluents. For the bounding χ/Q value quoted above, the staff calculated a total body dose (the limiting external dose) of 0.05 mSv/yr (5 mrem/yr), which meets the applicable Appendix I dose guideline.

(3) The ABWR design meets the requirements of 10 CFR Part 20 because the staff has considered the potential consequences resulting from reactor operation with a postulated fission product release rate consistent with an offgas noble gas release rate of 3.7×10^6 Bq/MWt-sec (100 μCi/MWt-sec) at 30 minutes decay for a BWR and estimated that, under these conditions, the concentration of radionuclides in gaseous effluents in unrestricted areas with a value of χ/Q that is equal to or less

than 9.8×10^{-6} sec/m^3 will be below the concentration values in 10 CFR Part 20, Appendix B, Table 2, Column 1. For KR-89 and XE-137, whose specific concentration values are not explicitly given in this table, the staff used relevant concentration values based on ratios of whole-body dose factors of noblegas radionuclides given in RG 1.109 (Rev. 1).

(4) The staff has considered the capability of the proposed gaseous waste management system to meet the anticipated demands of the plant resulting from anticipated operational occurrences and concludes that the system capacity and flexibility of design are adequate to meet the anticipated needs of the plant. (See item 3 above.)

(5) The staff reviewed the seismic design criteria including the quality group classifications used for the gaseous waste management system components and the structures housing the radwaste system and concludes that the design of the system and the structures meets the applicable criteria specified in RG 1.143.

(6) The staff reviewed the provisions incorporated in the ABWR design to control releases resulting from hydrogen explosions in the gaseous waste management system (SSAR Section 11.3, GE submittal dated June 29, 1990). The staff concludes that the features built into the design are adequate to prevent the occurrence of an explosion or adequate to withstand the effects of an explosion in accordance with GDC 3. (See Section 11.3.1 of this report regarding hydrogen recombiners and hydrogen analyzers.)

The staff concludes that the gaseous waste management system for the ABWR meets the acceptance criteria of SRP Section 11.3 and is, therefore, acceptable.

11.4 Solid Waste Management System

11.4.1 System Description and Review Discussion

The solid radioactive waste management system will consist of the equipment and instrumentation necessary to collect, solidify, incinerate, package, and store radioactive wastes resulting from the operation of the reactor water cleanup system, the fuel pool cooling and cleanup system, the suppression pool cleanup system, the condensate polishing system, the liquid radwaste system, the building ventilation systems, the SGTS, the offgas system, and miscellaneous solid wastes (e.g., paper, rags, contaminated clothing, gloves, shoe coverings) arising from the operation and maintenance of the plant. The solid radwaste management system is located in the radwaste building.

The ABWR solid waste system is designed to process two general types of solid wastes: wet solid wastes, which will be solidified or dewatered before being shipped off site, and dry solid wastes, which will be either incinerated or compacted and/or packaged before being shipped. Combustible dry wastes (e.g., rags, uniforms, paper) will be burned in an incinerator and discharged to an ash storage drum. The offgas from the incinerator will be passed through two ceramic filters in series and a HEPA filter before being released into the atmosphere through a monitored vent. On the basis of GE's submittal dated June 29, 1990, the staff estimates this release to be 592 MBq/yr (0.016 Ci/yr) and to be in particulate form. Incinerated ash will be discharged to an ash storage drum by ash discharge equipment located on the bottom of the incinerator. The description of the incinerator to be used, source of incinerator heat, storage facility for the heat source, and specific fire protection features to prevent any undue fire hazard resulting from incineration, were identified as DFSER COL Action Item 11.4.1-1. By amended SSAR Section 11.4.3.1 and June 2, 1993 submittal, GE included COL License Information (Section 11.4.3-1, Item 1) which states the COL applicant will provide the above information. The staff finds that GE's identification of the COL License Information pertaining to the incinerator, and testing of the major components of the incinerator and description of the incinerator operation in the SSAR (Section 11.4.2.2.6, and Figure 11.2-2, Sheet 26) are acceptable. GE has also included this action item in the final certified SSAR.

Noncombustible dry solid wastes will be compacted and placed in dry active waste drums for shipment.

There will be two forms of wet wastes: (1) slurries of spent resins and sludges from filters and filter demineralizer backwashes and (2) concentrated wastes from the HCW concentrators of the liquid radwaste treatment system. As stated in Section 11.2.1 of this report, the spent resins and the sludges will be dewatered and the resulting slurry will be loaded in HICs for eventual shipment.

The concentrated waste from the HCW concentrators will be routed through a thin-film dryer for dewatering. The water from this operation will be routed back to the HCW collector tanks. Air will be exhausted through the radwaste building HVAC exhaust. The dewatered, powdered waste will be pelletized, and the pellets will be mixed with cement glass in drums for eventual offsite shipment or an approved solidification process will be used. Air from the pelletizing and solidification process will be routed to the radwaste building HVAC exhaust via a particle filter and a HEPA filter.

SSAR Interface Requirement 11.4.3.1 stated the first COL applicant will provide detailed information to demonstrate that the wet waste solidification process using cement glass as the solidification agent will result in a product that complies with 10 CFR 61.56. In Open Item 74 in the DSER (SECY-91-235), the staff stated that the interface item should be modified to require all COL applicants to provide this information. After re-reviewing all interface items, GE committed to revise the SSAR to reclassify the need to demonstrate compliance of wet waste solidification product with 10 CFR 61.56, as a COL Action Item. In the DFSER, the staff found the GE commitment acceptable and, therefore, identified demonstration of the above compliance as DFSER COL Action Item 11.4.1-2. By amended SSAR Section 11.4.3.1, GE included COL License Information 11.4.3.1, Item (2) which states the COL applicant will provide detailed information to demonstrate that the wet waste solidification process will result in a product that complies with 10 CFR 61.56. This is acceptable. GE has also included this action item in the final certified SSAR.

On the basis of the Electric Power Research Institute report EPRI-NP-5528, Volume 1 and NUREG/CR-2907 (annual reports for 1986 and 1987, Volumes 7 and 8, only BWRs were considered), the staff estimates the processed wet wastes requiring shipment to be about 370 m^3/yr (13,000 ft^3/yr) containing approximately 4.4 x 10^7 MBq (1,200 Ci). The spent resin and filter and filter/demineralizer sludge slurries will be stored in HICs before shipment. The solidified concentrates will be stored in 0.21 m^3 (55-gal) drums before shipment. On the basis of NUREG/CR-2907 and GE's submittal dated June 29, 1990, the staff estimates that the processed dry wastes requiring shipment will be about 340 m^3/yr (12,000 ft^3/yr) containing approximately 4.4 x 10^5 MBq (12 Ci). However, with incineration of combustible dry wastes, the shipment volume will be less than 340 m^3/yr (12,000 ft^3/yr). The processed dry wastes will be stored in boxes or in 0.21 m^3 (55-gal) drums.

Because the establishment and implementation of a process control program (PCP) for solidifying the evaporator concentrates, using an approved solidification agent, and the dewatering process for the spent resins and filter sludges are dependent on the as-procured equipment for the ABWR standard design, the staff will review the PCP and the dewatering process for each COL applicant against BTP ETSB 11-3. This was identified as DFSER COL Action Item 11.4.1-3. By amended SSAR Section 11.4.3.1, GE included COL License Information in Section 11.4.3.1, Item 3 which states the COL applicant will provide a PCP for solidifying the evaporator concentrates using an approved solidification agent and the dewatering process for the spent resins and filter sludges. This is acceptable. GE has also included this action item in the final certified SSAR.

11.4.2 Conclusion

In evaluating the solid radioactive waste management system, the staff considered (1) system design objectives in terms of expected types, volumes, and activities of wastes processed for offsite shipment; (2) provisions for onsite storage of processed solid wastes before shipment; (3) procedures for disposal of incinerated waste; (4) system design to meet acceptance criteria of SRP Section 11.4; and (5) piping and instrumentation diagrams for the system.

On the basis of its review of amended SSAR Section 11.4 and GE's submittals dated June 7, June 29, October 26, and November 5, 1990, August 2, 1991, May 18, 1992, and June 2, 1993, the staff concludes that the solid waste management system design meets the requirements of 10 CFR 20.302(a), and GDC 60, 63, and 64. The design also complies with 10 CFR 61.56 and 10 CFR Part 71. The dewatered resin and filter sludge wastes will be Type B per 10 CFR 61.55 classification. As noted above, they will not be processed to a stable form. They will be stored in HICs to comply with 10 CFR 61.56 and 10 CFR Part 71 requirements. The conclusion on solid waste management system is based on the following findings:

(1) The design includes equipment and instrumentation for processing, packaging, and storing of radioactive solid wastes before shipment off site. Dedicated radwaste storage areas in the radwaste building can accommodate 221 0.21 m^3 (55-gal) drums and 13 boxes with storage capacity of approximately 62 m^3 (2200 ft^3) of processed wet waste (solidified concentrate) and dry solid wastes. However, before issuing the DSER (SECY-91-235), GE did not specify the capacity and maximum number of HICs that can be stored. Further, the staff was concerned that the dedicated radwaste storage areas mentioned above may not be able to accommodate the HICs before shipment. Therefore, the staff identified this concern as Open Item 75 in the DSER (SECY-91-235). By amended SSAR Chapter 11, GE responded to the above concern in SSAR Section 11.4.2.3.6. From its review, the staff finds that normally the HICs will

be shipped promptly after being filled, in the event of a shipping delay they will be stored with shielding in the truck area. SSAR Section 11.4.2.3.6 states that the truck area can accommodate 5 HICs containing approximately 24 m^3 (840 ft^3) of processed (i.e., dewatered) resins and sludges and the shielding for the HICs. On the basis of the available storage space for processed wet and dry solid wastes given above and the estimates of annual shipment volumes for these wastes given in Section 11.4.1 of this report, the staff concludes that the available storage space is sufficient to accommodate one full offsite waste shipment of dry wastes and 30 days of wet waste at normal generation rate in accordance with BTP ETSB 11-3, Positions B.III.2 and 3. Therefore, Open Item 75 is resolved. The capacities of tanks accumulating spent resins and filter sludges also meet BTP Position B.II.1.

Besides the GE submittals and SSAR revisions discussed above, in a teleconference on May 20, 1992, GE stated that onsite storage of low-level waste beyond that discussed above would be addressed by the COL applicant. Therefore, the staff identified the GE position as DFSER COL Action Item 11.4.2-1. By amended SSAR Section 11.4.3.1, GE included COL License Information in Section 11.4.3.1, Item 4 which states that the COL applicant will provide a discussion of onsite storage of low-level waste beyond that discussed in the SSAR. This is acceptable. The staff will review the COL applicant's discussion against the guidance provided in Generic Letter 81-38, "Storage of Low-Level Radioactive Wastes at Power Reactor Sites." (This guidance is similar to the one provided in Appendix 11.4-A to SRP Section 11.4.) GE has also included this action item in the final certified SSAR.

(2) The system will have the capability to process the types and volumes of wastes expected during normal plant operation, including anticipated operational occurrences, in accordance with GDC 60. Provisions for handling wastes meet the requirements of 10 CFR Parts 20. Specifically, the staff has determined that the offgases (resulting from incineration of combustible wastes) that will be exhausted through ceramic filters and a HEPA filter to a monitored vent will have minimal effect on compliance with 10 CFR 20.1302 relating to concentrations of radionuclides in gaseous or airborne effluents in unrestricted areas for sites with a χ/Q equal to or less than 9.8×10^{-6} sec/m^3. As stated in Section 11.3.2(3) of this report, the staff concludes that the subject concentrations will be below the applicable regulatory limit. By identifying the type, the expected quantity, the curie content, and the manner of disposal of the combustible wastes that will be incinerated, GE complies with 10 CFR 20.302 with regard to disposal of incinerator offgases that lies within the ABWR scope.

(3) The system for monitoring radiation levels and leakage complies with GDC 63 and 64. Radiation monitors at the end of a drum conveyor will monitor the radiation resulting from mixture in the drums and surface contamination of the drums. Devices such as position switches, weight elements, and level sensors will be used to prevent spillage while filling, pouring (solidification of evaporator concentrates), and overfilling the drums. Additionally, safety interlocks provided for the solidification process system will ensure that solidification will be performed only under certain conditions (identified in SSAR Section 11.4.2.2.5). In addition, the effluents resulting from the system inputs to the liquid radwaste and gaseous radwaste management systems are monitored by the respective monitors for these systems (together with other effluents from these systems).

(4) SSAR Sections 11.4.1.2 and 11.4.2.1 and GE's submittals dated June 7 and 29, 1990, and June 2, 1993, state that the quality group classification, seismic design, and other design features (such as heat tracing concentrate piping and tanks, and flushing connections for all components and piping that contain slurries) meet the guidelines of RG 1.143 and BTP ETSB 11-3, Position B.V.

(5) The staff finds that the proposed dewatering method for spent resin and filter sludges, namely, treatment by a thin-film dryer or by a vendor-supplied mobile dewatering system, is acceptable. However, as stated in Section 11.4.1 of this report, the staff will review the details demonstrating compliance of the dewatering process and solidification process with applicable positions of BTP ETSB 11-3 and 10 CFR 61.56, on a plant-specific basis for each COL application.

(6) Since radioactive material packaging is within the scope of the COL applicant, GE has included COL License Information in SSAR Section 11.4.3.1, Item (5) which states that the COL applicant will demonstrate the compliance of all radioactive waste shipping packages with 10 CFR Part 71 requirements for packaging such wastes. The staff

finds this acceptable and will review compliance with 10 CFR Part 71 on a plant-specific basis for each COL application.

(7) As part of Open Item 76 in the DSER (SECY-91-235), the staff stated that GE should identify the specific fire protection features available in the applicable area to prevent any undue fire hazard resulting from incineration. By amended SSAR Section 11.4.3.1, GE provided COL License Information in SSAR Section 11.4.3.1, Item 1 which addresses the issue raised in this part of Open Item 76. As discussed in Section 11.4.1 of this report, the staff finds this COL License Information acceptable. Therefore, part of Open Item 76 is resolved.

(8) In the DSER (SECY-91-235), the staff stated that SSAR Section 11.4.2.3.5 (last paragraph), Table 11.4-2, and the response to Question 430.171 were inconsistent and confusing. For example, the section stated that Table 11.4-2 represents the shipped volume of solid wastes and that Table 11.4-3 gives the corresponding curie content; however, the two tables could not be correlated since Table 11.4-2 gave the shipped volume only for the solidified concentrates (not the total volume of all solid wastes) and Table 11.4-3 gave curie content for the spent resin and filter sludges. Therefore, the staff requested GE to correct this section and Table 11.4-2 (Open Item 76). GE provided the requested information in Table 11.4-3. This information provides the volumes of various kinds of solid wastes expected to be shipped annually and their corresponding total curie content. Therefore, this part of Open Item 76, in the DSER (SECY-91-235), is resolved. Therefore, this resolved both parts of Open Item 76.

On the basis of these findings, the staff concludes that the solid radwaste management system design for the ABWR meets the acceptance criteria of SRP Section 11.4 and complies with 10 CFR 61.56 and is, therefore, acceptable.

11.5 Process and Effluent Radiological Monitoring and Sampling Systems

11.5.1 System Description and Review Discussion

The process and effluent radiological monitoring systems are designed to provide information about radioactivity levels in systems throughout the plant, indicate radioactive leakage between systems, monitor equipment performance, and monitor and control radioactivity levels in plant discharges to the environs.

On the basis of GE's telefax dated May 26, 1993, its submittal dated June 23, 1993, and SSAR Section 11.5.2.2.4, the staff finds that all airborne radioactive releases from the plant to the environment except the offgases from the incinerator to the environment will be exhausted through the plant vent. The major sources that will be combined and routed to the plant vent are the offgas exhaust, the radwaste building exhaust, the RB (secondary containment) exhaust, the service building controlled area exhaust, and the turbine building exhaust, which includes the gland seal system and the mechanical vacuum pump exhausts. A radiation-monitoring system (RMS) will monitor the plant vent discharge for gross radiation level and collect halogen and particulate samples. The offgases from the incinerator will be monitored and released to the environs via the incinerator exhaust stack. Besides the main plant vent gaseous effluent monitor and samplers, as indicated in ABWR SSAR Table 11.5-1, radiation monitors will be provided for monitoring the offgas post-treatment exhaust, TGSS exhaust, RB (secondary containment) exhaust, radwaste building vent exhaust, and turbine building exhaust. These monitors will be used to identify sources of airborne activity before mixing in the main plant vent. Gaseous process stream monitoring will include the offgas pretreatment RM, the carbon bed vault RM, and the control rod drive maintenance area exhaust RM.

The liquid effluent and process RMS include the liquid radwaste effluent and the RB closed cooling water system RMS.

The RMs, which will monitor the discharges from the gaseous and liquid radwaste treatment systems (i.e., offgas posttreatment effluent and processed liquid radwaste effluent), are designed to alarm and provide a signal to automatically close the waste discharge valve of the affected treatment system before exceeding the normal operation limits. The radiation monitor for the incinerator offgas discharge is also designed to alarm and initiate automatic termination of the exhaust to the environs before the radiation level in the exhaust exceeds the technical specification (TS) limit. The DFSER stated that the normal operation limits will be specified in the ABWR Radiological Effluent Technical Specifications (RETS) and, therefore, identified it as TS Item 11.5.1-1. However, GL 89-01 allows the RETS to be relocated in the Offsite Dose Calculation Manual (ODCM) which is plant specific. Since the requirement for the ODCM will be included in Section 5.0 (Administrative Controls) of the plant specific TSs to be provided by the COL applicant, TS Item 11.5.1-1 is resolved. Before being discharged from the radwaste treatment systems, liquid in the tanks will be sampled and analyzed. Release and dilution rates will be specified on the basis of the results of these analyses.

In addition to the gaseous and liquid effluent and processing RMS, there are systems required to initiate appropriate protective action in case of postulated accidents. These systems include the main steamline RMS, the fuel area (of the reactor building) ventilation exhaust RMS, the control building HVAC RMS, the SGTS RMS (the exhaust from this system goes to the plant stack), and the containment space-refueling mode RMS.

The ABWR design includes provisions for grab sampling and analysis of liquid sources (e.g., reactor coolant crud and filtrate, liquid radwaste system tanks, condensate storage tank, reactor building cooling water (RCW) system, reactor water cleanup system) and both liquid and gaseous effluent and process streams, including reactor service water and the circulating water system decant line for determination of gross radiation level and identity and quantity of specific radionuclides in the applicable stream or source. The stream or the source sampled, the parameters analyzed, the analysis frequency, and the sensitivity for analysis are listed in SSAR Tables 11.5-4 through 11.5-7.

The ABWR design includes accident monitoring instrumentation for monitoring noble gases, iodines, and particulate in gaseous or airborne effluent streams during an accident. As stated in SSAR Sections 1.A.2.15 and Section 7.5, GE considers that such instrumentation provided for the ABWR is generally in accordance with RG 1.97, Revision 3, and that, therefore, it meets the guidelines of NUREG-0737, Action Item II.F.1, Attachments 1 and 2, which is incorporated into 10 CFR 50.34(f)(2)(xvii). See the detailed discussion of this item in the following section.

11.5.2 Conclusion

The staff reviewed the amended SSAR Section 11.5 and GE's submittals dated June 2, 1989, February 28, 1990, December 19, 1991, and June 23, 1993, and GE's telefaxes dated May 26 and September 24, 1993, with regard to the process and effluent radiological monitoring and sampling systems for the ABWR. The review included piping and instrumentation diagrams (P&IDs) for the liquid and gaseous radwaste systems; SSAR Tables 11.5-1 through 11.5-7, which list the liquid and gaseous process and effluent RMSs and summaries of radiological analysis for liquid and gaseous process and effluent stream samples; information provided in SSAR Section 7.5.1.1 and Tables 7.5-1 and 7.5-2 (i.e., tables comparing ABWR design provisions for monitoring radioactive gaseous effluents during an accident with applicable RG 1.97 guidelines); and descriptions of the various building ventilation systems, the main condenser evacuation system, and the TGSS in so far as they relate to the radiation-monitoring provisions for these systems. However, in Open Item 77 in the DSER (SECY-91-235), the staff stated it could not review the location of the monitoring points relative to the effluent release points in the gaseous effluent streams because GE had not provided the corresponding P&IDs. Subsequently, GE submitted P&IDs that identify the location of monitoring points relative to the effluent release points. Therefore, Open Item 77 in the DSER (SECY-91-235), is resolved.

As stated at the beginning of Chapter 11 of this report, the staff will review specific compliance of the COL sampling program and quality assurance (QA) for radiological monitoring programs with ANSI N13.1 and RGs 1.21 and 4.15 on a plant-specific basis for each COL applicant. During such a plant-specific review, the staff will evaluate the plant-specific features and programs provided by the COL applicant to address the issue discussed in IE Bulletin 80-10 of possible contamination of nonradioactive systems and the resulting potential for unmonitored, uncontrolled release to the environment. Therefore, the staff's review of the sampling and analysis program for the ABWR standard design is limited to the identification of the streams required to be sampled, monitored and controlled.

The staff concludes that the design of the ABWR process and effluent radiological monitoring and sampling systems complies with the requirements (1) 10 CFR 20.1302 relating to radioactivity monitoring of effluents to unrestricted areas, (2) of GDC 60 that radioactive waste management systems be designed to control release of radioactive materials to the environment, and (3) of GDC 63 and 64 that radioactive waste management systems be designed to monitor radiation levels, leakage and releases to environment. The staff's conclusion is based on the following findings:

(1) The design includes provisions for monitoring the radioactivity of effluents to unrestricted areas. The exhausts from certain areas of the RB (serviced by the safety related RB electrical equipment and safety related RB diesel generator HVAC subsystems) and the turbine island (serviced by the electrical building ventilation system) will be directly released to the environs unmonitored. In addition to the above areas, the staff stated in Open Item 78 in the DSER (SECY-91-235) that the exhausts from the battery rooms and lube oil area in the turbine island and the reactor internal pump

(RIP) control panel room in the reactor building were unmonitored. However, the SSAR currently shows the lube oil exhaust being monitored at the plant vent and does not show a battery room in the turbine island (Figures 9.4-2a and 2b). The SSAR also shows the RB RIP ASD HVAC system (RIP control panel room HVAC system renamed) as a closed cooling HVAC system with no outside air supply to the room or exhaust from the room to the environs (Figure 9.4-5). Further, by telefax dated June 9, 1992, GE proposed to revise Section 11.5.2.2.4 to state that the exhausts from the areas serviced by the HVAC systems mentioned above are not monitored since the subject areas do not contain any radioactive systems and that the only releases to the environment by these systems would first have to be brought into the areas by their own HVAC system's supply fans. On the basis of the above telefax design information, the DFSER stated that lack of radiation monitoring of certain exhausts identified above is acceptable since GDC 64 does not require radiation monitoring of plant exhausts to the environs that do not have potential to be radioactive. Therefore, in the DFSER, the staff re-classified this part of DSER (SECY 91-235) Open Item 78 as DFSER Confirmatory Item 11.5.2-1 and required GE to incorporate the telefax design information in the applicable SSAR section. By submittal dated June 23, 1993, GE incorporated the subject information as a footnote to SSAR Section 11.5.2.2.4 after revising the subject information to include also the exhaust from the service building clean area since the exhaust from this area also qualifies for non-monitoring as explained above. In the above submittal, GE also stated that the exhaust from the area served by the control building essential electrical HVAC system will not be monitored prior to discharge to the environs though it contains (RCW) system components. This is because, the RCW, which is considered as a clean water system, is monitored to alarm at any radiation level in the system above background from potential leakage sources. Such contamination will require dumping the cooling water to radwaste treatment and replacing it with clean water. Therefore, the system will remain clean. Furthermore, at the system operating temperature below 35 °C (95 °F), airborne evolution of radioactivity from the cooling water system will be negligible. The staff has reviewed the above GE's justification for not monitoring this exhaust prior to its release to the environs and finds the justification, acceptable. GE has also included this information in the final certified SSAR. Therefore, Confirmatory Item 11.5.2-1 is resolved.

The staff further stated as part of Open Item 78 in the DSER (SECY-91-235) that GE had not provided sufficient information in its submittal dated April 26, 1991, to clarify that the service building ventilation system exhaust will be routed through the plant vent where radiation monitoring occurs. By amended SSAR Section 11.5.2.2.4, GE clarified that the service building controlled area ventilation system exhaust will be routed through the plant vent, where radiation monitoring occurs. This is acceptable. Therefore, this part of DSER (SECY-91-235) Open Item 78 is resolved.

(2) The design includes provisions for monitoring process streams (e.g., offgas post-treatment exhaust, secondary containment exhaust, RCW system) and provisions for initiating appropriate action in case of postulated accidents. Automatic control features include termination of liquid effluent release, incinerator offgas release, and the offgas system release as appropriate, when the preset radiation level for the applicable stream is exceeded. The automatic control features also include securing the normal secondary containment ventilation system and initiation of the SGTS under certain conditions identified in the SSAR. Also, since GE indicated a single monitor for the gland seal exhaust and mechanical vacuum pump exhaust, and totally clean steam will not normally be supplied for sealing the TGSS, the staff required in the DSER (SECY-91-235) that plant procedures include manual switchover to the auxiliary (backup clean) steam source whenever the monitor indicates that the exhaust stream concentration exceeds a preset level. As discussed in Section 10.4.3 of this report, GE has committed to provide a COL action item stating that the COL applicant will provide the necessary procedures for switchover to the auxiliary steam system when monitored radiation level in the TGSS exhaust exceeds an acceptable preset level. The staff identified the development of the procedures as DFSER COL Action Item 11.5.2-1. By amended SSAR Section 10.4.10, GE identified COL License Information in Section 10.4.10 for the TGSS effluents. The subject information among other requirements, spells out the requirement for switchover mentioned above. The staff agrees with GE's approach for providing sealing steam to the TGSS. GE has also included this action item in the final certified SSAR.

As part of Open Item 78 in the DSER (SECY-91-235), the staff questioned whether the secondary containment ventilation exhaust monitor will be sensitive enough to detect high-radiation level in the primary containment purge exhaust. In response to the above concern, GE has revised Section 11.5.2.1.2 to state that the detectors in the secondary containment ventilation exhaust will be sensitive enough to detect high-radiation levels during primary containment purge to alert the operator and to initiate appropriate measures. This part of Open Item 78 is resolved.

(3) As stated earlier in this section, the design includes provisions for sampling and analysis of radioiodines, particulates, and tritium in the process and effluent streams (for tritium, only in the effluent stream). However, as part of Open Item 78 in the DSER (SECY-91-235), the staff noted that SSAR Table 11.5-5 did not include grab sampling and analysis provisions for the gland seal process stream. SSAR Table 11.5-7 also did not include sampling and analysis provisions for the plant stack exhaust. In response to the above concern, GE added the sampling and analysis provisions for the gland steam condenser and plant stack exhausts (SSAR Table 11.5-7). Therefore, this part of Open Item 78 is resolved.

(4) SSAR Tables 7.5-1 and 7.5-2 provide design and qualification criteria for accident-monitoring instrumentation and the concentration ranges covered by the instrumentation. The staff finds that the design complies with RG 1.97 with regard to ranges and design and qualification criteria. However, as part of Open Item 78 in the DSER (SECY-91-235), the staff stated that neither these tables nor SSAR Section 11.5 contained sufficient information and that the following information was to be provided:

- type of instrumentation to be used, including the calibration frequency and technique

- monitoring locations (or points of sampling), including description of methods used to ensure representative measurements and background correction (the P&IDs for building ventilation systems were not provided to determine monitoring locations relative to the applicable release points for the gaseous effluent streams)

- location of instrument readout(s) and method of recording, including description of the method or procedure for transmitting or disseminating the data

- assurance of capability to obtain readings at least every 15 minutes during and following an accident

- description of procedures or calculational methods to be used for converting instrument readings to release rates per unit time, based on exhaust air flow and considering radionuclide spectrum distribution as a function of time after shutdown

- description of the sampling system design, including the sampling medium to demonstrate how the design meets the requirements identified in Clarification 2 of NUREG-0737, page II.F.1-7

- description of the sampling technique to be used under accident conditions to demonstrate how the technique meets the requirements identified in Clarification 3 of NUREG-0737, pages II.F.1-7 and II.F.1-8

- description of the sampling technique to ensure the system capability to collect and analyze or measure representative samples of radioactive iodines and particulate in plant gaseous effluents during and following an accident as identified in Table II.F.1-2 of NUREG-0737, page II.F.1-9

In response to the above request for information, GE provided COL License Information 11.5.6.1 through 11.5.6.5. The COL license information calls for the COL applicant to provide an operation and maintenance manual that describes or demonstrates (as appropriate) the following: calculation of radiation release rates from radiation measurements, sampling system design and its compliance with the shielding requirements identified in Clarification 2 of Attachment 2 to TMI Item II.F.1 of NUREG-0737, sampling technique and its compliance with the requirements identified in Clarification 3 of Attachment 2 to TMI Item II.F.1 of NUREG-0737, collection technique for extracting representative samples of radioiodines and particulates and calibration frequencies and techniques for the radiation sensors. The staff concludes that this part of Open Item 78 is resolved.

On the basis of these findings, the staff concludes that the design of the ABWR process and effluent radiological and sampling systems meets the acceptance criteria of SRP Section 11.5 and is, therefore, acceptable.

12 RADIATION PROTECTION

Standard safety analysis report (SSAR) Chapter 12 provides information on the radiation protection features and estimated occupation exposure associated with the advanced boiling water reactor (ABWR) design. The radiation protection measures for the ABWR are intended to ensure that internal and external occupational radiation exposures to plant personnel, contractors, and the general population, as a result of plant operations, including shutdown periods and anticipated operational occurrences (AOOs), will be within applicable limits of regulatory criteria and will be as low as is reasonably achievable (ALARA). The staff reviewed the SSAR for completeness against the guidelines of Regulatory Guide (RG) 1.70 (Rev. 3), "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," and against the criteria of Sections 12.1 through 12.5 of NUREG-0800, the standard review plan (SRP).

The staff reviewed GE's SSAR and supplemental information to determine if the ABWR design is sufficient to permit plant operations while maintaining radiation doses to personnel within the limits of 10 CFR Part 20, and if the design features are consistent with the guidelines of RG 8.8 (Rev. 3) "Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations will be As Low As Is Reasonably Achievable." On May 21, 1991, the Commission issued a revision of 10 CFR Part 20 that changed the system of radiation dose limitation. The previous occupational dose limit for whole-body radiation exposure was 12.5 mSv (1.25 rem) per quarter year with a provision to extend it to 30 mSv (3 rem) per year. The new limit is 50 mSv (5 rem) per year with a provision to extend it to 100 mSv (10 rem) per quarter year. The previous 10 CFR Part 20 limits for doses from licensed radioactive material inside the body (deposited through injection, absorption, ingestion, or inhalation) were separate from the dose limits for exposure to licensed sources outside the body. The new Part 20 limits the sum of the external whole-body dose (deep dose equivalent) and the committed effective equivalent doses resulting from radioactive material deposited inside the body. In addition, the new Part 20 requires that this sum (the total effective dose equivalent) be maintained ALARA for each individual. These changes to the regulation do not affect the acceptance criteria used by the staff to review the ABWR design. The SRP acceptance criteria provide assurance that the radiation doses resulting from exposure to licensed radioactive sources outside the body and inside the body can each be maintained well within the limits of 10 CFR Part 20 and ALARA. The balancing of internal and external exposure necessary to ensure that their sum is ALARA is an operational concern that will be reviewed in conjunction with a combined license (COL) application. The Part 20, as amended, contains a number of new programmatic requirements that do not affect plant design. Programmatic and operational radiation protection concerns will be addressed by the COL applicant.

The staff finds that the radiation protection measures incorporated in the ABWR design will provide reasonable assurance that occupational doses can be maintained ALARA and below the limits of 10 CFR Part 20 during all plant operations.

12.1 Ensuring That Occupational Radiation Doses Are As Low As Is Reasonably Achievable

The staff reviewed the ABWR design to ensure that GE had either committed to following the criteria of the RGs and staff positions referenced in SRP Section 12.1 or provided acceptable alternatives.

12.1.1 Policy Considerations

SSAR Section 12.1.1 describes GE's policy to ensure that ALARA considerations are factored into each stage of the ABWR design process. GE committed to ensure that the ABWR will be designed and constructed in a manner consistent with RG 8.8 (Rev. 3). The ALARA policy was applied through detailed engineering reviews and design modifications to ensure that the resulting plant design can maintain radiation exposures ALARA. This policy is consistent with the guidelines of RG 8.8 and is acceptable.

The policy considerations regarding plant operations contained in RG 8.8 (Rev. 3), RG 1.8 (Rev. 2) "Qualification and Training of Personnel for Nuclear Power Plants," and RG 8.10 (Rev. 1), "Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As Is Reasonably Achievable," are outside the scope of this review of the ABWR design. The COL applicant referencing the ABWR design will address these operational policy considerations to ensure that radiation doses are ALARA. This was identified as draft final safety evaluation (DFSER) COL Action Item 12.1.1-1. In Amendment 23 to the SSAR, GE revised Section 12.1.1 to clarify the policy considerations that will be addressed by the COL applicant. This is acceptable to the staff.

12.1.2 Design Considerations

The ABWR design aims to minimize the costs, both in terms of maintenance time and radiation exposure, associated with plant operation. The ABWR design employs features that will (1) eliminate the need for certain maintenance, (2) facilitate the required maintenance, and (3) minimize the sources of radiation exposure in the plant.

The ABWR design includes several design features consistent with the guidelines in RG 8.8 (Rev. 3). The plant layout has shielded rooms or cubicles for components that are the source of high radiation levels. Components in redundant systems are located in separate shielded rooms or cubicles so that radiation levels associated with an operating train of equipment will not significantly contribute to the radiological hazards associated with performing maintenance on the redundant train. The design of shielded rooms or cubicles have labyrinth access ways to reduce scattered radiation in adjacent areas. Removable shielded walls or hatches will be provided if space limitations in a room or cubicle prevent an adequate laydown area for maintenance of equipment. Appropriate use of remote operators and instrumentation will minimize the need to enter shielded rooms or cubicles. The remote back-flushing capability for plant filters and demineralizers will use gravity drains and piping that slope toward the backwash tank to minimize traps that would become radiation hot spots. Grafoil valve stem packing will reduce leakage of contaminated water from reactor systems and will minimize the need to repack the stems of these valves. These design features are consistent with the guidance in RG 8.8 (Rev. 3).

In addition to these design features, operational experience with previous boiling water reactor (BWR) designs has been factored into the ABWR design in several areas. Many unique ABWR features, designed to eliminate difficulties encountered in operating current BWRs, should also reduce occupational radiation exposure. An example is the elimination of reactor coolant recirculation piping inside primary containment. Several BWRs have experienced significant stress corrosion cracking, requiring replacement of this piping at the cost of thousands of person-rem radiation doses. Eliminating the coolant recirculation piping from the ABWR reactor not only eliminates the radiation exposure associated with recirculation pipe inspection and replacement but should also reduce the source of radiation in the drywell, thus reducing exposure to personnel performing other maintenance activities in the primary containment.

Other examples of design features that will reduce radiation exposure include the control rod drive (CRD) mechanism, layout of the lower drywell, and safety relief valve (SRV) design and layout. Current BWRs have external restraints on CRDs to prevent a rod ejection in the event of a CRD housing failure, which have to be cleared out of the way during CRD maintenance. The internal CRD restraint feature in the ABWR design will allow easier CRD removal and reduce radiation exposure associated with CRD maintenance. The arrangement of the lower drywell will also reduce radiation exposures during CRD maintenance by allowing easy access to the lower reactor vessel head for CRD and reactor internal pump (RIP) removal. A transport system is also provided to remove CRDs and RIPs from the drywell so that maintenance can be performed in a lower radiation area. Direct-action SRVs that require less maintenance than current pilot-operated valves are provided in the ABWR design. These SRVs are placed around the outside of the reactor vessel and have a dedicated hoist to facilitate maintenance.

In the draft safety evaluation report (DSER) (SECY-91-355), the staff identified two areas where the current BWR operating experience was not adequately accounted for in the ABWR design. These areas are the dose rates in the upper drywell during the transfer of irradiated spent fuel assemblies (SFA) (Open Item 112) and exposures resulting from a complete withdrawal of the traversing incore probe (TIP) (Open Item 106).

In a March 26, 1992, SSAR markup, GE provided additional information concerning the radiation protection design features of the ABWR TIP system in response to Open Item 106. These features include a shielded room for the TIP drive units and a separate shielded room for the parked TIP. Additional shielding is provided for the parked TIP and its drive cable to allow personnel to enter this room when the TIP is out of the reactor. The TIP drive units also have an electro-mechanical switch that will cut power to their drive spooler to prevent the activated portions of the TIP from being completely withdrawn into the drive housings. These features are designed such that radiation exposures resulting from TIP operations, and related abnormal AOOs, can be maintained ALARA. This was identified as DFSER Confirmatory Item 12.1.2-1. GE has also included this information in Amendment 20 of the SSAR and the staff finds it to be acceptable.

The potential for creating extremely high dose rates in the upper drywell during spent fuel handling operations and the potential for high dose rates around unshielded portions of the TIP conduit are discussed in Section 12.3.2 of this report.

12.1.3 Operational Considerations

Operational considerations regarding the implementation of a radiation protection program are outside the scope of this design certification review. The COL applicant referencing the ABWR design will address these operational considerations to the level of detail provided in RG 1.70 (Rev. 3). This was identified as DFSER COL Action Item 12.1.3-1. In Amendment 23, GE revised Section 12.1.3 of the SSAR to identify these operational considerations as an area to be addressed by the COL applicant. The staff finds it to be acceptable.

12.1.4 COL License Information

Section 12.1.4 of the DSER (SECY-91-355) identified three issues concerning compliance with RGs 8.10 (Rev. 1) and 1.8 (Rev. 2), and procedures for keeping occupational exposures ALARA, as outside the scope of this review. This was identified as DFSER COL Action Item 12.1.4-1. In Amendment 20 to the SSAR, GE revised Section 12.1.4 to properly characterize these issues. The staff finds it to be acceptable.

In Open Item F1.9-1, the staff identified the need for GE to include a COL action item related to the use of appropriate materials in the ABWR design which would reduce the potential for personnel exposures. GE provided a submittal dated February 7, 1994, which included a markup of SSAR Section 12.3.7.4, that added a COL action item stating that the applicant, following the design commitments included in SSAR Section 12.3.1.1.2, is responsible for material selection to ensure that radiation exposures are ALARA. The staff found this commitment for a COL action item to be acceptable.

12.2 Radiation Sources

The staff has audited the contained source terms and airborne radioactive material source terms in Section 12.2 and Chapter 11 of the ABWR SSAR for completeness against the guidelines in RG 1.70, (Rev. 3), and against the criteria set forth in Section 12.2 of SRP. The contained source terms are used as the basis for designing radiation protection features (including radiation shielding) and for personnel dose assessment. Airborne radioactive source terms are used in the design of ventilation systems and personnel dose assessment. The staff reviewed the source terms in the SSAR to ensure that GE had either committed to following the criteria of RGs and staff positions contained in SRP Section 12.2 or provided acceptable alternatives. In addition, the staff selectively compared source terms for specific systems against those used for plants of similar design. The staff finds that source term descriptions in the SSAR are not adequate to meet the criteria of RG 1.70 (Rev. 3) and NUREG-0800.

At the current stage of the ABWR design, GE does not have the specifications for the as-built systems or the as-procured hardware that would be available for a completed plant. Therefore, GE cannot describe the radioactive system components, which will be significant in-plant radiation sources, to the level of detail specified in RG 1.70 and the SRP. Although these details, such as radioactivity content, source geometry, equipment leakage, and plant location, are needed for the staff to verify the adequacy of the radiation shielding, ventilation and airborne radioactivity monitoring systems, the staff has determined that providing this information goes beyond the design requirements specified in 10 CFR Part 52. As an alternative, GE has provided a set of design acceptance criteria (DAC) that, if met, will verify the adequacy of the ABWR shielding design, plant ventilation design, and the design of the airborne radioactivity monitoring systems. Compliance with these DAC, as with other inspections, tests, analyses, and acceptance criteria (ITAAC), would be verified during plant construction prior to loading fuel into the reactor. The DAC in Table 3.2.a of the ABWR certified design material (CDM) specify the methods and assumptions for verifying the shielding design, including those for estimating the source terms to be used in the shielding analysis. Similarly, the DAC in Table 3.2.b of the CDM specify the acceptance methods for determining the airborne radioactivity concentrations used to verify the adequacy of the ventilation and airborne monitoring system designs. This alternative is acceptable to the staff.

12.2.1 Contained Sources and Airborne Radioactive Material Sources

GE describes radioactive sources in the ABWR design is contained in SSAR Chapters 11 and 12. Section 11.1 provides information on the radioactive source terms in reactor water and steam. Section 12.2 provides descriptions of plant components that will become significant sources of radiation during plant operations, including shutdown, and sources of airborne radioactive material.

During power operations, the greatest potential for personnel radiation dose is inside the primary containment drywell from nitrogen-16, noble gases, reactor neutrons, and prompt gammas. The steam and condensate systems outside the drywell are also significant sources of radiation because nitrogen-16 is generated during power operations. In other areas outside of the drywell, and inside the drywell after shutdown, the primary sources of personnel radiation exposure are the fission products in the coolant from fuel cladding defects and the activation products transported to and deposited in plant systems and components. Tables 12.1-7 through 12.2-30 in the SSAR list the radioactivity (or source terms) for typical components. These source terms are based on the assumed component geometry and locations listed in Table 12.2-5.

The estimates of concentrations of fission and activation products in the ABWR systems containing reactor water are based on American National Standards Institute/American Nuclear Society (ANSI/ANS)-18.1 "Radioactive Source Term for Normal Operation of LWRs" (1984), adjusted using the assumptions in RG 1.112 (Rev. 0), "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Power Reactors." Allowances are included for the buildup of activation products resulting from corrosion and wear on the basis of operating experience of reactors of similar design. Neutron and prompt gamma source terms are based on reactor core physics calculations. The source terms used to determine in-plant post-accident radiation levels meet the provisions in RG 1.3, "Assumptions used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors," as further discussed in Item II.B.2 of NUREG-0737, "Clarification of the TMI Action Plan Requirements."

In Open Item 107 in the DSER (SECY-91-355), the staff identified several deficiencies in GE's description of the contained radioactive source terms for the ABWR: sources inside the drywell and in the turbine building were omitted, sources in vital areas after an accident were not described, and source characterization was insufficient. GE amended SSAR Section 12.3.5 to indicate that the post-accident sources of concern in plant vital areas are limited to gamma radiation shine from the reactor building and the radioactive material in the post-accident coolant and effluent monitoring systems. GE also amended the tables of source terms in Section 12.2 to include sources inside the drywell and turbine building. SSAR Tables 12.2-7 through 12.2-30 provided nominal source strengths on the basis of expected system configuration and approximate component geometry. As discussed in Section 12.2 of this report, the actual source terms used in confirmatory shielding calculations will be determined as required by the DAC in Table 3.2.a of the CDM. The staff considers this item resolved.

Almost all of the airborne radioactivity within the plant results from equipment leakage. As discussed below, the leakage of contaminated fluids from system components cannot be quantified at this stage in the ABWR design. GE has proposed DAC in Table 3.2.b of the CDM to ensure that the airborne source terms in each room and operating area of the plant are calculated prior to fuel load. The lack of airborne source term description in the SSAR was also identified as Open Item 107 in the DSER (SECY-91-355). On the basis of the following discussion of DAC, the staff considers this item resolved.

12.2.2 Certified Design Material

As discussed in Section 12.2 of this report, the SSAR does not provide system layouts within rooms or cubicles or information about the type and size of components in these systems. Without this as-built or as-procured information, source term parameters needed to calculate radiation shielding for these systems cannot be provided as specified in the SRP. Similarly, since leakage characteristics of this unidentified equipment are not known, the concentrations of airborne radioactive material in equipment rooms or cubicles cannot be provided. As an alternative, GE provided DAC that require the COL applicant to determine source term parameters that will be verified during plant construction. DAC are discussed in Section 12.3.5 of this report. DAC describing the bases for the source term are consistent with the SRP acceptance criteria. Compliance with these DAC, supplemented by the information in SSAR Sections 12.2 and 12.3, is acceptable to adequately address the requirement to identify the kinds and quantities of radioactive materials expected to be produced by plant operation in 10 CFR 50.34(b)(3) and will ensure that the appropriate source terms (as supplemented by the guidance of RG 1.112 (Rev. 0), NUREG-0737, and ANSI/ANS 18.1) are used to demonstrate that the ABWR design meets the relevant requirements in 10 CFR Part 20 concerning the limitation of radiation does to personnel; 10 CFR 50.34(f) and GDC 19 with respect to operator access to plant areas during and following a reactor accident; and GDC 61 regarding adequate shielding, containment and confinement of fuel storage and handling, radioactive waste, and other systems which may contain radioactivity. The adequacy and acceptability of the ABWR design descriptions and ITAAC (including DAC) are evaluated in Section 14.3 of this report.

12.2.3 COL License Information

In the DFSER, the staff stated that two items were mischaracterized as interface items by the applicant: (1) the compliance with 10 CFR Parts 20 and 50 in Section 12.2.2.3 and (2) the determination of gamma shine from the turbine building in Section 12.2.1.3 of the SSAR. These SSAR sections referenced Section 12.2.3, which identified the issues as design interfaces. This was DFSER Confirmatory Item 12.2.3-1. In an SSAR markup of Chapter 12 dated April 16, 1993, GE deleted the discussion of interfaces in Sections 12.2.1.3, 12.2.2, and 12.2.3. Section 12.2.3 has been revised to identify the issues regarding compliance with 10 CFR Parts 20 and 50 as COL license information. As indicated in Section 12.2.2.4 of the SSAR, gamma shine from the turbine building is addressed in the DAC listed in Table 3.7 (see Section 12.3.5.1 of this report). GE has included this information in the SSAR. This change is acceptable. Therefore, this confirmatory item is resolved.

12.3 Radiation Protection Design

The staff has audited the facility design features in the SSAR, including the shielding, the ventilation, and the radiation and airborne radioactivity monitoring instrumentation for completeness against the guidelines in RG 1.70 (Rev. 3) and against the criteria set forth in SRP Section 12.3. The staff reviewed these design features to ensure that GE had either committed to following the criteria of RGs and staff positions referenced in SRP Section 12.3 or provided acceptable alternatives.

The staff concludes that GE has demonstrated that the ABWR design can meet the relevant requirements of 10 CFR Parts 20 and 50 and GDC 19 and 61 in all areas of the plant, as set forth below.

12.3.1 Facility Design Features

Several features in the ABWR design will help ALARA radiation doses associated with tasks such as maintenance, refueling, radioactive material handling, in-service inspection, decommissioning, and accident recovery (see Section 12.1.2 of this report). These features will facilitate access to work areas, reduce or allow the reduction of radioactive source intensity, reduce the occupancy requirements in high radiation fields, and provide for portable shielding, and remote-operation and instrumentation of radioactive systems. These ABWR features are consistent with the guidance of RG 8.8 (Rev. 3) and the SRP and are acceptable.

GE's drawings of the plant layout indicate six radiation zones which are the basis for occupancy and access restrictions for various areas within the plant during normal operations and accident conditions. Maximum design dose rates are established for each zone and are used as the basis for shielding each zone. This method of plant zoning is consistent with the guidance in RG 1.70 (Rev. 3) and the SRP and is acceptable.

In Open Items 108 through 110 in the DSER (SECY-91-355), the staff identified several deficiencies in SSAR Chapter 12 Figures 12.3-1 through 12.3-73, which depict plant radiation zones (during normal operations, normal shutdown, and accident conditions) and area radiation monitor locations. This was identified as DFSER Confirmatory Item 12.3.1-1. Amendments 21 and 22 to the SSAR provided more legible figures for the reactor, control, and radwaste buildings and resolved the discrepancies noted in the figures of the turbine building. These updated figures also indicate the normal controlled and uncontrolled access routes to the plant as well as the access and egress routes to and from plant vital areas under accident conditions. In response to a staff question, GE acknowledged the radiation zone designation above the spent fuel pool (from greater than 1.0 mSv/hr (100 mrem/hr)) was an error. GE revised the radiation zone designation for this area to less than 0.05 mSv/hr (5 mrem/hr). This dose rate for the area above the spent fuel pool is consistent with industry experience and, therefore, is acceptable to the staff. GE has also included this information in the SSAR and the staff finds it to be acceptable.

The buildup of activation products from corrosion and wear is a major contributor to occupational radiation doses. As discussed in Section 12.1.4 of this report, the COL applicant is responsible for material selection to ensure that radiation exposures are ALARA. Design features provided in the SSAR to minimize exposure sources include a reduction of cobalt-bearing components used in reactor systems (activated cobalt is a major contributor to plant radiation levels) and pre-filming (establishing a corrosion resistant layer on internal surfaces) of reactor systems before plant operation to minimize activated material deposition on system interior surfaces. Main condenser tubes and tube sheets will be made of titanium alloys to minimize the introduction of foreign material (which become activated and/or promote corrosion) into the reactor system as a result of condenser tube leakage. Other features, such as the use of seamless piping, straight-through valve design wherever possible, butt-welded piping connections, and back-flushing connections on instrument lines, will minimize buildup of radioactivity in plant piping systems.

In Open Item 111 in the DSER (SECY-91-355) concerned the provision in the ABWR design to facilitate chemical decontamination of heat exchangers in systems that carry radioactive water. The staff identified DFSER Confirmatory Item 12.3.1-2 on the basis of an April 9, 1992, draft SSAR amendment. Subsequently, Amendment 20 to the SSAR stated that the reactor water cleanup (CUW) non-regenerative and regenerative heat exchangers have separate decontamination connections. Heat exchangers in the residual heat removal (RHR) system and the heat exchangers for RIP cooling have fittings that will allow flushing with clean water. GE's corrosion product control features are consistent with the guidance in RG 8.8 (Rev. 3) and the SRP and are acceptable. The staff finds it to be acceptable. This item is resolved.

The ABWR is designed so that operation will not require alternate high- radiation area controls (pursuant to 10 CFR 20.203(c)(5)), as used in current operating BWRs. All high radiation areas (with greater than 1.0 mSv/hr (100 mrem/hr)) can be locked to control unauthorized access. No credit is taken for the relief provided in Section 12.6 of the BWR standard technical specifications (i.e., area locked at 10.0 mSv/hr (1,000 mrem/hr)). This design position meets the requirements of 10 CFR Part 20 and is acceptable.

12.3.2 Shielding

Radiation shielding will protect personnel against radiation exposure inside and outside the plant during normal operation, including AOOs, and during reactor accidents. All radioactive sources will be shielded on the basis of the access and exposure level requirements of the designed radiation zoning. Concrete used for radiation shielding meets the design guidance provided in RG 1.69, "Concrete Radiation Shields for Nuclear Power Plants" (Rev. 0). GE performed shielding calculations with the QAD-F, GGG, and DOT.4 computer codes. These are commonly accepted shielding calculational codes referenced in the SRP and are acceptable.

GE has not provided the thickness of specific radiation shields, contrary to the guidance of RG 1.70 (Rev. 3) and the acceptance criteria of the SRP. GE's position is that, because the system layouts and the physical dimensions of the as-procured radioactive system components are not known, the shielding requirements for these systems cannot be provided at this stage of the ABWR design. Therefore, the staff cannot conduct confirmatory calculations of shielding effectiveness. As an alternative method, GE has provided DAC to verify the adequacy of the ABWR shielding design. The staff's review of these DAC is discussed in Section 12.3.5.1 below. This alternative is acceptable, and the staff considers this item resolved.

The adequacy of the shielding in the upper drywell was identified as an Open Item 112 in the DSER (SECY-91-355). The biological shield surrounding the reactor vessel did not cover a significant portion of the top of the reactor vessel. A fuel handling mishap resulting in dropping an SFA across the reactor flange would result in extremely high dose rates in the upper drywell with this design. In addition to the radiological hazard presented by this AOO, it appears that raising an SFA in proximity of the vessel wall could result in significant radiation dose rates in the upper drywell. Amendment 21 revised the SSAR to reflect a design change to the shielding in the upper drywell, raising the biological shield to within 4 inches of the upper drywell ceiling. This design change would provide sufficient shielding during the normal withdrawal of SFAs from the reactor. However, a dropped SFA resting across the reactor flange would still produce significant radiation streaming into the upper drywell. Personnel in the upper drywell during this AOO could receive lethal radiation doses before they could escape. This was DFSER Confirmatory Item 12.3.2-1. In response to the staff's concerns, GE revised the design change to add a shielding ledge to the opening in the upper drywell ceiling. This ledge significantly reduces the radiation streaming into the upper drywell from a SFA resting on the reactor vessel flange. According to the staff's analysis, this shield design would reduce the dose rates in the upper drywell to less than 5 Sv/hr (500 rem) during a worse-case fuel-drop AOO. Therefore, the staff concludes that there is reasonable assurance that individuals could escape the upper drywell without receiving life-threatening radiation doses. Amendment 23, which revised the SSAR to reflect this final design, is acceptable to the staff. This item is resolved.

The shielding of the TIP system was identified as Open Item 106 in the DSER (SECY-91-355). As discussed in Section 12.1.2 of this report, the TIP drive and the TIP storage are located in separate shielded rooms. However, the conduit that guides the TIP from the reactor to its storage is virtually unshielded. This conduit shares the primary containment penetration with the lower drywell personnel access. Personnel at the lower drywell access hatch or in the access tunnel would be exposed to the unshielded activated TIP and the drive cable as they are retracted from the reactor core. DFSER Confirmatory Item 12.3.2-2 was identified on the basis of a March 26, 1992, draft SSAR amendment. Amendment 21 revised the description of the radiation design features associated with the TIP system. This amendment notes that the lower drywell access is located in a separate shielded room that can be locked to prevent access to these areas while the TIP is being withdrawn from the core. In addition, flashing alarms at the door to this room and at the lower drywell access hatch will warn personnel when power is applied to the TIP drives. Also, the TIP system will operate so that the TIP will be withdrawn in the high-speed mode, minimizing the transit time of the activated components through the unshielded portions of the system. These features ensure that the personnel radiation exposures resulting from the operation of the TIP system can be maintained ALARA and are acceptable. The staff finds it to be acceptable.

12.3.3 Ventilation

The ABWR ventilation systems are designed to protect personnel and equipment from extreme environmental conditions and to ensure that plant personnel are not inadvertently exposed to airborne contaminants exceeding the concentration limits given in 10 CFR Part 20. Design features intended to maintain personnel exposures ALARA include the following:

- Airflow between areas potentially having airborne contamination will always be from the area of lower potential contamination to the area of higher potential contamination.

- Negative or positive pressure will be used in areas to prevent exfiltration or infiltration of possible airborne contamination, respectively.

- The control room ventilation has dual fresh air intakes designed so that at least one intake will be free of contamination following a loss-of-coolant accident.

These design features are in accordance with the guidelines of RG 8.8 (Rev. 3) and are acceptable. However, as noted in Section 12.2 of this report, the expected leakage of radioactive fluids from plant systems cannot be determined at this stage of the ABWR design. Without this source term, GE is not able to provide the concentrations of airborne contamination in cubicles, rooms, and corridors as specified in the SRP. Therefore, the staff cannot verify that the plant ventilation system design meets the criteria in the SRP. This was identified as Open Item 113 in the DSER (SECY-91-355). As an alternative, GE provided DAC that require the COL applicant to calculate the expected concentrations of airborne radionuclides as specified in the SRP, to verify that adequate ventilation is provided. The staff identified DFSER Confirmatory Item 12.3.3-1 on the basis of a May 1, 1992, draft SSAR amendment. Section 12.3.5.2 of this report contains the staff's evaluation of these DAC. Amendment 21 added Appendix 12A to the SSAR which describes the calculational methods and assumptions that will be used to satisfy the DAC in Table 3.2.b of the CDM. These calculational methods and assumptions are consistent with provisions of the SRP and are acceptable. The staff finds it to be acceptable. This item is resolved.

12.3.4 Area Radiation and Airborne Radioactive Monitoring Instrumentation

Open Item 114 in the DSER (SECY-91-355), questioned the description of the ABWR area radiation monitoring system. GE revised the SSAR with the following information. The area radiation monitoring system consists of 25 gamma sensitive detectors and their associated digital monitors. The detectors are in key locations of the plant and will have operating ranges (sensitivity) commensurate with the expected radiation levels in the areas. Monitored radiation levels will be recorded and indication will be provided in the control room. These area monitors will be powered from the non-1E vital 120-Vac bus. The monitors will have local audible alarms with adjustable settings (both up-scale and down-scale) to warn personnel of abnormal conditions such as higher-than-normal radiation levels or detector failure. High-range radiation accident monitors that meet the criteria of RG 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident" (Rev. 3), are provided in the RHR equipment area. In addition, to assess the magnitude of the release of radioactive material from the core during an accident, four high-range gamma sensitive ion chambers in the containment will be able to measure up to 0.72 c/Kg per second (10^7 R/hr). The staff concludes that the area radiation monitoring system meets the applicable criteria in RG 8.8 (Rev. 3), RG 1.97 (Rev. 3), and the provisions in Item II.F.1.3 of NUREG-0737 that are required by 10 CFR 50.34(f)(2)(xvii)(D) and is acceptable. The COL applicant will address the operational considerations, such as monitor alarm set points, listed in RG 1.70 (Rev. 3) Section 12.3.4. This was DFSER COL Action Item 12.3.4-1. GE has included this information in a markup of SSAR Section 12.3.4 dated April 16, 1993. GE has also included this action item in the SSAR and the staff finds it to be acceptable.

The staff noted in the DSER (SECY-91-355) that criticality accident monitors were not provided in the ABWR design to meet the requirements of 10 CFR 70.24 as provided in the SRP. In response to the staff's request, GE amended the SSAR to state that these monitors are unnecessary because the ABWR is designed to ensure subcritical conditions during fuel handling and storage. Several licensees of operating BWRs with similar design features and fuel handling procedures have received a license condition exempting them from this 10 CFR 70.24 requirement. The requirements of 10 CFR Part 70 are outside the scope of this review. The COL applicant will provide information showing that their plant meets the requirements of 10 CFR 70.24 or request an exemption. This was DFSER COL Action Item 12.3.4-2. GE has included this action item in a markup of SSAR Section 12.3.4 dated April 16, 1993. GE has also included this action item in the SSAR and the staff finds it to be acceptable.

Monitoring of airborne radioactive materials in nuclear power plants typically is provided by fixed continuous air monitors. These monitors sample the ventilation air exhausted from plant areas having the highest potential for radioactivity release. Movable continuous air monitors are positioned in plant areas that have a potential for airborne radioactivity release during certain operating modes (i.e., an area where a radioactive system is opened during maintenance) to supplement the fixed monitors. GE has not described the airborne monitoring for the ABWR design. As discussed in Section 12.3.3 of this report, the expected concentrations of airborne radionuclides cannot be determined at the current level of the ABWR design detail. As an alternative, GE has provided DAC that would require the COL applicant to verify that airborne monitors provided in the final ABWR design meet the criteria of the SRP. The staff's review of these DAC is in

Section 12.3.5 below. The COL applicant will address the operational considerations, such as the procedures for operation and calibration of the monitors as well as the placement of the movable monitors, in the COL application. This was DFSER COL Action Item 12.3.4-3. GE has included this information in a markup of SSAR Section 12.3.4 dated April 16, 1993. GE has also included this action item in the SSAR and the staff finds it to be acceptable.

12.3.5 Certified Design Material

The staff initially identified three areas where the level of design detail in the SSAR did not allow the staff to conclude that the ABWR design meets the acceptance criteria in Chapter 12 of the SRP. These areas are the adequacy of the plant radiation shielding, the adequacy of the plant ventilation system, and the adequacy of the plant airborne radionuclide monitoring system. As an alternative, GE provided DAC requiring the COL applicant to perform shielding analysis and airborne radionuclide concentration calculations that will be verified by the ITAAC during plant construction to verify that the final ABWR design is acceptable. Details of the staff's review of these DAC follow.

12.3.5.1 Plant Shielding DAC

Chapter 12 of the SSAR contains layout drawings of the plant that indicate the designed maximum radiation level (or zone) for each room, equipment cubicle, and operating space during normal power operations, shutdown operations and accident conditions. As discussed in Section 12.2 above, the piping layout and component selection have not been set for the ABWR systems; therefore, parameters such as source strength and geometry needed to verify the adequacy of the radiation shields around these systems are not available. In addition, nitrogen-16 gammas from the plant can significantly contribute to offsite dose rates. The adequacy of the plant shielding needed to comply with the radiation dose limits for individual members of the public in 10 CFR Part 20 cannot be verified since the turbine design and site-specific parameters such as distance to the site boundary are unknown.

GE has submitted DAC for plant shielding in Table 3.2.a of the CDM. These DAC require the COL applicant to verify the adequacy of (1) the shielding around rooms and spaces during normal operations and shutdown conditions, (2) the shielding and temporary shield space provided between plant systems during maintenance activities, (3) the shielding provided around vital plant areas during accident conditions (Three Mile Island (TMI) Action Plan Item II.B.2 (10 CFR 50.34((f)(2)(vii)), and (4) the plant shielding needed to limit public dose. The staff's review indicates that the analysis assumptions, methods, and acceptance criteria in these DAC are consistent with the criteria in the SRP. Therefore, the staff concludes that compliance with these DAC, as supplemented by the information in SSAR Sections 12.3.2, is acceptable to adequately address the relevant requirements in 10 CFR 50.34(b)(3) and 10 CFR Part 20 concerning the limitation of radiation exposures to personnel, including the requirement to maintain doses ALARA as supplemented by the guidance in RG 8.8 (Rev. 3); 10 CFR 50.34(f) and GDC 19 with respect to operator access to plant areas during and following a reactor accident as supplemented by the guidance in Item II.B.2 of NUREG-0737; and GDC 61 regarding adequate shielding of fuel storage and handling, radioactive waste, and other systems which may contain radioactivity. GE provided a revised set of design descriptions and ITAAC (including DAC). The adequacy and acceptability of the ABWR Tier 1 material and ITAAC (including DAC) are evaluated in Section 14.3 of this report.

12.3.5.2 Ventilation and Airborne Monitoring DAC

The level of detail in the current ABWR design is not sufficient to provide the expected airborne concentrations in rooms and operating areas within the plant as specified in RG 1.70 (Rev. 3). Therefore, GE has not provided a description of the airborne monitoring system consistent with the criteria in the SRP.

GE has submitted DAC for ventilation and airborne monitoring in Table 3.2.b of the CDM. These DAC requires the COL applicant to calculate the expected concentrations of airborne radioactivity in each equipment cubicle, corridor, and operating area that require personnel access. These DAC also require an analysis by the COL applicant to identify those areas of the plant that require continuous monitoring of airborne radioactive materials. The staff's review indicates that the assumptions and acceptance criteria in these DAC are consistent with the criteria in the SRP. Therefore, the staff concludes that compliance with these DAC, as supplemented by the information in SSAR Sections 12.3.3 and 12.3.4 and Appendix 12A, will meet the relevant requirements in 10 CFR 50.34(b)(3) and 10 CFR Part 20 concerning the limitation of radiation exposures to personnel from airborne radioactive material, including the requirement to maintain doses ALARA as supplemented by the guidance in RG 8.8 (Rev. 3); GDC 61 regarding adequate shielding of fuel storage and handling, radioactive waste, and other systems which may contain radioactivity; and the requirements in 10 CFR Part 20, 10 CFR 50.34(f) and GDC 64 related to in-plant monitoring of airborne radioactive materials during routine operating conditions.

The adequacy and acceptability of the ABWR Tier 1 material and ITAAC (including DAC) are evaluated in Section 14.3 of this report.

12.3.5.3 Radiation Design Features

In the DSER (SECY-91-355), the staff identified a number of radiation design features to be addressed in the ABWR Tier 1 design description and ITAAC. This was identified as DFSER Open Item 12.3.5.3-1. GE provided a revised set of design descriptions and ITAAC (including DAC). The adequacy and acceptability of the ABWR Tier 1 material and ITAAC (including DAC) are evaluated in Section 14.3 of this report. On the basis of this evaluation, this item is resolved.

12.3.6 10 CFR 50.34(f): TMI-Related Items

SSAR Section 12.3 addresses two items from the TMI Action Plan (NUREG-0660), II.F.1.3 (10 CFR 50.34(f)(2)(xvii)(D), and II.B.2 (10 CFR 50.34(f)(2)(vii)). Item II.F.1.3 requires that high-range radiation accident monitors be provided in the containment. Item II.F.1.3 specifies that high-range dose monitors be capable of detecting dose rates up to 10^8 rads per hour. NUREG-0737 modified this position to specify that gamma sensitive monitors be capable of reading up to 10^7 R per hour. As discussed in Section 12.3.4 of this report, the ABWR design has four monitors (two in the drywell and two in the suppression chamber) that meet the provisions of Item II.F.1.3. Item II.B.2 specifies that radiation shielding be provided so that operators can access vital equipment in the plant during an accident without receiving excessive radiation dose. As discussed in Section 12.3.5.1 of this report, the DAC in Table 3.2.a of the CDM require the COL applicant to demonstrate compliance with Item II.B.2, as part of the analysis to verify the adequacy of the plant's radiation shielding.

The COL applicant will be responsible for demonstrating compliance with II.B.2 because the final hardware and system design specifications need as inputs to shielding calculations are not available now.

12.4 Dose Assessment

The staff has audited the dose assessment in Section 12.4 of the SSAR for completeness against the guidelines in RG 1.70 (Rev. 3) and against the criteria set forth in SRP Section 12.3.II.5. This review consisted of ensuring that GE had either committed to following the criteria of RGs and staff positions in Section 12.3 of the SRP or provided acceptable alternatives. In addition, the staff selectively compared the dose assessment made by GE for specific functions against the experience of operating BWRs. Details of the review follow.

GE provided an assessment of the radiation dose that would be received by operating a plant of the ABWR design. Estimated person-rem doses for major work within areas of the plant during maintenance and refueling periods and for power operations are given in SSAR Table 12.4-1 and result in an estimated total annual dose of 0.989 person-sievert (98.9 person-rem).

In Open Item 115 in the DSER (SECY-91-355), the staff identified several deficiencies in GE's dose assessment, including mathematical errors, inconsistencies between text and tables, a lack of bases for assumptions, and a level of detail that was not consistent with the guidance in RG 8.19, "Occupational Radiation Dose Assessment in Light-Water Reactor Power Plants--Design Stage Man-Rem Estimates" (Rev. 1). GE amended the SSAR to correct the math errors and inconsistencies noted and to give the bases for the assumptions and values used in the assessment.

In GE's dose assessment, the stay times and frequencies for each task are based on a detailed task analysis of maintenance activities in a BWR with a similar containment design (MARK III containment). Average dose rates are based on past experiences for similar tasks. These values were then adjusted to account for ABWR design features to obtain the expected doses. The cumulative annual dose of 0.989 person-sievert (98.9 person-rem) for personnel operating an ABWR plant is consistent with the Electric Power Research Institute design guideline of 1.0 person-sievert (100 person-rem) per year and compares favorably with the average current BWR experience (which is more than twice the ABWR projected dose). Although not in the format specified in RG 8.19, this detailed dose assessment meets the intent of RG 8.19 and is acceptable. The staff considers this item resolved.

12.5 Organization

The organization required to implement an effective health physics program and ensure that radiation exposures are within the limits of 10 CFR Part 20 and are ALARA is outside the scope of this review. The COL applicant seeking an operating license by referencing the ABWR certified design will address this concern to the level of detail discussed in RG 1.70 (Rev. 3). This was DFSER COL Action Item 12.5-1. GE has included this information in a markup of SSAR Section 12.5.3.1 dated April 16, 1993. GE has also included this action item in the SSAR and the staff finds it to be acceptable.

12.5.1 10 CFR 50.34(f): TMI-Related Items

The regulation in 10 CFR 50.34(f)(2)(xxvii) requires in-plant radiation and airborne radioactivity monitoring in accordance with Item III.D.3.3 of the TMI Action Plan. Item III.D.3.3 requires that operating reactors be capable of accurately measuring radio-iodine concentrations in plant areas under accident conditions. The NUREG-0737 clarification of Item III.D.3.3 specifies that this capability use portable instruments and includes requirements for training and procedures for the use of these instruments.

These programmatic requirements are outside the scope of this review and have been identified as items to be addressed by the COL applicant. This was identified as DFSER COL Action Item 12.5.1-1. Appendix A to SSAR Chapter 1, Section 1A.3.3, as revised by Amendment 23, identifies post-accident radio-iodine monitoring as an issue to be addressed by the COL applicant. GE also has included this information in a markup of SSAR Section 12.5.3.2 dated April 16, 1993. GE has included this action item in the SSAR and the staff finds it to be acceptable.

13 CONDUCT OF OPERATIONS

13.1 Organizational Structure of Applicants

The staff completed its review of Standard Safety Analysis Report (SSAR) Section 13.1, "Organizational Structure of Applicants," and finds this section to be adequate. The staff agrees that the information related to the combined license (COL) applicant's organizational structure is outside the scope of the advanced boiling water reactor (ABWR) standard plant design. This information will be the responsibility of the COL applicant referencing the ABWR design at the COL phase described in 10 CFR 52.79(b).

13.2 Training

The staff's review of SSAR Section 13.2 was based on the current regulatory requirements in 10 CFR 52.47, 10 CFR 50.34(g), and 10 CFR 50.34(f) and the guidance in SRP Section 13, NUREG-0700, and NUREG-0933. The staff developed additional review criteria to provide a basis for the review of aspects of the ABWR human factors engineering (HFE) program that were not fully addressed by the previously mentioned documents. These criteria are contained in the staff's "HFE Program Review Model (PRM) and Acceptance Criteria for ABWR," which was forwarded to the Commission in SECY-92-299 dated August 27, 1992, and is attached as Appendix J of this report. The HFE PRM considered aspects of training as it pertained to the verification and validation (V&V) of the main control room and remote shutdown system designs.

GE has not included training development in the scope of the ABWR design certification application and has stated that development of training will be the responsibility of the COL applicant. However, information on the incorporation of operating experience (Three Mile Island (TMI) I.C.5) and preoperational and low power testing (TMI I.G.1) into training programs is provided by GE to the COL applicant in SSAR Section 13.2.

Although training is not considered to be part of the information required for certification of the ABWR design, the staff identified the following issues related to training during its review of SSAR Chapter 18, "Human Factors Engineering:" training materials, the use of a simulator for training, and the incorporation of operational experience into training programs. Each of these items is discussed below.

DSER Issue 18.22:Training Materials

In the DSER the staff stated that it "expects GE to develop and submit for certification a detailed program description for developing the training material as part of the design certification for the ABWR." Following issuance of the DSER, the staff determined that training materials were already a part of the established licensing review process under 10 CFR Part 50 for the COL applicant and, therefore, did not need to be addressed in the ABWR design certification review under 10 CFR Part 52. The staff agrees that the submittal of training materials is beyond the scope of the ABWR design certification and finds GE's approach to be acceptable; therefore, this item is resolved.

TMI Action Item I.A.4.2

10 CFR 50.34(f)(2)(i) corresponds to TMI Action Item I.A.4.2, "Long Term Simulator Training Upgrade," with regard to simulator capabilities. GE states that "simulator facilities for use in performing operator training are outside the scope of the standard plant design certification." This is acceptable because training will be addressed by the COL applicant. This was draft final safety evaluation report (DFSER) COL Action Item 18.7.2.2-1. GE has included the COL action item, that the operator training program meets 10 CFR Part 50, as Item 18.8.8 in SSAR Section 18.8, and the staff finds this approach to be acceptable.

TMI Action Item I.C.5

The staff reviewed TMI Action Item I.C.5, "Feedback of Operating Experience," on the incorporation of operational experience into the training and procedure development programs. The staff determined that development of detailed procedures and training materials is beyond the scope of the ABWR design certification and is the responsibility of the COL applicant. GE has included the training and procedure development process as a COL license information item in SSAR Sections 13.2 and 13.5, and the staff finds this approach to be acceptable.

TMI Action Item I.G.1

The staff reviewed TMI Action Item I.G.1, "Training Requirements for Preoperational and Low-Power Testing." The staff determined that I.G.1 is beyond the scope of ABWR certification and is the responsibility of the COL applicant. GE has included the training requirements for preoperational and low power testing activities as a COL license information item in SSAR Section 13.2.3.2, and the staff finds this approach to be acceptable.

The staff also notes that V&V of training materials will be examined further during the V&V of the ABWR main control room and remote shutdown system as described in Appendix 18E of the SSAR. Therefore, the issue of training is resolved for the ABWR design certification.

13.3 Emergency Planning

In Section 13.3 of the SSAR, GE states that emergency planning is not within the scope of the ABWR design and that the COL applicant will provide emergency plans in accordance with 10 CFR 50.33(g) and 52.72(d). The staff agrees that the requirement to provide the emergency plans is the responsibility of the COL applicant referencing the ABWR design and will depend significantly on both plant and site-specific characteristics. This was identified as DFSER COL Action Item 13.3-1. GE included this action item in Amendment 31 of the SSAR. The staff finds it to be acceptable.

Nevertheless, GE also acknowledges that there are design features, facilities, functions, and equipment necessary for emergency planning that must be considered in the design basis of a standard plant. These are addressed in SSAR Table 13.3-1, "ABWR Design Considerations for Emergency Planning Requirements," which specifies a technical support center (TSC) that complies with all of the TSC design requirements and is located in the service building adjacent to the control room. (The TSC is shown in SSAR Figure 1.2-19, "Control and Service Building, Arrangement Plan at Elevation 7900 mm.") GE further states that the TSC will contain the necessary facilities and equipment called for in Section 2 of NUREG-0696 ("Functional Criteria for Emergency Response Facilities," U.S. NRC, February 1981).

The staff performed its review in accordance with the requirements of 10 CFR 50.47(b), Appendix E to 10 CFR Part 50, and 10 CFR 50.34(f)(2)(xxv), which requires an onsite TSC, an onsite operational support center (OSC), and a nearsite emergency operations facility (EOF). The staff's review also considered the guidance provided in NUREG-0654 and NUREG-0696. It is the staff's position that the facilities and equipment for the ABWR standard plant TSC should be compatible with the control room and meet the applicable criteria of NUREG-0696. In this regard, the staff noted in the DFSER that whereas GE specified a TSC to support 20 people, NUREG-0696 specifies the following:

> The TSC working space shall be sized for a minimum of 25 persons, including 20 persons designated by the licensee and five NRC personnel. This minimum size shall be increased if the maximum staffing level specified by the licensee's emergency plan exceeds 20 persons.

Therefore, the staff stated in the DFSER that the TSC for the ABWR standard plant should be sized for 25 persons and be compatible with the control room in order to meet the criteria of NUREG-0696. This was DFSER Open Item 13.3-1. GE revised Table 13.3-1 in Amendment 25 to the SSAR to indicate that the TSC will be of sufficient size to support 25 people. The staff finds it to be acceptable. This item is resolved.

The ABWR standard plant design also includes considerations for decontamination of onsite individuals in the service building adjacent to the main change rooms as shown in Figure 1.2-20 of the SSAR. The staff finds these design considerations for an onsite decontamination facility acceptable for meeting the requirements of 10 CFR Part 50, Appendix E, Section IV.E.3.

GE considers other facilities that support emergency planning to be outside the scope of the ABWR standard plant design scope. These include an offsite EOF for the management of overall licensee emergency response, including coordination with Federal, State and local officials. The staff agrees that the EOF is not within the scope of the ABWR standard plant design, but must be addressed by the COL applicant referencing the ABWR standard plant design. This was identified as DFSER COL Action Item 13.3-2. GE included this action item in Amendment 31 of the SSAR. The staff finds it to be acceptable.

GE originally considered an onsite OSC (assembly area) separate from the control room and TSC where licensee operations support personnel report in an emergency to be outside the scope of the ABWR standard plant design. However, the staff noted in the DFSER that an OSC should be provided as part of the ABWR standard plant design. This was DFSER Open Item 13.3-2. GE revised Table 13.3-1 in Amendment 25 to the SSAR to state that the ABWR standard plant will comply with all of the OSC design requirements, and that the lunch room adjacent to the TSC in the service building will be identified as the OSC. The staff finds it to be acceptable. This item is resolved.

<u>Certified Design Material</u>

The requirements for the TSC and OSC were not covered in the Tier 1 design descriptions or inspections, tests, analyses, and acceptance criteria (ITAAC). This was DFSER Open Item 13.3-3. GE has provided a revised set of design descriptions and ITAAC. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Section 14.3 of this report. On the basis of this evaluation, this item is resolved.

<u>10 CFR 50.34(f):TMI-Related Items</u>

SSAR Section 13.3 addresses TMI Action Plan (NUREG-0660) Item III.A.1.2 (10 CFR 50.34(f)(2)(xxv)).

This item requires the COL applicant to upgrade its emergency support facilities by establishing a TSC, OSC, and a near site EOF for command and control, support, and coordination of onsite and offsite functions during reactor accident situations. As discussed in Section 13.3 of this report, the ABWR design provides for a TSC and an OSC. The near-site EOF is considered by the staff not to be within the scope of ABWR standard plant design and will be addressed by the COL applicant referencing the ABWR standard plant design (DFSER COL Action Item 13.3-2). GE has included this action item in the SSAR. The staff finds it to be acceptable.

13.4 Review and Audit

The staff determined that the review and audit information is outside the scope of the ABWR standard plant design. The COL applicant will provide the necessary information on reviews and audits for plant operation. It will be reviewed in detail during the COL stage. GE has included this action item in the SSAR and the staff finds it to be acceptable.

13.5 Plant Procedures

The staff's review was based on the current regulatory requirements established in 10 CFR 52.47, 10 CFR 50.34(g), and 10 CFR 50.34(f) and the guidance contained in SRP Section 13, NUREG-0700 and NUREG-0933. The staff developed additional review criteria to provide a basis for the review of aspects of the ABWR HFE program that were not fully addressed by the previously mentioned documents. These criteria are contained in the staff's HFE PRM, which is attached as Appendix J of this report. The HFE PRM considered aspects of plant procedures as they pertained to the V&V of the main control room and remote shutdown system designs.

GE has not included procedure development in the scope of its ABWR design certification application and has identified procedure development as a COL responsibility in SSAR Section 13.5. GE's description of this COL license information item is consistent with the staff's HFE PRM - Element 7, procedures, developed to support the review of the ABWR HFE effort. The HFE PRM is described in Appendix J of this report.

Although plant procedures are not considered to be part of the information required for certification of the ABWR design, the staff identified several issues related to procedure development during the review of SSAR Chapter 18, "Human Factors Engineering." Items regarding procedure development as well as differences between Element 7 of the HFE PRM and SSAR Section 13.5 are discussed below.

(1) <u>General Criterion 5 of Element 7 of the HFE PRM</u>

General Criterion 5 of Element 7 of the HFE PRM states: "All procedures shall be verified and validated. A review shall be conducted to assure procedures are correct and can be performed. Final validation of operating procedures shall be performed in a simulation of the integrated system as part of V&V activities described in Element 8." Although GE has not included a requirement for final validation of operating procedures, in its design scope, as a COL action item, GE has stated that procedures will be available for the human-system interface (HSI) V&V activities (as specified in SSAR Section 13.5.3.3, - Item e). These activities are described further in SSAR Appendix 18E and SSAR Table 18E-4. The details of GE's V&V process are provided in SSAR Table 18E-4 and require that final procedures be included by the COL applicant in the V&V activities for both the main control room and the remote shutdown system. On the basis of SSAR Section 13.5.3.3 and the inclusion of procedures in SSAR Table 18E-4, the staff finds specification of this criterion in the SSAR as a COL action item to be acceptable.

(2) <u>General Criterion 6 of Element 7 of the HFE PRM</u>

General Criterion 6 of Element 7 of the HFE PRM states: "An analysis shall be conducted to determine the impact of providing computer-based procedures and to specify where such an approach would improve procedure utilization and reduce operating crew errors related to procedure use." This activity is not part of GE's design scope. GE states that an analysis of computer-based procedures will be conducted as part of the task analysis to be conducted by the COL applicant and evaluated further as part of the HSI. The description of the HSI requirements is contained in Table 3.1 of the ABWR certified design material. Because the computerization of procedures is an aspect of the HSI design implementation, the incorporation of the analyses as part of HSI Element 4 is acceptable. Therefore, the staff finds the specification of COL completion of this criterion to be acceptable.

Although plant procedures are not considered to be part of the information required for design certification of the ABWR, the staff identified the following issues related to procedures during its review of GE's HFE program: procedure development (DSER — SECY-91-320 — Issue 18.21), guidelines for updating procedures (HF-4.4), short-term accident procedures review (TMI Action

Item I.C.1), incorporation of operating experience (TMI Action Item I.C.5) and a long-term plan for upgrading procedures (TMI Action Item I.C.9). Each of these items is addressed below.

DSER (SECY-91-320) Issue 18.21: Procedure Development

In the DSER, the staff stated that system-level operating procedures would be developed concurrent with the development of the ABWR systems design. These procedures and the associated task analyses on which the HSI requirements are based are not included in the design certification; thus, they could not be evaluated.

The staff determined that development of detailed procedures and associated training materials is beyond the scope of the ABWR design certification and is the responsibility of the COL applicant. This was DFSER Confirmatory Item 18.9.2.2.7-1. GE has included the COL license information in the SSAR Section 13.5 which adequately addresses procedure development for the certification of the ABWR design. The staff finds GE's approach to be acceptable. Therefore, Confirmatory Item 18.9.2.2.7-1 is resolved.

USI/GSI Item HF-4.4 and TMI Action Items I.C.1, I.C.5 and I.C.9

The staff reviewed Item HF-4.4, "Guidelines for Upgrading Other Procedures," TMI Action Item I.C.1, "Short-Term Accident and Procedures Review," TMI Action Item I.C.5, "Feedback of Operating Experience," and TMI Action Item I.C.9, "Long Term Plan for Upgrading Procedures." It was determined that development of detailed procedures is beyond the scope of the ABWR certification and is the responsibility of the COL applicant. GE has included the procedure development process as a COL license information item in SSAR Section 13.5.3. SSAR Section 13.5.3.1 states that the methods and criteria for the development, V&V, implementation, maintenance and revision of procedures will include consideration of TMI Action Items I.C.1, I.C.5, and I.C.9. The staff finds GE's approach to be acceptable. The staff interprets this process to include the analysis of HF-4.4.

The staff also notes that V&V of plant procedures will be examined further during the V&V of the ABWR control room and remote shutdown system as described in GE's certified design material, Table 3.1, "Human Factors Engineering," and in Appendix 18E of the SSAR. Therefore, GE's proposed treatment of plant procedures is acceptable for the purposes of ABWR design certification.

13.6 Physical Security

13.6.1 Preliminary Planning

SSAR Section 13.6 states that preliminary planning is not required because the security plan will be the responsibility of COL applicants who reference the ABWR standard plant design. The staff finds this approach to be acceptable. Since the COL application will include a physical security plan, safeguards contingency plan, and a guard qualification and training plan, a preliminary planning submittal is not necessary for the design certification.

13.6.2 Security Plan

The SSAR states that the development of the security plan is beyond the scope of the ABWR standard plant design. The staff finds this approach to be acceptable. In addition to the action items listed in Section 13.6.3 of the SSAR, the COL applicant will provide site-specific security, contingency, and guard training plans in accordance with 10 CFR 50.34 and 10 CFR Part 73. This was identified as DFSER COL Action Item 13.6.2-1. By SSAR Amendment 25, GE identified in Section 13.6.3.8 that the COL applicant will provide site-specific security, contingency, and guard training plans. The staff finds it to be acceptable.

The staff requires that at least 60 days before loading fuel, the COL applicant will confirm that the security systems and programs described in its physical security plan, safeguards contingency plan, and guard qualification and training plan have achieved operational status and are available for NRC inspection. Operational status means that the security systems and programs are functioning in entirety as they would when the reactor is operating and will remain so. The COL applicant's determination that operational status has been achieved must be based on tests conducted under realistic operating conditions of sufficient duration to demonstrate (1) that the equipment is properly operating and capable of long-term, reliable operation; (2) that procedures have been developed, approved, and implemented; and (3) that personnel responsible for security operations and maintenance have been appropriately trained and have demonstrated their capability of performing their assigned duties and responsibilities. This was identified as DFSER COL Action Item 13.6.2-2. GE has included this action item described above in the SSAR and the staff finds it to be acceptable.

13.6.3 Control of Access to Areas Containing Vital Equipment

Section 13.6.3 of the SSAR identifies a number of interfaces between the ABWR standard plant design and the remainder of the plant that must be addressed by COL applicants who reference the ABWR standard plant design.

The staff reviewed the interfaces and determined that these were not interfaces as described in 10 CFR 52.47 but were actions to be accomplished as part of the COL application. The staff found them to be acceptable as COL action items subject to the addition of the requirement that the applicant provide plant-specific security, contingency, and guard training plans in accordance with 10 CFR 50.34 and 10 CFR Part 73. This was identified as DFSER COL Action Item 13.6.3-1. GE has included this action item as described above in the SSAR and the staff finds it to be acceptable.

13.6.3.1 Introduction

Section 13.6.3.1 of the SSAR states that SSAR Section 13.6.3 deals with the control of access to areas containing vital equipment.

13.6.3.2 Design Bases

Section 13.6.3.2 of the SSAR states that security functions described in Section 13.6.3 are incorporated into the overall ABWR design so that the plant is in compliance with the requirements of 10 CFR Part 73.

The Electric Power Research Institute (EPRI) Advanced Light Water Reactor (ALWR) Utility Requirements Document (Volume II, Revision 1, Chapter 11, Section 8.4.1) specifically requires the protected area lighting to be powered from an uninterruptible power source. In its response to a request for additional information (RAI) Q910.18, GE identified a site security load on the non-Class 1E vital (uninterruptible) load list (SSAR Table 20B-1), but the staff considered the description of this interconnection to be insufficiently defined. In response to staff comments, GE added SSAR Section 19B.3.12, which clarified the connection between the security system uninterruptible power requirements (to be later so determined by the plant-specific security system designer as to meet required security system performance) and the non-Class 1E vital power supply capacity. SSAR Section 19B.3.12 requires that the site security system be powered from a non-Class 1E vital (uninterruptible) ac power source. The protected area boundary lighting subsystem is the only exception. At the discretion of the COL applicant, the protected area boundary lighting subsystem may be powered from a non-Class 1E vital (uninterruptible) ac power source or from an interruptible power source provided adequate compensatory measures are established by the site's physical security implementation plan. The staff considers this resolution acceptable and this portion of DFSER Open Item 1.1-1 is resolved.

In NRC Information Notice 83-83, the staff suggested that new plant designs that make extensive use of solid-state devices in instrument and control circuits may experience reactor system malfunctions and spurious actuation as a result of portable communication devices in their vicinity. In RAI Questions (Q)910.10 and Q910.17, the staff asked that radio-frequency interference design criteria be established to ensure that security personnel within the reactor and control building could maintain radio communication without adversely affecting plant operation. GE's response to Q910.17 referenced discussions of system tolerance to electromagnetic interference. The staff finds that the plant security systems criteria in SSAR Section 9.5.13.11 and the amendment to SSAR Appendix 7A, in response to a concern raised in DSER Section 7.1.3.3, adequately resolve staff's concern.

13.6.3.3 Vital Areas

Section 13.6.3.3 of the SSAR itemizes by location the plant equipment to be considered vital equipment pursuant to 10 CFR 73.2 and the vital areas containing that equipment. SSAR Figures 13.6-1 through 13.6-14 outline the vital areas.

In RAIs Q910.9, Q910.11, and Q910.20, the staff questioned the completeness of the list of vital equipment in SSAR Section 13.6.3.3. GE clarified this list on February 22, 1991, after discussions with staff. The staff is satisfied that the list of vital equipment includes all active and passive plant equipment essential to safe shutdown of the reactor, including necessary support systems, the reactor vessel and the remainder of the reactor coolant system pressure boundary within the primary containment, the suppression pool, spent fuel in the fuel pool, and any associated piping, equipment, and controls whose failure could result in an offsite release in excess of 10 CFR Part 100 limits. The staff finds this to be compatible with NRC Review Guideline 17 (January 23, 1978, memorandum from R. Clark to safeguards licensing staff). Prior to issuance of a COL, the staff's review of the designation of equipment as vital in plant-specific applications will focus on plant support equipment outside the scope of the certified ABWR design. In addition, 10 CFR 73.55(e) requires that the central alarm station be considered a vital area and secondary power supply system for alarm annunciator equipment and that non-portable communications equipment be located in vital areas. The secondary alarm station also is typically on site and treated as a vital area. Vital area classification of the central and secondary alarm stations was identified as DFSER COL Action Item 13.6.3.3-1. In SSAR Amendment 25, GE stated in Section 13.6.3.8 that the COL applicant will provide site-specific security, contingency, and guard training plans. Specifically, as stated in Section 13.6.3.10, the COL applicant will provide the classification of the control and secondary alarm stations. Vital area classification will be addressed at the time of the plant-specific security plan review. The staff finds it to be acceptable.

The staff expects that at least 60 days before loading fuel, a licensee will have confirmed that no portion of as-built vital systems is located outside of designated vital areas or can be prevented from performing their safety functions from outside the vital areas (e.g., by reach-rod valve manipulation). This verification should include piping, valves, and motor control centers that are required for maintaining boundary integrity, for performing the safe reactor shutdown cooling function, and for isolating safety-related equipment from non-safety-related equipment. This was identified as DFSER COL Action Item 13.6.3.3-2. In SSAR Amendment 25, GE specified in Section 13.6.3.11 that the COL applicant will confirm that locations of vital systems and system operations meet the above requirements. The staff finds it to be acceptable.

The plant-specific licensing review of the security and contingency response plan also will include an evaluation of whether the security response force's capability to interdict the violent external assault postulated in 10 CFR 73.1(a)(1)(i) properly accounts for the minimum penetration delay provided by the vital area barriers and doors. This was identified as DFSER COL Action Item 13.6.3.3-3. In SSAR Amendment 25, GE stated in Section 13.6.3.12 that the COL applicant will provide an evaluation of the interdiction capability of the security response force. The security response force's capability to interdict the violent external assault postulated in 10 CFR 73.1(a)(1)(i) will be addressed at the time of the plant-specific security plan review. The staff finds it to be acceptable.

13.6.3.4 Methods of Access Control

SSAR Section 13.6.3.4 describes, in general terms, the types of door controls that will be used to control access to vital areas. In response to RAIs Q910.12, Q910.21, and Q910.22, GE added statements to the SSAR that all doors and hatches connecting vital to non-vital areas are to be alarmed and emergency egress will not require keys or card readers. The staff finds this approach to be acceptable.

13.6.3.5 Access Control and Security Measures Through Exterior Doors to the Nuclear Island

SSAR Section 13.6.3.5 describes the specific security measures at portals into the reactor and control buildings from exterior areas and facilities of the remainder of the plant. For the DSER (SECY-91-235), the staff stated the types of door controls specified in SSAR Section 13.6.3.4 were generally acceptable, but insufficient detail was provided to determine compatibility with RG 5.12, Revision 0. This description of access control methods also did not address the positive control requirement of 10 CFR 73.55(d)(7)(i)(B) and the record-keeping requirement of 10 CFR 73.70(d), which requires logging individuals' times of entry to and exit from each vital area. This was identified as DSER (SECY-91-235) Open Item 36 and DFSER COL Action Item 13.6.3.5-1. In SSAR Amendment 25, GE stated in Section 13.6.3.13 that the COL applicant will demonstrate that door controls are compatible with RG 5.12, the positive control requirement of 10 CFR 73.55(d)(7)(i)(B) and the record-keeping requirement of 10 CFR 73.70(d). This issue will be considered during review of the plant-specific security plan. The staff finds it to be acceptable.

In RAI Q910.19, the staff asked why the parameters for environmental conditions in SSAR Appendix 3I should not apply to the design and qualification of security access control components. TMI Action Item II.B.2 (NUREG-0737) identifies areas for which environmental qualification of equipment necessary to ensure post-accident access may need to be considered. Although the "security center" is not safety related, it is included in NUREG-0737 because access to it may be necessary to give access to the rest of the plant. In NRC Information Notice 86-106, Supplement 2, the staff discussed an event at the Surry Power Station in which condensed steam saturated a security card reader and caused a short circuit in the card reader system for the entire plant. As a result, key cards would not open doors controlled by the security system. In the same event, a security communications system radio repeater was temporarily degraded because a thick layer of ice had formed on it from actuation of a carbon dioxide discharge nozzle. In response to RAI Q 910.19, GE stated that (1) this equipment is not safety related and is not required to operate under accident conditions, (2) the card reader design is required to preclude the possibility of failure of one card reader affecting the operation of any other card reader, (3) card reader doors are required to have a key-operated override, and (4) emergency exits are required to be designed so that personnel can exit without using keys or card readers. The staff considers this response to be consistent with currently accepted industry practice and is, therefore, acceptable.

Furthermore, SSAR Chapter 19, Appendix 19B.3.10, requires a COL applicant to evaluate the effects of the security system on required operator actions during all emergency modes of operation. The staff's position is that this analysis should include consideration of an emergency requiring evacuation of the control room in the control building to the remote shutdown panel in the reactor building. This was identified as DSER (SECY-91-235) Open Item 37 and DFSER COL Action Item 13.6.3.5-2. In SSAR Amendment 25, GE stated in Section 13.6.3.8 that the COL applicant will provide site-specific security, contingency, and guard training plans. Evaluation of compliance with the vital equipment prompt access requirements of 10 CFR 73.55(d)(7)(ii) will be resolved during review of the plant-specific security plan. The staff finds it to be acceptable.

13.6.3.6 Bullet-Resistant Walls and Doors, Security Grills, and Screens

SSAR Section 13.6.3.6 discusses bullet-resistant walls and doors and security grills and screens incorporated into the building design, with the stated intent of minimizing forcible access to the control room. In its responses to RAIs Q910.13, Q910.23, and Q910.24, GE did not resolve staff uncertainty as to the adequacy of barriers in all man-sized openings in physical barriers that separate other vital from non-vital areas. Also, the staff position on the effectiveness of the ventilation system barriers described in SSAR Section 13.6.3.6 remained as described in RAI Q910.13; that is, consideration may need to be given to how accessible, isolated, and hidden from view these barriers will be, as well as whether they can be penetrated with hand tools available on site. While SSAR Section 13.6.3.6 only addresses the main control room heating, ventilation, and air conditioning (HVAC) ducting and exterior air exhaust systems, SSAR Chapter 19, Appendix 19B.2.4(13) was changed to include the EPRI ALWR requirements on utility port openings (e.g., HVAC, cooling, and piping) through all vital or protected area boundaries, in accordance with EPRI Evolutionary Requirements Document, Chapter 9, Section 5.2.5.1, Revision 0. Specifically, the SSAR states that the ABWR design will minimize the use of utility port openings through all vital or protected area boundaries and will provide security access control of these utility ports. GE's change in Appendix 19B.2.4, which clarifies that the ABWR design will comply with the above ALWR requirements, satisfactorily resolved this issue.

Certified Design Material

The staff expects that at least 60 days before loading fuel, the COL applicant referencing the ABWR design will have confirmed that the as-built bullet-resistant feature of walls and doors and the penetration-resistant feature of barriers in HVAC ducting and exhausts, committed to in SSAR Section 13.6.3.6, have been installed in all locations required by the commitment. This inspection requirement should be included in appropriate building ITAAC and was identified as DFSER Open Item 13.6.3.6-1. In its submittal dated May 30, 1992, GE stated that the issue was not a Tier 1 ITAAC issue. Additional review by the staff agreed with GE that the issue did not meet the criteria to be a Tier 1 ITAAC requirement. This issue is discussed in detail in Section 14.3.2.5. Therefore, this item is resolved.

13.6.3.7 Compatibility With the Remainder of the Plant

SSAR Section 13.6.3.7 states that access control for the remainder of plant buildings, including the turbine building side of the main steam tunnel, must be compatible with the site-specific physical security program which will be the responsibility of the COL applicant who references the ABWR standard plant design. The staff agrees and notes that an acceptable security barrier to bar unauthorized access from the turbine building into the steam tunnel must also permit venting of steam into the turbine building in accordance with SSAR Section 9A.3.2 of Appendix A to Chapter 9.

14 INITIAL TEST PROGRAM

14.1 Preliminary Safety Analysis Reports Information

Section 14.1 of the standard safety analysis report (SSAR) is specified in Regulatory Guide (RG) 1.70 and the standard review plan (SRP). However, the preliminary safety analysis report information specified by RG 1.70 is only required for facilities licensed under 10 CFR Part 50, and is not required for a design certification under 10 CFR Part 52. This is because under Part 52, the staff gives its final design approval (FDA) for the design, rather than its preliminary design approval as done under Part 50. Thus, the design information in the SSAR is final rather than preliminary, and therefore, no information is required to be in this section of the ABWR SSAR.

The staff provided its preliminary evaluation of GE's draft certified design material in Section 14.1 of the DFSER. Subsequently, as requested by the staff, GE provided SSAR information discussing the certified design material in a new Section 14.3 of the SSAR. The staff's review of the certified design information is summarized in Section 14.3 of this report, which conforms with SSAR Section 14.3.

This section is not applicable to the ABWR application. The DFSER included a discussion of piping design in this section. This discussion has been moved to Section 3.12 of this report.

14.2 Initial Plant Test Programs

Introduction

Chapter 14, Section 14.2, of the ABWR SSAR, "Specific Information to be Included in Final Safety Analysis Reports," describes the plant initial test program (ITP) for the ABWR. The GE ABWR ITP consists of a series of tests categorized as construction, preoperational, or initial startup tests. The construction acceptance tests determine installation and functionality of equipment. Preoperational tests are those tests normally conducted prior to fuel loading to demonstrate the capability of plant systems to meet performance requirements. Initial startup tests begin with fuel loading and demonstrate the capability of the integrated plant to meet performance requirements. This report documents the staff's evaluation of the GE ABWR ITP.

The staff performed the review of the plant ITPs in accordance with Section 14.2, of NUREG-0800, the SRP, and RG 1.68 (Rev. 2), "Initial Test Program for Water Cooled Nuclear Power Plants." The staff reviewed thirteen areas relating to initial plant test programs, described in Chapter 14 of the SSAR, and submitted by GE as part of its design certification application. These areas of review are listed below:

- Summary of the Test Program and Objectives
- Organization and Staffing
- Test Procedures
- Conduct of Test Program
- Review, Evaluation, and Approval of Test Results
- Test Records
- Conformance of Test Program With Regulatory Guides
- Utilization of Reactor Operating and Testing Experience in the Development of Test Program
- Trial Use of Plant Operating and Emergency Procedures
- Initial Fuel Loading and Initial Criticality
- Test Program Schedule
- Individual Test Descriptions
- Combined License (COL) Information - ITP

The acceptance criteria used by the staff for this review were contained in SRP Section 14.2, Subsection II.

Background

A meeting was held with GE, on May 7, 1991, in order to discuss a list of questions, comments, and errata information generated by the staff's initial review of the SSAR and GE's potential responses to these items. Subsequently, a draft SSAR amendment was submitted by GE via a letter dated May 20, 1991, from R.C. Mitchell (GE) to C.L. Miller (NRC), in response to these items.

The DSER (SECY-91-355), dated November 5, 1991, consisting of acceptable and open items, was subsequently forwarded to GE for its use in further revising the SSAR. In October 1991, GE formally submitted SSAR Chapter 14, Amendment 20. GE subsequently submitted a response to the DSER open items via letter dated March 11, 1992, from R.C. Mitchell (GE) to R.C. Pierson (NRC) and additional SSAR markups were subsequently provided to the staff. These changes were incorporated in SSAR Amendments 21, 22, and 23.

Evaluation

The staff reviewed the GE ABWR ITP in accordance with SRP Section 14.2. This evaluation includes information in staff requests for additional information (RAIs) following its earlier review, GE's letter of May 20, 1991, in response to the staff's RAIs and the staff's findings regarding each response. The staff also evaluated GE's responses to DSER open items as contained in its letter of March 11, 1992. This evaluation also includes all other information supplied to the staff as docketed markups to the SSAR, as well as changes contained in subsequent SSAR amendments. This evaluation for the ABWR included open items, confirmatory items, and resolution of those sections of the SSAR that the staff initially found to need modification or additional information.

Based on the review of the GE ABWR ITP description in SSAR Chapter 14 and the responses to the RAIs and the DSER open items, DFSER open items and confirmatory items, the staff concludes that the ITP was generally comprehensive and covered all areas of staff concern. The staff also conducted an in-depth review of system-specific testing requirements within each test abstract. The staff concludes that GE provided a sufficient level of detail to adequately describe system-specific test prerequisites and acceptance criteria. The staff also reviewed cross references to acceptance criteria information in other parts of the SSAR and finds them acceptable.

14.2.1 Summary of Test Program and Objectives

As stated in SSAR Section 14.2.1, the objectives of the ITP will be to

- ensure that construction is completed and acceptable
- demonstrate the capability of structures, components, and systems to meet performance requirements
- effect fuel loading in a safe manner
- demonstrate, where practical, that the plant is capable of withstanding anticipated transients and postulated accidents
- evaluate and demonstrate, to the extent possible, that plant operating procedures provide assurance that the operating group is knowledgeable about the plant and procedures and fully prepared to operate the facility in a safe manner
- bring the plant to rated capacity and sustained power operation

14.2.1.1 Construction Test Objectives

SSAR Section 14.2.1.1 states that construction tests will be performed to demonstrate that components and systems are correctly installed and operational. These tests will include, but will not be limited to; flushing and cleaning, hydrostatic testing, initial calibration of instrumentation, checks of electrical wiring and equipment, valve testing, and initial energization and operation of equipment and systems. Completion of this phase will ensure that systems are ready for preoperational testing.

14.2.1.2 Preoperational Test Objectives

SSAR Section 14.2.1.2 states that preoperational tests will be conducted before fuel loading to verify that plant systems are capable of operating in a safe and efficient manner compatible with the system design bases. The general objectives of the preoperational test phase will be to:

- ensure that design specification and test acceptance criteria are met
- provide documentation of the performance and safety of equipment and systems
- provide baseline test and operating data on equipment and systems for future reference
- run-in new equipment for a sufficient period so that any design, manufacturing, or installation defects can be detected and corrected
- ensure that plant systems operate together on an integrated basis to the extent possible
- give maximum opportunity to the permanent plant operating staff to obtain practical experience in the operation and maintenance of equipment and systems
- help demonstrate safe and efficient system operating and surveillance testing procedures to the extent possible
- demonstrate that safety systems and equipment are operated to allow fuel loading and entry to the startup phase

14.2.1.3 Startup Test Objectives

SSAR Section 14.2.1.3 states that after the preoperational test phase has been completed, the startup phase will begin with fuel loading and extend to commercial operation. The tests conducted during the startup phase consist of major and minor plant transients, steady-state tests, and process control system tests. These tests will be directed toward demonstrating correct performance of the nuclear boiler and the various plant systems while at power and the validation of analytical models used in the design.

The general objectives of the startup phase will be to:

- achieve an orderly and safe initial core loading
- accomplish all testing and measurements necessary to ensure that the approach to initial criticality and subsequent power ascension is safe and orderly
- conduct low-power physics tests sufficient to ensure that test acceptance criteria have been met
- conduct initial heatup and hot functional testing so that hot integrated operation of all systems is shown to meet test acceptance criteria
- conduct an orderly and safe power ascension program, with requisite physics and systems testing, to ensure that integrated plant operation at power meets test acceptance criteria
- demonstrate, to the extent possible, the adequacy of various component, system, and plant procedures
- conduct a successful warranty demonstration

14.2.1.4 Conclusion

The staff finds that the objectives for the ITP meet the acceptance criteria in SRP Section 14.2 and are acceptable.

14.2.2 Organization and Staffing

14.2.2.1 Normal Plant Staff

SSAR Section 14.2.2.1 states that the normal plant staff responsibilities, authorities, and qualifications are given in SSAR Chapter 13. DFSER Confirmatory Item 14.2.2.1-1 addressed an issue that plant organizational staff was outside the scope of the ABWR SSAR. GE added information in Amendment 30 of SSAR Section 14.2.2.1 to state that plant organizational staff will be provided by the COL applicant. The staff finds this acceptable. This resolved DFSER Confirmatory Item 14.2.2.1-1.

14.2.2.2 Startup Group

SSAR Section 14.2.2.2 states that the startup group will be an ad hoc organization created to ensure that the ITP is conducted in an efficient, safe, and timely manner. The startup group will be responsible for planning, executing, and documenting all startup and testing activities that occur between the completion of the construction phase and commencement of commercial operation of the plant. Upon completion of the startup program, the startup group will be dissolved and the normal plant staff will assume complete responsibility for the plant. The normal plant staff will be included in as many aspects of the test programs as is practicable, considering their normal duties in the operation and maintenance of the plant.

14.2.2.3 General Electric Company

SSAR Section 14.2.2.3 states that GE will be the supplier of the boiling water reactor (BWR) nuclear steam supply system and is responsible for generic and specific BWR designs. During the construction and testing phases of the plant cycle, GE personnel will be on site to offer consultation and technical direction with regard to GE-supplied systems and equipment.

14.2.2.4 Others

SSAR Section 14.2.2.4 states that other concerned parties outside the plant staff organization--such as the architect-engineer, the constructor, the turbine-generator supplier, and vendors of other equipment--will be involved in the testing program to various degrees.

14.2.2.5 Interrelationships and Interfaces

SSAR Section 14.2.2.5 states that the effective coordination between the various site organizations involved in the test program will be achieved through the startup coordinating group (SCG) that will be composed of representatives of the plant owner/operator, GE, and others. The duties of the SCG will be to review and approve project testing schedules and to effect timely changes to construction or testing in order to facilitate execution of the preoperational and initial startup test programs.

14.2.2.6 Conclusion

The staff finds that the organization and staffing plan meet the acceptance criteria in SRP Section 14.2 and are acceptable. As discussed in Section 14.2.2.1, the normal plant staff aspects will be evaluated during the COL review.

14.2.3 Test Procedures

SSAR Section 14.2.3 states that testing during all phases of the ITP will be conducted using detailed, step-by-step written procedures to control the conduct of each test.

Such test procedures will specify testing prerequisites, describe desired initial conditions, include appropriate methods to direct and control test performance (including the sequencing of testing), specify acceptance criteria by which the test will be evaluated, and provide for or specify the format by which data or observations will be recorded. The procedures will be developed, reviewed, and controlled by personnel with appropriate technical backgrounds and experience in accordance with the startup administrative manual. This will include participation of principal design organizations in the establishment of test performance requirements and acceptance criteria. GE will provide the COL applicant with scoping documents (i.e., preoperational and startup test specifications) containing testing objectives and acceptance criteria applicable to its scope of design responsibility. Such documents also will include, as appropriate, delineation of specific plant operational conditions at which tests will be conducted, testing methodologies to be used, specific data to be collected, and acceptable data reduction techniques as well as any reconciliation methods needed to account for test conditions, methods, or results if testing is performed at conditions other than representative design operating conditions. Available information on operating and testing experiences of operating power reactors will be factored into test procedures as appropriate.

The staff finds that the content, development, and review of test procedures meet the criteria in SRP Section 14.2, and are acceptable. However, a COL applicant will need to provide the following for staff review:

- The scoping document (i.e., preoperational and startup test specifications) containing testing objectives and acceptance criteria applicable to its scope of design responsibility. This was DFSER COL Action Item 14.2.3-1.

- The scoping document that delineate and any other documents which delineate plant operational conditions at which tests are to be conducted, testing methodologies to be utilized, specific data to be collected, and acceptable data reduction techniques to be reviewed by the staff at the time of combined operating license. This was DFSER COL Action Item 14.2.3-2.

- The scoping document that delineate any reconciliation methods needed to account for test conditions, methods or results if testing is performed at conditions other than representative design operating conditions. This was DFSER COL Action Item 14.2.3-3.

- The approved preoperational test procedures approximately 60 days before their intended use and startup test procedures approximately 60 days before fuel loading. This was DFSER COL Action Item 14.2.3-4.

GE incorporated the above COL action items in SSAR Sections 14.2.3 and 14.2.13. This is acceptable.

14.2.4 Conduct of Test Program

SSAR Section 14.2.4 states that the ITP will be conducted by the startup group in accordance with the startup administrative manual. This manual will contain the administrative procedures and requirements that govern the activities of the startup group and their interfaces with other organizations. The startup administrative manual will receive the same level of review and approval as do other plant administrative procedures. It will define the specific format and content of preoperational and startup test procedures as well as the review and approval process for both initial procedures and subsequent revisions or changes. The startup manual also will specify the process for review and approval of test results and for resolution of failures to meet acceptance criteria and of other operational problems or design deficiencies. It will describe the various phases of the ITP and establish the requirements for progressing from one phase to the next as well as those for moving beyond selected hold points or milestones within a given phase. It will also describe the controls in place that will ensure the as-tested status of each system is known and track modifications, including retest requirements, deemed necessary for systems undergoing or already having completed specified testing. Additionally, the startup manual will delineate the qualifications and responsibilities of the different positions within the startup group.

<u>Staff Evaluation of DSER Items</u>

In the DSER, the staff determined that GE should specify whose approval must be obtained before increasing power to the next higher test plateau. (This was incorrectly identified as an interface requirement in Section 14.2.5 of the DSER). GE indicated that such specifics will be a function of the plant owner/operator's unique organizational structure and detailed plant administrative procedures and thus left to the COL applicant. GE revised SSAR Section 14.2.13 to reflect that the COL applicant should specify whose approval is needed to proceed to the next testing plateau. This is acceptable.

The staff finds that the conduct of the test program meets the acceptance criteria in SRP Section 14.2 and is acceptable. However, a COL applicant will need to provide a startup administrative manual (procedures) and any other documents that delineate the conduct of the test program to be reviewed by the staff at the time of the COL application. This was DFSER COL Action Item 14.2.4-1. GE included the above action item in SSAR Section 14.2.13 and the staff finds this acceptable.

14.2.5 Review, Evaluation, and Approval of Test Results

SSAR Section 14.2.5 states that individual test results will be evaluated and reviewed by cognizant members of the startup group. Test exceptions or acceptance criteria discrepancies will be communicated to the affected and responsible organizations who will help resolve the issues by suggesting corrective actions, design modifications, and retests. GE and others outside the plant staff organization, as appropriate, will have the opportunity to review the results for conformance to predictions and exceptions. Test results, including final resolutions, then will be reviewed and approved by designated startup group supervisory personnel.

Staff Evaluation of DSER Items

In the DSER, the staff determined that GE should specify whose approval must be obtained before increasing power to the next higher test plateau. GE indicated that such specifics will be a function of the plant owner/operator's unique organizational structure and detailed plant administrative procedures and are thus will be defined by the COL applicant. GE revised SSAR Section 14.2.13 to reflect that the COL applicant should specify whose approval is needed to proceed to the next testing plateau. This is acceptable.

The staff finds that the process for review, evaluation, and approval of test results meets the acceptance criteria described in SRP Section 14.2 and is acceptable. However, a COL applicant will need to provide a startup administrative manual (procedures) and any other documents that delineate the review, evaluation, and approval of test results for staff review. This was DFSER COL Action Item 14.2.5-1. GE included the above COL action item in SSAR Section 14.2.13 and the staff finds this acceptable.

14.2.6 Test Records

SSAR Section 14.2.6 states that the ITP results will be compiled and maintained according to the startup manual, plant administrative procedures, and applicable regulatory requirements. Test records that demonstrate the adequacy of safety-related components, systems, and structures will be retained for the life of the plant. Retention periods for other test records will be based on consideration of their usefulness in documenting initial plant performance characteristics. As discussed in SER Section 14.2.4, the startup administrative manual is identified as a COL action item.

The staff finds that the test records compilation, maintenance, and retention program meets the acceptance criteria in SRP Section 14.2, and is acceptable.

14.2.7 Conformance of Test Program With Regulatory Guides

SSAR Section 14.2.7 lists the NRC RGs that will be used in the development of the ITP and states that the applicable tests will comply with these guides. The applicable revisions to these RGs are listed in SSAR Table 1.8-20.

In the DSER, the staff determined that GE needed to add additional RG references to Table 1.8-20 and SSAR Section 14.2.7. This was DSER Open Item 6.C. The DFSER tracked this issue as DFSER Confirmatory Item 14.2.7-1.

In its letter of May 20, 1991, GE agreed to amend the SSAR to include RGs 1.95, "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release," Revision 1, and 1.139, "Guidance for Residual Heat Removal," Revision 0, and to document the applicable revision number of each RG listed in SSAR Section 14.2.7 or to amend the section to reference Table 1.8-20 for the applicable revision numbers of the listed guide. GE made these changes in Amendment 18 of the SSAR. GE further agreed to correct the reference to RG 1.68.3, "Preoperational Testing of Instrument and Control Air," Revision 0, contained in SSAR Table 1.8-20 or Section 14.2.7, as appropriate, to Revision 0, issue date of April 1982. GE stated this change in its response on March 11, 1992. The staff determined that GE made the required changes in Amendment 21 of the SSAR. The staff finds this acceptable. This resolved the above portions of DFSER Confirmatory Item 14.2.7-1.

The staff's review identified that SSAR Section 14.2.7 lists RG 1.140 as applicable to the ABWR, however, SSAR Table 1.8-20 indicates RG 1.140 is not applicable to the ABWR. SSAR Section 14.2.12.1.34(3)(1) includes RG 1.140 guidance as acceptance criteria for visual inspection and airflow distribution, testing for penetration of dioctyl phthalate and bypass leakage testing for high-efficiency particulate air (HEPA) and charcoal absorber sections where installed. The staff found that the utilization of RG 1.140 guidance for normal ventilation exhaust system air filtration and absorber units was not clear. GE was asked to make the appropriate clarifications to Table 1.8-20, Section 14.2.7, and individual test abstracts, as appropriate, to clearly and consistently describe the degree of conformance to RG 1.140, any exceptions taken, and to identify any aspects of RG 1.140 that are not applicable to the ABWR.

GE responded by deleting the reference to RG 1.140 in SSAR Amendment 30 from the list of acceptance criteria in SSAR Section 14.2.12.1.34 and from the list of RGs in SSAR Section 14.2.7 to make those sections consistent with Table 1.8-20. GE added RG 1.52 to Section 14.2.12.1.34 and Section 14.2.7 in SSAR Amendment 30 to address testing of engineered safety feature (ESF) heating, ventilation, and air conditioning (HVAC) systems that require in place testing of HEPA and carbon adsorber filters. The staff finds this acceptable. This resolved DFSER Confirmatory Item 14.2.7-1.

14.2.8 Utilization of Reactor Operating and Testing Experience in the Development of Test Program

SSAR Section 14.2.8 states that since every reactor/plant in a GE BWR product line is an evolutionary development of the previous plant in the product line it is evident that the ABWR plants will benefit from the experience acquired with the successful and safe startup of more than 30 previous BWR plants. The operational experience and knowledge gained from these plants and other reactor types have been factored into the design and test specifications of GE-supplied systems and equipment that will be demonstrated during the preoperational and startup test programs. Additionally, reactor operating and testing experience of similar nuclear power plants obtained from NRC licensee event reports and through other industry sources will be used to the extent practicable in developing and carrying out the ITP.

The staff finds that the use of reactor operating and testing experience in the development of the test program meets the acceptance criteria in SRP Section 14.2, and is acceptable.

14.2.9 Trial Use of Plant Operating and Emergency Procedures

SSAR Section 14.2.9 states that, to the extent practicable, throughout the preoperational and initial startup test program, operating, emergency, and abnormal procedures will be incorporated, where applicable, in the performance of tests. The use of these procedures is intended to

- prove the adequacy of the specific procedure or illustrate changes that may be required
- provide training of plant personnel in the use of these procedures
- increase the level of knowledge of plant personnel on the systems being tested

GE further indicated, that to meet the above goals, test procedures will actually reference or extract steps from operating, emergency, or abnormal procedures.

The staff finds that the method of incorporating operating, emergency, or abnormal procedures into the test program meets the acceptance criteria in SRP Section 14.2, and is acceptable. However, as discussed in SSAR Section 14.2.3, a COL applicant will need to provide the approved preoperational test procedures 60 days before their intended use and the startup test procedures approximately 60 days before fuel loading.

14.2.10 Initial Fuel Loading and Initial Criticality

SSAR Section 14.2.10 states that the fuel loading and initial criticality will be conducted in a very controlled manner in accordance with specific written procedures as part of the startup test phase. The NRC approves fuel loading after it has verified that prerequisite testing has been satisfactorily completed or after the COL applicant provides appropriate justification to proceed with fuel loading and complete the preoperational testing after fuel loading.

The intent of the testing program is to complete all preoperational tests and approve the results before commencement of fuel loading. However, there may be unforeseen circumstances that arise that would prevent this from occurring, but that would not necessarily justify the delay of fuel loading.

In the DFSER, the staff determined that SSAR Section 14.2.10 should address the completion of preoperational testing (including the review and approval of test results required before fuel loading). If portions of any preoperational tests are intended to be conducted, or their results approved, after fuel loading, GE should (1) list each test, (2) state which portions of each test will be delayed until after fuel loading, (3) provide technical justification for delaying these portions, and (4) state when each test will be completed and the results approved.

GE revised SSAR Section 14.2.10 in Amendment 18 to require that the above stated conditions be appropriately documented should the COL applicant decide to request permission from the NRC to proceed with fuel loading under such circumstances. In the DFSER, the staff found that the methods of controlling pre-fuel load checks, initial fuel loading, precritical testing and initial criticality outlined in Section 14.2.10 met the acceptance criteria in SRP Section 14.2, and were acceptable. However, a referencing applicant will need to provide a startup administrative manual (procedures) and any other documents that delineate the methods of controlling pre-fuel load checks, initial fuel loading, precritical testing and initial criticality. These are to be reviewed by the staff before issuing a COL. This was DFSER COL Action Item 14.2.10.4-1. GE included this COL action item in SSAR Sections 14.2.10 and 14.2.13. The staff finds this acceptable.

14.2.10.1 Pre-Fuel Load Checks

SSAR Section 14.2.10.1 states that once the plant has been declared ready to load fuel, there are a number of specific checks that must be made before proceeding. These include a final review of the preoperational test results and the status of any design changes, work packages, and/or retests that were initiated as a result of exceptions noted during this phase. Also, the technical specifications (TS) surveillance program requirements, as described in the ABWR TS Chapter 16, will be instituted at this time to assure the operability of systems required for fuel loading. Just before the initiation of fuel loading, the proper vessel water level and chemistry will be verified and the calibration and response of the nuclear instruments will be checked. This was DFSER TS Item 14.2.10.1-1.

The staff reviewed GE submittals dated March 31 and April 16, 1993, which contained GE's response to resolve TS Item 14.2.10.1-1. SSAR Section 14.2.12.2.3(2), prerequisites (c) and (d) will require neutron detectors instrument channels to be properly calibrated and operable prior to fuel loading. A cross-reference to a response dated March 31, 1993, of TS LCO 3.3.1.3, Surveillance Requirements (SR) for startup range nuclear monitor instrumentation, excluded calibration of neutron detectors. The staff determined that calibration of neutron detector instrumentation is required before fuel loading.

The ABWR Standardized Technical Specification (STS) bases section for SR 3.3.2.1.6 states "the neutron detectors are excluded from channel calibration because they cannot readily be adjusted. The detectors are fission chambers that are designed to have relatively constant sensitivity over the range, and with an accuracy specified for a fixed useful life." The staff determined that the startup test program prerequisite requirements verify calibration of the neutron detectors instrumentation channels before fuel loading and before startup. Neutron sources would be used to verify proper calibration and instrument response of the detectors. The ABWR STS do not require the detectors to be calibrated during startup, therefore, the staff determined that the cross reference to the STS in SSAR Section 14.2.12.2-3 should be deleted. The staff verified that the cross reference to the STS was deleted from acceptance criteria 14.2.12.2-3(2)(c) and (d) in SSAR Amendment 31. The acceptability of the ABWR TS is discussed in Chapter 16 of this report. This resolved DFSER TS Item 14.2.10.1-1.

14.2.10.2 Initial Fuel Loading

SSAR Section 14.2.10.2 states that fuel loading will require the movement of the full core complement of assemblies from the fuel pool to the core, with each assembly being identified by number before being placed in the correct coordinate position. The procedure controlling this movement will specify that partial core shutdown margin and subcritical checks be made at predetermined intervals throughout the loading, thus ensuring safe loading increments as described in test abstract 14.2.12.2.3, Fuel Loading. In-vessel neutron monitors will provide continuous indication of the core flux level as each assembly is added. A complete check will be made of the fully loaded core to ascertain that all assemblies are properly installed, correctly oriented, and occupying their designated positions.

14.2.10.3 Pre-Criticality Testing

The control rods shall be verified functional and scram tested with the fuel in place. The post-fuel-load flow test of the reactor internals vibration assessment program, if applicable, shall be conducted at this time. Additionally, a final verification shall be made that the required TS surveillances have been performed.

14.2.10.4 Initial Criticality

SSAR Section 14.2.10.4 states that during initial criticality, the full core shutdown margin shall be verified for the fully loaded core as described in startup test abstract 14.2.12.2.4, "Full Core Shutdown Margin Demonstration." SSAR Section 14.2.10.4 also states that initial criticality shall be achieved in an orderly, controlled fashion following specific detailed procedures in an approved rod withdrawal sequence. Core neutron flux shall be continuously monitored during the approach to criticality and periodically compared to predictions to allow early detection and evaluation of potential anomalies.

14.2.11 Test Program Schedule

SSAR Section 14.2.11 states that the schedule, relative to the initial fuel load date, for conducting each major phase of the ITP will be provided by the COL applicant. This includes the time table for generation, review, and approval of procedures as well as the actual testing and analysis of results. As a minimum, at least 9 months before the fuel loading date should be allowed for conducting the preoperational phase and at least 3 months should be allowed for conducting the startup and power ascension testing that commences with fuel loading. To allow for NRC review, test procedure preparation for power ascension will be scheduled so that approved procedures are available approximately 60 days prior to fuel load. Although there will be considerable flexibility available in the sequencing of testing within a given phase, testing should be performed as systems are turned over from construction. However, the interdependency of systems should also be considered so that common support systems, such as electrical power distribution, service and instrument air, and the various makeup water and cooling water systems, will be tested as early as possible. Sequencing of testing during the startup phase will depend primarily on specified power and flow conditions and inter-system prerequisites. To the extent practicable, the schedule should establish that, before exceeding 25-percent power, the test requirements will be met for those plant structures, systems, and components (SSCs) that are relied on to prevent, limit, or mitigate the consequences of postulated accidents. Additionally, testing will be sequenced so that the safety of the plant is never totally dependent on untested systems, components, or features.

Staff Evaluation of DSER Items

In the DSER, the staff determined that SSAR Section 14.2.11 should be modified to include the following:

- A figure that illustrates the power-flow operating map
- A table that lists the startup tests and states at which test condition(s) each test is to be conducted

The staff stated in DSER Open Items 116 and 117 that SSAR Figure 4.4-1 did not provide sufficient detail regarding test condition identification to determine that each startup test will be conducted at appropriate power-flow conditions in accordance with RG 1.68, "Initial Test Programs for Water Cooled Nuclear Power Plant" (Rev. 2), Appendix A.5, "Power Ascension Tests." In addition, the SSAR did not contain a table of startup tests.

GE subsequently submitted a power-to-flow operating map, SSAR Figure 14.2-1, that provides an appropriate indication of test conditions and GE provided a table of startup tests, SSAR Table 14.2-1, "Startup Test Matrix." This was acceptable subject to incorporation into a future SSAR revision and staff review. This resolved DSER Open Items 116 and 117, and became DFSER Confirmatory Item 14.2.11-1. The staff finds that the power-to-flow map and the table of startup tests that were incorporated into Amendment 23 of the SSAR meet the acceptance criteria of SRP Section 14.2, and are acceptable. This resolved DFSER Confirmatory Item 14.2.11-1.

GE indicated that SSAR Section 14.2.11 would be revised to include the following additional information on the test program schedule. Power ascension testing will be conducted in essentially three phases: (1) initial fuel loading and open vessel testing, (2) testing during nuclear heat up to rated temperature and pressure, and (3) power operation testing from 5-percent to 100-percent rated power. Further, power operation testing will be divided into three sequential testing plateaus as shown on SSAR Figure 14.2-1. The testing plateaus consist of low-power testing at less than 25-percent power, mid-power testing up to 75-percent power between approximately the 50-percent and 75-percent rod lines, and high-power testing along the 100-percent rod line up to rated power. Thus, there will be a total of five different testing plateaus designated as shown on SSAR Figure 14.2-1. Table 14.2-1 indicates in which testing plateaus the various power ascension tests will be performed. Although the order of testing within a given plateau will be somewhat flexible, the normal recommended sequence of tests will be; (1) core performance analysis, (2) steady state tests, (3) control system tuning, (4) system transient tests, and (5) major plant transients (including trips). Also, for a given testing plateau, testing at lower power levels generally should be performed before that at higher power levels. The detailed testing schedule will be generated by the COL applicant and will be made available to the NRC before actual implementation. The schedule then will be maintained to reflect actual progress and subsequent revised projections. The information above was acceptable subject to incorporation into a future SSAR revision. This was DFSER Confirmatory Item 14.2.11-2.

GE revised the test program schedule in SSAR Section 14.2.11 and Table 14.2-1 in SSAR Amendments 23 and 33. The staff finds the changes acceptable. This resolved DFSER Confirmatory Item 14.2.11-2.

Additionally, a COL applicant will need to provide a startup administrative manual (procedures) and any other documents that delineate the test program schedule for staff review. GE identified this as a COL action item in a response dated April 6, 1993. This information was incorporated into SSAR Amendment 31. This was DFSER COL Action Item 14.2.11-1. GE included this COL action item in SSAR Section 14.2.3, and the staff finds this acceptable.

14.2.12 Individual Test Descriptions

14.2.12.1 Preoperational Test Procedures

SSAR Section 14.2.12.1 states that the general descriptions relate the objectives of each preoperational test. During the final construction phase, it may be necessary to modify the preoperational test methods as operating and preoperational test procedures are developed. Consequently, methods in the descriptions are general, not specific.

Specific testing to be performed and the applicable acceptance criteria for each preoperational test will be documented in detailed test procedures to be made available to the NRC approximately 60 days before their intended use. Preoperational testing will be in accordance with the detailed system specifications and associated equipment specifications for equipment in those systems (provided as part of scoping documents to be supplied by GE and others as described in SSAR Subsection 14.2.3). The tests will demonstrate that the installed equipment and systems will perform within the limits of these specifications. To allow for verification that the detailed test procedures are developed in accordance with established methods and appropriate acceptance criteria, the plant and system preoperational test specifications will also be made available to the NRC.

The preoperational tests anticipated for the ABWR standard plant design are listed and described in the SSAR Subsection 14.2.12.1. The staff finds that the scope of preoperational tests described in SSAR Subsection 14.2.12.1 meets the acceptance criteria of SRP Section 14.2, and is acceptable. Testing of systems out of the scope of the SSAR are discussed in SSAR Section 14.2.13 along with other COL information related to the ITP.

14.2.12.2 General Discussion of Startup Tests

SSAR Section 14.2.12.2 discusses those tests proposed and expected to comprise the startup test phase. For each test a general description is provided for test purpose, test prerequisites, test description, and test acceptance criteria, where applicable. Because changes will occur as the test program is developed and implemented, the descriptions remain general in scope. However, an attempt is made in describing a test to identify those operating and safety-oriented characteristics of the plant design that are being explored and evaluated.

Where applicable, the relevant acceptance criteria for the test are discussed. Some of the criteria relate to the value of process variables assigned in the design or analysis of the plant, component systems, and associated equipment. If a criterion of this nature is not satisfied, the plant will be placed in a suitable hold condition until resolution is obtained. Tests compatible with this hold condition may be continued. Following resolution, applicable tests or portions of these tests may be repeated to verify that the requirements of the criterion are ultimately satisfied. Other criteria may be associated with expectations relating to the performance of systems. If this type of criterion is not satisfied, operating and testing plans would not necessarily be altered. However, investigations of the measurements and of the analytical techniques used for the predictions would be started. Specific actions for dealing with criteria failures and other testing exceptions or anomalies will be described in the startup administrative manual.

The specifics of the startup tests relating to test methodology, plant prerequisites, initial conditions, acceptance criteria, and analysis techniques will come from the appropriate design and engineering organizations of the COL applicant in the form of plant, system, and component performance and testing specifications. The COL applicant shall provide test documents for the Office of Nuclear Reactor Regulation review as discussed in SSAR Section 14.2.3.

14.2.12.3 Staff Evaluation of DSER Items

In the DSER, the staff determined that testing of systems outside the scope of the ABWR standard plant design should be included or referenced in SSAR Section 14.2.12. (This was not tracked as a separate open item.)

GE revised Section 14.2.12.1 in SSAR Amendment 18 to state that testing of systems outside the scope of the ABWR standard plant design are discussed in SSAR Section 14.2.13. The staff finds this acceptable.

In the DSER, the staff determined that SSAR Section 14.2.12 test abstracts should be modified to address the following concerns:

(1) Several preoperational and startup test prerequisites include the requirement that interfacing support systems will be available.

 The staff asked GE to identify which support systems will be required for each test and to specify which individuals or groups will be authorized to make this determination. (This was incorrectly identified as an interface requirement.)

 GE stated that the interfacing support system requirements will be specified in the detailed test procedures (and operating and maintenance procedures, if appropriate) that are required by RG 1.68 to be made available to NRC personnel at least 60 days before the intended use of preoperational tests procedures and 60 days before fuel loading for startup test procedures. Additionally, the startup administrative manual will delineate how such determinations of operability and availability will be authorized. Thus, these details are the responsibility of the COL applicant. This was DFSER COL Action Item 14.2.12.3-1. GE included this action item in Section 14.2.13, and the staff finds this acceptable.

In the DSER, the staff determined that the level of detail in the test abstracts was insufficient to determine conformance with RG 1.68, Position C.2. This was DSER Open Item 118. The individual test abstracts in SSAR Sections 14.2.12.1.1, 14.2.12.1.4, 14.2.12.1.7, 14.2.1.11, 14.2.12.1.12, 14.2.12.1.13, 14.2.12.1.18, 14.2.12.1-21, 14.2.12.1.22, 14.2.12.1.43, 14.2.12.1.44, 14.2.12.1.45.1, 14.2.12.1.45.2, 14.2.12.1.45.3, 14.2.12.1.45.4, 14.2.12.1.53, 14.2.12.1.59, 14.2.12.1.67, 14.2.12.1.68, and 14.2.12.1.69 did not specify basic systems required to be available, interface systems, or criteria required as prerequisite or initial conditions for the preoperational tests. Specific prerequisites should be addressed in these individual test abstracts. This was tracked in the DFSER as Open Item 14.2.12.3-1.

The staff completed its review of GE's February 12, 1993, submittal, which addresses DFSER Open Item 14.2.12.3-1. The staff determined that the level of detail for specific preoperational test prerequisites was sufficient with the exception of two individual test abstracts. GE provided revisions to the prerequisite sections of preoperational test abstracts 14.2.12.1.4 and 14.2.12.1.44 in responses dated May 13 and 21, 1993. The staff finds both of these revisions acceptable. The revisions were incorporated into SSAR Amendment 30. This resolved DFSER Open Item 14.2.12.3-1.

(2) The use of the word "should" in most, if not all test abstracts, is not a commitment by the COL applicant to perform certain tasks. It should, therefore, be reevaluated and revised accordingly (i.e., "will," "must"). This was DSER Open Item 119.

In the DFSER, the staff verified that GE incorporated the word change from "should" to either "will" or "shall" into most test abstracts. This was acceptable subject to incorporation into a future SSAR revision. This resolved DSER Open Item 119, and was tracked in the DFSER as DFSER Confirmatory Item 14.2.12.3-1. The above word change was incorporated into the applicable test abstracts in SSAR Amendment 23. The staff finds this acceptable. This resolved DFSER Confirmatory Item 14.2.12.3-1.

(3) Several preoperational and startup test abstracts included imprecise acceptance criteria (e.g., "applicable intervals," "applicable design specifications," "specified amounts," "specified tolerances," "perform as specified," and "function properly"). This was DSER Open Item 120.

In the DFSER, the staff determined that GE should address in the individual test abstracts the bases for determining acceptable system and component performance. Acceptable criteria includes specific references to RGs, TS, assumptions used in the safety analysis, other ABWR SSAR sections, and applicable codes and standards.

GE indicated that SSAR Chapter 14 was written primarily to document the appropriate testing commitments contained in RG 1.68. In its May 20, 1991, submittal, GE indicated that precise acceptance criteria would be provided as part of the inspections, tests, analyses, and acceptance criteria (ITAAC) effort. However, GE did not provide this information in either the ITAAC or in SSAR Chapter 14. This was tracked in the DFSER as DFSER Open Item 14.2.12.3-2.

The staff reviewed the individual test abstracts submitted on February 12 and May 13, 1993, to assess the adequacy of the test abstract coverage of system-specific test requirements. Clarifications and additional cross- reference information for roughly half of the preoperational tests were determined necessary. The staff requested GE to supplement the SSAR with this additional information in order to determine that these tests adequately address system-specific test requirements.

GE submitted Amendment 30, Revision 0, dated June 7, 1993, to the staff which provided cross-reference acceptance criteria information for all but two of the preoperational test abstracts. The two tests were the integrated emergency core cooling systems (ECCS) loss of offsite power (LOOP)/loss of coolant accident (LOCA) preoperational test and the extraction steam system preoperational test. GE provided the necessary cross references for these test abstracts in Revision 1 of SSAR Amendment 30. The staff found all of the cross-references acceptable. Additional information on system-specific testing requirements were also provided for all test abstracts. The staff finds this information acceptable. This resolved DFSER Open Item 14.2.12.3-2.

(4) SSAR Section 14.2.12.2 states that failure to satisfy some acceptance criteria (e.g., those related to values of process variables important to plant design) will result in the plant being placed in a suitable hold position until resolution is obtained, while failure to satisfy other acceptance criteria (e.g., expectations relating to system performance) may only result in the need for further data analysis. This was DSER Open Item 121.

In the DFSER, the staff determined that the distinction between these types of acceptance criteria was unclear, and that GE should clearly address the various types of acceptance criteria and the resultant actions for each type if unsatisfactory test results were obtained.

SSAR Section 14.2.12.2, as modified by GE, stated: "Specific actions for dealing with criteria failures and other testing exceptions or anomalies will be described in the startup administrative manual." This response to the staff's open item was not acceptable. Of 35 individual startup test abstracts in SSAR Section 14.2.12.2, 33 did not specify the required actions or precautions for dealing with criteria failures and other testing exceptions or anomalies. GE did not adequately modify Section 14.2.12.2 or the individual test abstracts to address the subject acceptance criteria or actions. This was DFSER Open Item 14.2.12.3-3.

The staff requested GE provide a general distinction between startup test level 1 and level 2 acceptance criteria, and to specify individual startup test abstract acceptance criteria as level 1 or level 2. Level 1 criteria relates to the value of the process variables assigned in the design or analysis of the plant, component systems, or associated equipment. If a level 1 criteria is not satisfied, the plant will be placed in a hold condition until resolution is obtained. A level 2 criterion is associated with expectations relating to the performance of systems. If a level 2 criteria requirement is not satisfied then an engineering evaluation must be completed to verify that overall system performance is acceptable.

The staff reviewed a submittal by GE on April 16, 1993, of SSAR Section 14.2.12.2 which identified the distinction between startup test level 1 and level 2 acceptance criteria. The distinction between levels 1 and 2 acceptance criteria is found to be acceptable. This information was incorporated into SSAR Amendment 30.

The staff's review of individual startup test abstracts, submitted by GE on June 7, 1993, identified that more precise acceptance criteria has been provided and the criteria is acceptable. The staff also determined that the system-specific test requirements are adequately addressed. GE incorporated the individual startup test abstract acceptance criteria information, the startup test levels 1 and 2 acceptance criteria, and the system-specific startup test requirements into SSAR Amendment 30. The staff finds the startup testing abstracts as described in SSAR Amendment 30 acceptable. This resolved DFSER Open Item 14.2.12.3-3.

(5) GE should identify startup tests listed in SSAR Section 14.2.12.2 that are not essential to the demonstration of conformance with design requirements for SSCs and design features that

- will be used for safe shutdown and cooldown of the reactor under normal plant conditions and for maintaining the reactor in a safe condition for an extended shutdown period
- will be used for safe shutdown and cooldown of the reactor under transient (infrequent or moderately frequent events) conditions and postulated accident conditions and for maintaining the reactor in a safe condition for an extended shutdown period following such conditions
- will be used for establishing conformance with safety limits or limiting conditions for operation that will be included in the facility TS
- are classified as engineered safety features (ESFs) or will be used to support or ensure the operations of ESFs within design limits

- are assumed to function or for which credit is taken in the accident analysis for the facility, as described in the SSAR

- will be used to process, store, control, or limit the release of radioactive materials

GE stated that the test abstracts contained in SSAR Section 14.2.12.2 are intended to meet the requirements of RG 1.68, updated and/or modified as necessary to reflect the actual ABWR design. In the DSER it was stated that a required screening will be performed by GE to identify and document any testing that is currently specified for systems that are not essential for demonstrating conformance with the aforementioned criteria. This was DSER Open Item 122.

GE revised SSAR Section 14.2.13 (in draft) to state that criteria contained in the RG 1.68, Position C.1, will be used to determine if any testing is currently specified for systems that are not essential for demonstrating conformance with the aforementioned criteria. The staff found this item acceptable subject to incorporation into a future SSAR revision. This resolved DSER Open Item 122 and was tracked in the DFSER as DFSER Confirmatory Item 14.2.12.3-2.

The testing described in Section 14.2.12.2 includes testing of a limited number of ABWR SSCs, and design features that do not meet the referenced RG 1.68 criteria. The staff subsequently determined that it was premature to develop a specific list of tests that do not meet the criteria in RG 1.68, Position C.1. The staff has not yet defined the applicable COL license conditions related to such tests (i.e., timely notification to the NRC for major test changes that affect systems that meet the criteria in RG 1.68, Position C.1). This aspect will be further reviewed for a COL applicant as part of the COL action items listed in SSAR Section 14.2.13. Therefore, GE did not include this information in the SSAR. The staff finds this acceptable. This resolved DFSER Confirmatory Item 14.2.12.3-2.

14.2.12.4 Conformance of the ABWR with RG 1.68, Revision 2

In the DSER, the staff's review of the preoperational and startup test phase descriptions disclosed that the operability of several of the systems and components listed in RG 1.68 may not be adequately demonstrated by the tests described in the SSAR. This was DSER Open Item 123. Each item is evaluated separately below. This evaluation resolved DSER Open Item 123.

<u>Staff Evaluation of DSER Items</u>

The staff determined that GE should either expand the test descriptions to address the following items, insert cross-references in SSAR Section 14.2.12 if complete test descriptions for the following items are provided elsewhere in the SSAR, or modify SSAR Section 14.2.7 or Table 1.8-20 of the SSAR, as appropriate, to provide technical justification for any exception to RG 1.68. It was stated in the DSER that the following items should be reflected in a subsequent amendment to the SSAR. (Note: Each item is numbered in accordance with RG 1.68.)

- <u>1.a.(2)(d) Supports and restraints for discharge piping of safety relief valves (SRVs)</u>

GE stated in its response of May 20, 1991, that a statement had been added to SSAR Section 14.2.12.1.1 indicating that testing of SRV discharge piping supports and restraints is specifically covered by that testing described in SSAR Section 14.2.12.1.51.

GE incorporated into SSAR Section 14.2.12.1.51 cross-references to SSAR Sections 3.9.2.1 and 5.4.14.4. The references were acceptable subject to incorporation into a future SSAR revision. This was DFSER Confirmatory Item 14.2.12.4-1.

The staff reviewed these cross-references in SSAR Amendment 23, Chapters 3 and 5, respectively and finds them to be acceptable. This resolved DFSER Confirmatory Item 14.2.12.4-1.

- <u>1.a.(4) Pressure boundary integrity tests</u>

GE stated in its response of May 20, 1991, that integrity tests of the reactor coolant pressure boundary are specified in SSAR Section 5.2.4.6.2.

The staff verified this and the staff also determined that a cross-reference to Section 5.2.4.6.2 was incorporated into Subsection 14.2.12.1.1 in SSAR Amendment 18. The staff evaluated this reference in Chapter 5 of this report, and finds this acceptable.

- <u>1.c Protection of facility for anticipated transients without a scram (ATWS)</u>

GE stated in its response of May 20, 1991, that ATWS protection functions will be tested as part of the respective systems that perform such functions (i.e., standby liquid control system, rod control and information system

(RCIS), fine motion control rod system, recirculation flow control system). For the purpose of more explicitly demonstrating compliance with RG 1.68, GE revised the appropriate parts of SSAR Section 14.2.12.1 (in SSAR Amendment 18) as shown below to more specifically indicate where ATWS-related testing requirements are being fulfilled, particularly those related to the alternate rod insertion function.

- 14.2.12.1.3(3)(a) - Recirculation Flow Control
- 14.2.12.1.6(3)(b) - CRD (Control Rod Drive) System
- 14.2.12.1.7(3)(b) - RCIS

The staff finds this acceptable.

- 1.h.(4) Demonstration that containment hydrogen monitoring is functional without the operation of the hydrogen recombiner

GE stated in its response of May 20, 1991, that in the ABWR design, containment hydrogen monitoring is accomplished separately from the hydrogen recombiners. Therefore, the specific test described in RG 1.68 is not applicable. Proper functioning of containment hydrogen monitors is verified by the testing described in SSAR Section 14.2.12.1.26. The staff finds this acceptable.

- 1.h.(9) Demonstration that containment recirculation fans can operate in accordance with design requirements at the containment design peak accident pressure

GE stated in its response of May 20, 1991, that the ABWR design does not use containment recirculation fans during normal operation or accident conditions. Therefore, the specific test described in RG 1.68 is not applicable. The staff finds this acceptable.

- 1.i.(1) Containment design over pressure structural tests (and vacuum tests)

GE stated in its response of May 20, 1991, that containment structural integrity testing requirements are specified in SSAR Section 3.8.1.7.1. GE also incorporated into Section 14.2.12.1.40.2 in SSAR Amendment 18 a cross-reference to Section 3.8.1.7.1. This issue was evaluated in SER Chapter 3. The staff finds this acceptable.

- 1.j.(12) Failed fuel detection system

GE stated in its response of May 20, 1991, that the ABWR design failed fuel detection function will be performed by the leak detection and isolation system and the process radiation monitoring system. In particular, gross fuel failure would be detected first by the main steamline radiation monitors and then by the offgas pre-treatment radiation monitors. In addition, the normal reactor water sampling system will allow for identification of trends indicative of possible fuel failure. Testing of the applicable features of the associated systems, as described in SSAR Sections 14.2.12.1.13 and 14.2.12.1.23, will ensure proper operation of the failed fuel detection function.

GE revised Section 14.2.12.2.1 in SSAR Amendment 18 to include specific reference to proper operation of failed fuel detection functions. The staff finds this acceptable.

- 1.j.(15) Automatic dispatcher control systems

GE stated in its response of May 20, 1991, that automatic load following will be performed by the automatic power regulator for which testing is described in SSAR Section 14.2.12.1.17. This system will have the capability, if enabled, to accept external demand signals (e.g., from the load dispatcher). If the COL applicant decides to seek approval for using this capability, designation of the appropriate testing will be included in that application.

GE revised SSAR Section 14.2.13 to include automatic dispatcher control systems as a responsibility of the COL applicant. This is acceptable. This was also DFSER COL Action Item 14.2.12.4-1. GE incorporated this action item in SSAR Section 14.2.13, and the staff finds this acceptable.

- 1.k.(2) Personnel monitors and radiation survey instruments

GE stated in its response of May 20, 1991, that traditional preoperational testing of personnel monitors and radiation survey instruments is not appropriate in the ABWR design because these instruments will be subject to very specific calibration programs. It is the responsibility of the plant operator to verify and maintain the proper calibration and operation of such devices. Therefore, GE revised the SSAR to indicate that any required testing of personnel monitors and radiation survey instruments would be a COL responsibility. Section 14.2.13 in SSAR Amendment 18 was modified to include this as a COL License Information. This is acceptable. This was also DFSER COL Action Item 14.2.12.4-2. GE incorporated this action item into SSAR Section 14.2.13, and the staff finds this acceptable.

- 1.n.(14)(f) Control room habitability systems. Demonstrate proper operation of smoke and toxic chemical detection systems and ventilation shutdown devices, including leak tightness of ducts and flow rates, proper direction of air flows, and proper control of space temperatures

GE revised the test description in SSAR Section 14.2.12.1.34 (Amendment 18) to indicate that the control room habitability function is to be included in the testing specified for the dedicated HVAC system of the main control room. Additionally, GE added a specific requirement to demonstrate the system capability to detect smoke and/or toxic chemicals and to remove and/or prevent in-leakage of smoke or chemicals. The staff finds this acceptable.

- 2.c Final functional testing of the reactor protection system to demonstrate proper trip points, logic, and operability of scram breakers and valves. Demonstrate the operability of manual scram functions

GE stated in its response of May 20, 1991, that final functional testing will have been completed as part of the preoperational testing described in SSAR Section 14.2.12.1.14. Additionally, these tests are part of the plant TS surveillance program that will be required to be instituted before fuel loading, as specified in SSAR Section 14.2.10.1.

GE revised Section 14.2.12.2.3 in SSAR Amendment 18 to specifically require that the final functional testing required by RG 1.68, Position 2.c, be completed as prerequisites to fuel loading. This is acceptable. This was also tracked as DFSER TS Item 14.2.12.4-1.

The staff found that GE's response to TS Item 14.2.12.4-1 dated April 16, 1993, stated: "Final functional surveillance testing will demonstrate operability of scram breakers and valves prior to fuel loading. Such demonstrations will be assured as part of the TS program (LCO 3.3.1.1 and 3.3.1.2) which must be instituted prior to entry into mode 5." The staff found GE's response acceptable; however, GE had not submitted a response to the ABWR STS LCO 3.3.1.1 and LCO 3.3.1.2, which incorporates surveillance testing of scram breakers and valves to demonstrate their operability before fuel loading. The staff concluded that STS SR 3.3.1.2.1 or STS Bases Section 3.3.1.2 should be revised to include testing the operability of scram breakers, scram solenoid valves, and backup scram solenoid valves. The staff revised the bases portion of the STS to include a description of scram valves and backup scram valves that are tested under STS SR 3.3.1.2. The description does not include testing of scram breakers since these breakers do not exist in the ABWR design. The description for testing the solenoid valves is in the final revision to STS Bases Section B 3.3.1.2 which was issued to GE on August 30, 1993. The acceptability of the ABWR technical specifications is discussed in Chapter 16 of this report. This resolved DFSER TS Item 14.2.12.4-1.

- 2.d Final reactor coolant system leak rate test to verify that system leak rates are within specified limits

GE stated in its response of May 20, 1991, that final reactor coolant leak rate testing will have been completed as part of the preoperational testing described in SSAR Section 14.2.12.1.1, which references the required reactor coolant leak rate tests specified in SSAR Section 5.2.4.6.1.

GE revised Section 14.2.12.2.3 in SSAR Amendment 18 to specifically require that the leak rate tests required by RG 1.68, Position 2.d, be completed as prerequisites to fuel loading. This is acceptable.

- 4.k Steam driven plant auxiliaries and power conversion equipment

The staff verified that Section 14.2.12.2 was revised in draft to add SSAR Section 14.2.12.2.39 to address testing of steam and power conversion systems. This was acceptable subject to incorporation into a future SSAR revision. This was DFSER Confirmatory Item 14.2.12.4-2.

The staff verified that Section 14.2.12.2.39 was revised to address testing of steam and power conversion systems in SSAR Amendment 23. The staff finds the revisions acceptable. This resolved DFSER Confirmatory Item 14.2.12.4-2.

- 4.1 Branch steamline valves and bypass valves used for protective isolation functions at rated temperature and pressure conditions

GE stated in its response of May 20, 1991, that the only branch steamline valves used for ABWR protective isolation functions will be those on the reactor core isolation cooling (RCIC) steamline and the common drainline from the main steamlines.

GE revised the description of the RCIC system testing in Section 14.2.12.2.22 in SSAR Amendment 18 to include specific testing of the RCIC steamline isolation valves and revised Section 14.2.12.2.26 in the same amendment to include specific testing of the main steamline branch drain line isolation valves in addition to the main steam isolation valve (MSIV) testing already specified. The staff finds this acceptable.

- 5.j Plant performance is as expected for rod runback and partial scram

GE stated in its response of May 20, 1991, that the ABWR design has no partial scram function. Rod runback will be accomplished by the select control rod run-in (SCRRI) function. GE revised Section 14.2.12.2.5 in SSAR Amendment 18 to ensure that appropriate testing will be performed to demonstrate proper functioning of SCRRI logic and hardware. GE also revised Section 14.2.12.2.30 to ensure that proper plant response is demonstrated during an event that will result in initiation of SCRRI. The staff finds this acceptable.

- 5.n Reactor coolant system loose parts monitoring system

GE added Section 14.2.12.2.36 to address loose parts monitoring system baseline data collection. This was acceptable subject to incorporation into a future SSAR revision. This was DFSER Confirmatory Item 14.2.12.4-3.

GE added Section 14.2.12.2.36 in SSAR Amendment 23 to address loose parts monitoring system baseline data collection. The staff finds Section 14.2.12.2.36 acceptable. This resolved DFSER Confirmatory Item 14.2.12.4-3.

- 5.o Reactor coolant leak detection systems

GE stated in its response of May 20, 1991, that testing of reactor coolant leak detection systems will be completed during the preoperational stage. The staff finds this acceptable.

- 5.q Proper operation of failed fuel detection systems

GE stated in its response of May 20, 1991, that the failed fuel detection function is performed by the process radiation monitoring system, the testing of which is described in SSAR Section 14.2.12.2.1.

GE revised Section 14.2.12.2.1 in SSAR Amendment 18 to require the appropriate demonstration of the related failed fuel detection function. (Also see Item 1.j.(12) above.) The staff finds this acceptable.

- 5.u Branch steamline isolation valve operability and response times

GE revised Sections 14.2.12.2.22 and 14.2.12.2.26 in SSAR Amendment 18 to address branch steamline isolation valve operability and response times. The staff finds this acceptable.

- 5.w Demonstration that concrete temperatures surrounding hot penetrations do not exceed design limits with the minimum design capability of cooling system components available

GE agreed to add SSAR Section 14.2.12.2.37 to address concrete penetration temperature surveys. This was acceptable subject to incorporation into a future SSAR revision. This was DFSER Confirmatory Item 14.2.12.4-4.

GE added Section 14.2.12.2.37 to address concrete penetration temperature surveys in SSAR Amendment 23. The staff finds Section 14.2.12.2.37 acceptable. This resolved DFSER Confirmatory Item 14.2.12.4-4.

- 5.x Auxiliary systems required to support operation of engineered safety features

GE stated in its response of May 20, 1991, that the auxiliary systems required to support operation of ESFs include the cooling water and HVAC systems for which testing is described in SSAR Sections 14.2.12.2.23 and 14.2.12.2.24, respectively.

GE revised Sections 14.2.12.2.23 and 14.2.12.2.24 in SSAR Amendment 18 to ensure that the testing performed and results obtained will ultimately demonstrate the adequacy of a particular auxiliary system's performance under limiting accident conditions. The staff finds this acceptable.

- 5.z Demonstration that process and effluent radiation monitoring systems are responding correctly by performing independent laboratory or other analyses

GE stated in its response of May 20, 1991, that this testing is part of that described in SSAR Section 14.2.12.2.1(3).

GE revised Section 14.2.12.2.1(3) in SSAR Amendment 18 to specifically address, RG 1.68, Position 5.z. The staff finds this acceptable.

- 5.c.c Demonstration that gaseous and liquid radioactive waste processing, storage, and release systems operate in accordance with design

GE agreed to add Section 14.2.12.2.38 to address radioactive waste system testing. This was acceptable subject to incorporation into a future SSAR revision. This was DFSER Confirmatory Item 14.2.12.4-5.

GE added Section 14.2.12.2.38 to address radioactive waste system testing in SSAR Amendment 23. The staff finds Section 14.2.12.2.38 acceptable. This resolved DFSER Confirmatory Item 14.2.12.4-5.

- 5.g.g Demonstration of design features to prevent or mitigate anticipated transients without scram (ATWS)

GE stated in its response of May 20, 1991, that ATWS design features will consist primarily of dedicated logic and some hardware, which will be thoroughly checked as part of the preoperational test program. Most hardware design features perform ATWS-related functions in their normal mode only when initiated by dedicated ATWS logic. Therefore, the functioning of these features has already been adequately verified during the preoperational testing. Thus, no dedicated testing of ATWS-related features will be planned during the power ascension test phase.

GE incorporated in SSAR Amendment 18 specific ATWS-related testing requirements into individual test abstracts (see Item 1.c above). The staff finds this acceptable.

- 5.h.h Demonstration that the dynamic response of the plant to load swings for the facility, including step and ramp changes, is in accordance with design

GE stated in its response of May 20, 1991, that this testing is intended to be a part of that described in SSAR Section 14.2.12.2.16.

GE revised Section 14.2.12.2.16 in SSAR Amendment 18 to demonstrate the dynamic response of the plant to load swings for the facility, including step and ramp changes, is in accordance with design. The staff finds this acceptable.

14.2.12.5 Three Mile Island (TMI) Items

Staff Evaluation of DSER Items

Appendix A to SSAR states that testing described in Chapter 14 is consistent with the BWR Owner's Group response to Action Item I.G.1 of NUREG-0737 as documented in a letter dated February 4, 1981, from D.B. Waters to D.G. Eisenhut. The staff determined that the test abstracts in SSAR Section 14.2.12 that describe testing outlined in Appendix E of this letter should be identified or modified accordingly. This was DSER Open Item 124.

GE stated in its response of May 20, 1991, that testing outlined in Appendix E of the referenced document would be specified in the applicable test abstracts.

GE revised test abstracts 14.2.12.1.1(3)(a), 14.2.12.1.9(3)(j), and 14.2.12.1.44(3)(a) to include a reference to 1A.2.4 of Appendix A to SSAR Chapter 1, and GE revised 1A.2.4 to discuss the requirements of Action Item I.G.1 Appendix E applicable to the ITP. The staff found this item acceptable subject to incorporation into a future SSAR revision. This resolved DSER Open Item 124, and was tracked in the DFSER as DFSER Confirmatory Item 14.2.12.5-1.

GE revised the above test abstracts in SSAR Amendment 23. The staff finds the changes acceptable. This resolved DFSER Confirmatory Item 14.2.12.5-1.

14.2.12.6 Conformance With Other Regulatory Guides

Staff Evaluation of DSER Items

In the DSER, the staff determined that GE should address the concerns of RG 1.56, "Maintenance of Water Purity in Boiling Water Reactor," (Rev. 0), in SSAR Sections 14.2.12.1.19, 14.2.12.1.54, and 14.2.12.2.21. This was DSER Open Item 125.

GE stated in its response of May 20, 1991, that RG 1.56 deals mainly with design related issues, specifically the equipment and instrumentation needed to ensure proper BWR reactor water chemistry. SSAR Sections 14.2.12.1.19, 14.2.12.1.54, and 14.2.12.2.21 describe preoperational and power ascension testing that is adequate to demonstrate that acceptable reactor water chemistry will be maintained by the reactor water clean up system and the condensate filter/demineralizer system. Subsection 14.2.12.1.22 describes the preoperational testing intended to demonstrate the proper functioning of the instrumentation required by RG 1.56. Likewise, SSAR Section 14.2.12.2.1 indicates that a proper reactor water chemistry monitoring program will be in place.

GE agreed to revise SSAR Sections 14.2.12.1.22 and 14.2.12.2.1 to more specifically address functioning of conductivity meters, which are a major focus of RG 1.56. In the DFSER, the staff found this item acceptable subject to incorporation into a future SSAR revision. This resolved DSER Open Item 125, and the issue was tracked in the DFSER as DFSER Confirmatory Item 14.2.12.6-1.

GE made the above revisions to Sections 14.2.12.1.22 and 14.2.12.2.1 in SSAR Amendment 23. The staff finds the revisions acceptable. This resolved DFSER Confirmatory Item 14.2.12.6-1.

In the DSER, the staff determined that Section 14.2.12.2.14, "Feedwater Control," should address the following items in accordance with RG 1.68, "Preoperational and Initial Startup Testing of Feedwater and Condensate Systems for Boiling Water Reactor Power Plants," Revision 1:

(1) Modify the test description for demonstration of the required functionality of the feedwater system at low reactor power (less than or equal to 15-percent reactor power) (RG 1.68. Position 1.C.2.a). This was DSER Open Item 126.

GE stated in its response of May 20, 1991, that this testing is already specified in the current description, and supplemented by information to be included in a test matrix.

The staff verified that GE's proposed test matrix, to be included in the SSAR, identified feedwater system performance and feedwater control system adjustment/confirmation tests to be performed at the nuclear heat-up and low-power testing plateaus. In the DFSER, the staff found this acceptable subject to incorporation into a future SSAR revision. This resolved DSER Open Item 126, and this issue became DFSER Confirmatory Item 14.2.12.6-2.

GE added the test matrix in SSAR Amendment 23. The test matrix contained feedwater system performance and feedwater control system adjustment/confirmation tests which are performed at the nuclear heat-up and low-power testing plateaus. The staff finds this to be acceptable. This resolved DFSER Confirmatory Item 14.2.12.6-2.

(2) Modify or clarify the test acceptance criteria to provide assurance that vibration levels for system components and piping are within predetermined limits (RG 1.68, Position 1C.2.f); piping movement during heat up and steady state and transient operation are within predetermined limits (RG 1.68, Position 1.C.2.g); and adequate margins exist between system variables and set points of instruments monitoring these variables to prevent spurious actuation or loss of system pumps and motor-operated valves (RG 1.68, Position 1.C.2.h). This was DSER Open Item 127.

GE stated in its response of May 20, 1991, that the testing called for by Positions C.2.f and C.2.g is included in the test abstracts of SSAR Sections 14.2.12.1.51, 14.2.12.1.53(3)(b) and (k), 14.2.12.2.10, 14.2.12.2.11, and 14.2.12.2.18. GE further stated that it revised Section 14.2.12.2.18 in SSAR Amendment 18 to more specifically address Position C.2.h. In the DFSER, the staff verified the above, and found it acceptable. This resolved DSER Open Item 127.

In the DSER, the staff determined that, for SSAR Section 14.2.12.1.27, GE should address the following items in accordance with RG 1.68:

(1) Determine that the total air demand at normal steady-state conditions, including leakage from the system, is in accordance with design (RG 1.68, Position 3.C.5). This was DSER Open Item 127.

GE revised Section 14.2.12.1.27(3)(f) in SSAR Amendment 18 to include testing of the total air demand at normal steady state-conditions, including leakage from the system, in accordance with the design as specified in Position 3.C.5. The staff finds this acceptable. This resolved DSER Open Item 127.

(2) Demonstrate that the plant equipment designated by design to be supplied by the instrument and control air system is not being supplied by other compressed air supplies (such as service air (SA)) that may have less restrictive air quality requirements (RG 1.68, Position 3.C.9). This was DSER Open Item 128.

For Position 3.C.9, GE stated in its response of May 20, 1991, that the SA system acts as a backup to instrument air upstream of the instrument air filters. Furthermore, although totally separate (except for the manual backup cross-tie), the design of the two systems is essentially identical. Thus, the air supplied to the inlet of the instrument air filters is of the same quality, whether it is sourced from the instrument or SA system; therefore, the outlet air will be of the same quality. Because the design precludes occurrence of the conditions hypothesized, no specific test demonstration is needed beyond the construction verification and preoperational testing already planned. The staff finds this acceptable. This resolved DSER Open Item 128.

(3) Demonstrate that functional testing of instrument and control air systems important to safety is performed to ensure that credible failures resulting in an increase in the supply system pressure will not cause loss of operability (RG 1.68, Position 3.C.11). This was DSER Open Item 129.

For Position 3.C.11, GE revised Section 14.2.12.1.27 in SSAR Amendment 18 to include testing that will demonstrate continued operability of supplied loads in response to credible failures that result in an increase in the supply system pressure. The staff finds this item acceptable. This resolved DSER Open Item 129.

In the DSER, the staff determined that GE should address the control room habitability concerns of RG 1.95 in SSAR Section 14.2.12.1.34. This was DSER Open Item 130.

GE revised Section 14.2.12.1.34 in SSAR Amendment 18 to indicate that the control room habitability function will be included in the testing specified for the dedicated HVAC system of the main control room. GE also added a specific requirement to demonstrate the system capability to detect smoke and/or toxic chemicals and to remove and/or prevent in-leakage of such. (Also see RG 1.68, Position 1.n.(14)(f), in Section 14.2.12.4 of this report.) The staff finds this acceptable. This resolved DSER Open Item 130.

In the DSER the staff determined that GE should address, in SSAR Section 14.2.12.1.34, HVAC Systems Preoperational Tests, or other appropriate preoperational tests, the concerns for in-place testing of HEPA and charcoal filters of Position C.5, In-Place Testing Criteria, of RG 1.140. This was DSER Open Item 131.

GE revised Section 14.2.12.1.34 in SSAR Amendment 18 to include testing requirements specified by RG 1.140 and by the industry standards referenced therein. The staff finds this acceptable.

GE later determined that Position C.5, of RG 1.140 only discussed testing criteria for HEPA and carbon absorber filters in HVAC systems. The HVAC systems described in Section 14.2.12.1.34 do not contain HEPA or carbon absorber filters, therefore, Position C.5 of RG 1.140 does not apply. RG 1.52 addresses testing of HEPA and carbon absorber filters in ESF systems. GE added RG 1.52 and removed RG 1.140 in Sections 14.2.12.1.34 and 14.2.12.1.36 of a markup dated June 7, 1993, to address testing of ESF HVAC systems (i.e., control room habitability HVAC system and the standby gas treatment system (SGTS)) that require in place testing of HEPA and carbon adsorber filters. This information was incorporated into Amendment 30 of the SSAR. The staff finds this acceptable. This resolved DSER Open Item 131.

Finally, the staff determined in the DSER that GE should address the following items in accordance with RG 1.139 in SSAR Section 14.2.12.1.8.

(1) Residual heat removal (RHR) system isolation (RG 1.139, Position C.2). This was DSER Open Item 132.

GE stated in its response of May 20, 1991, that the applicable demonstrations were intended to be a part of the testing described in SSAR Section 14.2.12.1.8(3)(i).

Further, GE agreed to revise SSAR Section 14.2.12.1.8 (1) to specifically address testing of features designed to ensure isolation of low-pressure portions of the RHR system from the reactor coolant system (RCS) at high pressure. In the DFSER, the staff found this item acceptable subject to incorporation into a future SSAR revision. This was DFSER Confirmatory Item 14.2.12.6-3.

GE revised SSAR Section 14.2.12.1.8 (1) in SSAR Amendment 23 to include testing of features designed to ensure isolation of the low pressure portions of the RHR system at high pressures. The staff finds this acceptable. This resolved DFSER Confirmatory Item 14.2.12.6-3.

(2) RHR system pressure relief (RG 1, Position 139.C.3). This was DSER Open Item 133.

GE stated that the design of the RHR system will include the relief capacity required by Position C.3 in accordance with the applicable ASME Code. GE indicated that no specific additional preoperational test was needed because verification of the proper setting of relief valves was a vendor bench test required per the same ASME Code.

Section 14.2.12.1.8 was revised to allow for verification of proper set points of system relief valves per ASME code requirements (including those intended to meet the requirements of RG 1.139 using the results of vendor tests and the appropriate documentation of such). In the DFSER, the staff found this item was acceptable subject to incorporation into a future SSAR revision. This was DFSER Confirmatory Item 14.2.12.6-4.

GE added system pressure relief valve verification tests per ASME code and RG 1.139 requirements into Section 14.2.12.1.8 (3) in SSAR Amendment 23. The staff finds the above tests acceptable. This resolved DFSER Confirmatory Item 14.2.12.6-4.

14.2.12.7 Certified Design Material

The staff reviewed the GE certified design material and determined that the ITPs were not included in the Tier 1 or ITAAC material. The staff expected that this Tier 1 material would be at a high level limited to a commitment to an ITP and a description of the program and the major program documents (i.e., a site-specific startup administrative manual, test specifications, and test procedures). This was DFSER Open Item 14.2.12.7-1.

GE provided the Tier 1 ITP information. An evaluation of the ITP information is contained in Section 14.3.3.5 of this report. The staff has found the description acceptable. This resolved DFSER Open Item 14.1.12.7-1.

14.2.13 COL License Information - Initial Test Program

The preceding discussion of preoperational and startup tests were limited to those systems and components within the scope of, or directly related to, the ABWR standard plant. Other testing, with respect to utility specific aspects of the plant will be necessary to satisfy certain ABWR requirements. Testing of such systems and components should be adequate to demonstrate conformance to such requirements as defined throughout the specific chapters of the SSAR. Below are systems that may require such testing

(1) electrical switchyard and equipment
(2) the site security plan
(3) personnel monitors and radiation survey instruments
(4) the automatic dispatcher control system (if applicable)

Also to be supplied by the applicant referencing the ABWR design is the startup administrative manual described in Section 14.2.4, which will describe, among other things, what specific permissions are required for the approval of test results and the permission to proceed to the next testing plateau.

The staff received a GE response dated May 7, 1993, containing a new section (i.e., Section 14.2.13, COL License Information). Section 14.2.13 lists all the COL Action Items (i.e., DFSER COL Action Items 14.2.3-1 through 14.2.12.4-2) that were identified in the DFSER. The staff found that the new section lists all COL license information that will be provided by the COL applicant. The COL Action Items list was subsequently incorporated into the SSAR. The staff finds the COL license information provided in Section 14.2.13 acceptable.

<u>Conclusion</u>

The staff performed the review of the plant ITPs in accordance with Section 14.2, of NUREG-0800, the SRP, and RG 1.68 (Rev. 2), "Initial Test Programs for Water Cooled Nuclear Power Plants." The staff reviewed thirteen areas relating to initial plant test programs, described in Chapter 14 of the SSAR, and submitted by GE as part of its design certification application. These areas of review included:

- summary of the test program and objectives
- organization and staffing
- test procedures
- conduct of test program
- review, evaluation, and approval of test results
- test records
- conformance of test program with regulatory guides
- utilization of reactor operating and testing experience in the development of test program
- trial use of plant operating and emergency procedures
- initial fuel loading and initial criticality
- test program schedule
- individual test descriptions
- combined license (COL) information - ITP

The acceptance criteria used by the staff for this review were contained in SRP Section 14.2, Subsection II.

Based on the review of the GE ABWR ITP description in SSAR Chapter 14 and the responses to the RAIs and the DSER open items, DFSER open items and confirmatory items, the staff concluded that the ITP was generally comprehensive and covered all areas of staff concern. The staff also conducted an in-depth review of system-specific testing requirements within each test abstract. The staff concluded that GE provided a sufficient level of detail to adequately describe system-specific test prerequisites and acceptance criteria for design certification. The staff also reviewed cross-references to acceptance criteria information in other parts of the SSAR and found them acceptable. An additional review of COL action items and interfacing systems will be performed at the time a COL applicant referencing the ABWR design submits an application for a license under 10 CFR Part 52.

14.3 Certified Design Material

Introduction

The objective of this section of the report is to provide the basis for the staff's approval of the certified design material (CDM) for the ABWR design. This section of the report is based on the staff's review of the GE document titled "ABWR Certified Design Material" and standard safety analysis report (SSAR) Section 14.3, "Certified Design Material." The requirement to submit this information as part of an application for design certification is contained in 10 CFR 52.47.

The GE document titled "ABWR Certified Design Material" contains the information that will be certified by the rule that approves the ABWR design. The CDM consists of an introductory section, design descriptions and corresponding inspections, tests, analyses, and acceptance criteria (ITAAC) for the systems of the design, design material applicable to multiple systems of the design, interface requirements, and site parameters for the ABWR design. This information is also referred to as Tier 1 information. The ABWR SSAR Section 14.3, "Certified Design Material," provides the bases and methods that were used to develop the information for each of these CDM items for the ABWR. This report documents the staff's review of the ABWR CDM, as supported by the design information contained in the ABWR SSAR.

The information in the CDM and SSAR Section 14.3 is derived from the detailed information contained in the SSAR. Further, the purpose of the ITAAC, which are part of the CDM, is to verify that a facility that references the design certification has been built and will operate in accordance with the design certification and the applicable regulations. Consequently, there is no design information presented in the CDM or SSAR Section 14.3 that is not also contained in the various sections of the SSAR. Therefore, the staff did not base its safety evaluations for the design on the information in the CDM.

Basis for Approval of the CDM

The ABWR was the lead design for the initial development of CDM for design certification. Although the staff was able to make its safety determinations for the design based on its review of the information in the SSAR, this was the first time design material had been developed for certification. Therefore, there was no precedent for GE to follow to develop information for the CDM. Furthermore, the staff had no regulatory guidance on which to base its review. Recognizing this, the staff sought and received Commission guidance on several key issues associated with the design certification reviews in the staff requirements memorandum (SRM) dated February 15, 1991, relating to SECY-90-377, "Requirements for Design Certification Under 10 CFR Part 52." These issues included the development of regulatory guidance, the role of ITAAC, the level of design detail needed for design certification, issue finality, the two-tiered approach to the design certification rule structure, and flexibility in design change process. In its review, the staff evaluated the CDM in the context of Part 52 requirements and the Commission guidance contained in that SRM.

As the lead design, GE submitted the CDM to the staff for review in stages, so that lessons learned at each stage could be incorporated into later submittals. The staff reviewed the material in an iterative manner and provided comments on the CDM to GE at each stage. The staff informed the Commission of the development of the CDM in multiple SECY papers issued in 1991 and 1992 (SECYs 91-178, 91-210, 92-053, 92-196, 92-214, 92-287 and 287A, 92-299, and 92-327). These papers are listed as references in Appendix B of this report.

In SSAR Section 14.3, GE provided the process it used in the development of the CDM, based on the design presented in the appropriate sections of the SSAR. GE provided CDM based on the structures and systems of the ABWR design rather than based on the format of the ABWR SSAR. In addition, GE adopted a graded approach to the level of design detail for the information in the CDM, based on the safety significance of particular structures, systems, and components (SSCs). GE applied various selection criteria to the information in the SSAR to determine the level of design information for a given structure or system in the CDM. The results of this process were illustrated with cross-references from the SSAR information to the CDM for important parameters that were selected for treatment in the CDM. Although many issues and analyses could have been cross-referenced, the listings in Section 14.3 were developed only for selected integrated plant safety analyses for the ABWR design. GE provided additional cross-references of key insights and assumptions from probabilistic risk analyses (PRA) and analyses for severe accidents which are contained in SSAR Chapter 19. GE provided more detailed cross-references to the CDM for these analyses in a letter dated March 31, 1994.

The staff also utilized a graded approach to the level of detail in its review of the CDM based on the safety significance of the SSCs. Thus, consistent with the guidance of Part 52 and the SRM related to SECY-90-377, the staff recognized that although many aspects of the design were important to safety, the level of design detail in the CDM and verification of the key design features and performance characteristics should be commensurate with the significance of the safety functions to be performed.

The intent of the CDM is to ensure that the key characteristics and performance requirements of safety-significant SSCs are implemented in an as-built facility referencing the certified design. Although all these aspects of the design are described in the CDM, not all can be verified by the ITAAC because Part 52 requires that the ITAAC be satisfied prior to fuel loading. The initial test program (ITP) serves to verify the remaining aspects of the design after fuel load, but prior to operation. Examples of these requirements are the post-fuel load startup and power ascension test program verification of fuel, control rod, and core characteristics, as well as system and integrated plant operating characteristics. The treatment of these issues will be similar to their treatment at facilities licensed under 10 CFR Part 50, in that verification of the satisfactory completion of these requirements will be a condition of the license.

The staff recognized that other programs also ensure the continued safe operation of a facility after fuel load. For example, the continued operability of a facility after the ITAAC are satisfied is ensured through the technical specifications (TS), as well as various programs such as the maintenance program, quality assurance program, and the in-service inspection and testing program. The operator ensures the facility is operated as designed, through the use of appropriate plant operating and emergency procedures. Additionally, a utility referencing the design is required by 10 CFR Part 50, Appendix B, to have a quality assurance (QA) program that ensures these SSCs are appropriately designed, procured, and perform satisfactorily in service.

The above considerations provided an overall framework for GE's development of the CDM and the staff's review. The staff utilized multiple sources of information to determine the safety significance of SSCs in the CDM. These sources included the SRP, applicable rules and regulations, general design criteria (GDC), regulatory guides (RGs), unresolved safety issues (USIs) and generic safety issues (GSIs), NRC generic correspondence, operating experience, NRC inspection programs, facility testing programs, PRA, and insights from ABWR safety and severe accident analyses. For selected portions of the review, the staff also utilized the regulatory guidance from the Commission related to SECY-90-016, "Evolutionary Light Water Reactor Certification Issues and Their Relationship to Current Regulatory Requirements," as modified by the Commission guidance related to SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor Designs."

Nonetheless, because this was the first time CDM had been developed for a design, because there was no precedent for GE to follow and no detailed review guidance for the staff to base its review on, and because of the iterative nature of the ITAAC development process, considerable judgement was inherent in the approval of the final material for the CDM.

Background

Part 52 of Title 10 of the Code of Federal Regulations was issued April 18, 1989. The concept in Part 52 of certifying a design and specifying the required ITAAC in a rule prior to construction of a facility had not been attempted before. Consequently, GE and the Nuclear Management and Resources Council (NUMARC) held extensive discussions with the staff in 1990 and 1991 on the CDM, especially the form and content of the ITAAC. After several senior management meetings in mid-1991, the GE ABWR emerged as the lead plant in the development of the first CDM for a standardized design.

The development of the CDM was an iterative process. As the lead design, GE submitted the CDM to the staff for review in stages, so that lessons learned at each stage could be incorporated into later submittals. The staff reviewed the material as it was submitted and provided comments on the CDM to GE at each stage. Senior management meetings between GE and the staff were held at intervals of approximately six to eight weeks to resolve difficult policy and technical issues associated with the development and review of the material.

GE submitted the first stage of the CDM, consisting of a set of nine "pilot" ITAAC, in a submittal dated September 20, 1991. The staff provided comments on this submittal in a letter of October 23, 1991, and after several meetings with the staff to discuss the comments, general agreement was reached on the pilot ITAAC in January 1992.

GE submitted the second stage of the CDM, consisting of approximately one-half the ITAAC, in a submittal titled "Tier 1 Design Certification Material for the GE ABWR Design - Stage 2 Submittal," dated March 31, 1992. The staff provided preliminary comments to GE on the Stage 2 submittal in a letter dated May 7, 1992, as well as during several senior management meetings.

GE submitted the third stage of the CDM, consisting of the remainder of the Tier 1 material, in its submittal titled "Tier 1 Design Certification Material for the GE ABWR," dated June 1, 1992, as supplemented by a letter dated June 17, 1992. The staff provided detailed comments on the CDM in a letter to GE dated August 12, 1992, including the comments of an NRC review group comprised of senior managers from several headquarters and regional offices of the NRC.

In September 1992, a group comprised of representatives from various vendors, utilities, and industry groups provided industry comments on the CDM to GE. GE revised the CDM to incorporate the comments provided by both the NRC and this review group in late 1992. Subsequently, a team of NRC reviewers met with GE in January, February, and March 1993, to review the revised ITAAC.

GE submitted the next stage of the CDM, consisting of a revised, complete CDM that incorporated all previous lessons learned and review comments, in incremental submittals in April, May, and June of 1993. In May 1993, the staff formed several task groups to perform a multidisciplinary review of the CDM. The task groups provided extensive comments on the submittals to GE in a letter dated July 9, 1993. The task groups met with GE on July 27 through 29, 1993, to discuss the comments, and the disposition of all of the comments were documented in a meeting summary dated August 10, 1993.

GE submitted the next stage of the CDM, consisting of a revised, complete CDM that incorporated the comments from the staff's task group review, in a submittal titled "ABWR Certified Design Material," dated August 31, 1993. GE supplemented this material with CDM Revision 1 in September 1993, and with CDM Revision 2 in December 1993. Changes to the detailed CDM supporting information in the SSAR were contained in various amendments to the SSAR, up to and including Amendment 33.

The staff had identified multiple inconsistencies between the SSAR, the Tier 1 design descriptions, and the related ITAAC during the reviews. Consequently, the staff formed an independent review group to ensure consistency of the CDM and the SSAR in November 1993. The independent review group completed its review in February 1994, and its comments were provided to GE after review by the staff. GE provided satisfactory resolutions to all the comments in CDM Revision 3 and SSAR Amendment 34. In addition to resolving the comments of the independent review group, CDM Revision 3 also included resolutions to comments by the ACRS and the staff. Subsequent revisions to the CDM prior to the start of the administrative review of the ABWR design were minor in nature.

Format of the CDM

GE developed a format for the design certification information to meet the requirements of 10 CFR 52.47, including bounding parameters for siting of the standard design, design descriptions and corresponding ITAAC, and interface requirements for the design. The CDM is the portion of the design information that is certified by the rule certifying the design, and will be incorporated into the rule as the Tier 1 part of the design control document (DCD). The DCD is the master document that contains the information that must be conformed with by an applicant who references the rule. The format for this material is listed below, and is discussed in more detail in the following sections of this report.

(1) Introduction - Definitions of terms used in the CDM, and a listing of general provisions that are applicable to all CDM entries.

(2) System Design Descriptions and ITAAC - System design descriptions and ITAAC are provided for: (a) structures and systems that are fully within the scope of the ABWR design certification, and (b) the in-scope portions of those systems that are only partially within the scope of the ABWR design certification. The system design descriptions are accompanied by the appropriate ITAAC.

(3) Additional Certified Design Material - Design descriptions and their related ITAAC for design and construction activities that are applicable to more than one system of the design. This additional material was provided because in selected areas of the design, GE did not provide sufficient design detail in the SSAR. GE did not provide complete design information in these areas because they were either areas of rapidly changing technology where GE believed it was unwise to prematurely freeze the design, or because the information was dependent on as-built or as-procured information. For these areas, GE provided the design related processes in the CDM and in the SSAR, with appropriate codes and standards, that a COL applicant or licensee would follow to complete the design.

(4) Interface Requirements - Requirements that must be met by the site-specific portions of a facility that are not within the scope of the certified design. This section also identifies the scope of the design to be certified. Interface requirements are defined for: (a) systems that are entirely outside the scope of the design, and (b) the out-of-scope portions of those systems that are only partially within the scope of the ABWR design.

(5) Site Parameters - Bounding parameters of the design to be used in the selection of a suitable site

for a facility referencing the ABWR certified design. The design was evaluated in terms of these parameters. A suitable site must be demonstrated to be within the bounding parameters and characteristics, and a facility must be constructed at the site in accordance with their use in the approved design. If a site cannot meet them, an exemption must be requested and the facility must be reevaluated in terms of these parameters for the actual selected site.

14.3.1 Introduction to CDM

14.3.1.1 Definitions

This section of the CDM provides terms used in the CDM that could be subject to various interpretations. The intent of the terms used in the CDM was to be consistent and as closely aligned as possible with the terminology in the SSAR, in common industry use, industry codes and standards, and NRC rules, regulations, and guidance. Thus, should questions on terminology arise, these references would aid in understanding the intent of the information in the CDM.

14.3.1.2 General Provisions

This section of the CDM provides general provisions that are applicable to the design descriptions, figures, and the ITAAC.

14.3.1.2.1 Verifications for Basic Configuration for Structures and Systems

This section of the CDM includes provisions related to the verification of the ITAAC for basic configuration for systems and structures of the design. This ITAAC is contained in the buildings and many of the systems described in Section 2 of the CDM. The verification consists of an inspection of the system functional arrangement in its final as-built condition at the plant site, and includes the elements of the design descriptions and the system figures in the CDM. This functional arrangement inspection verifies, using as-built system drawings, design documentation, and in-situ plant walkdowns, that the as-built facility is in conformance with the certified design and applicable regulations.

Several other aspects of the design were considered to have significance to the performance of safety functions of SSCs of a facility. The basis for selecting these aspects included its importance to safety as well as its past experience with construction and operating problems. Thus, specific inspections for these aspects are part of the basic configuration ITAAC for systems and structures. The other inspections to be conducted to satisfy this ITAAC include, and are limited to, verification of the following:

(1) Verifications of the quality of pressure boundary welds for ASME Code Class 1, 2, and 3 components and systems described in the design descriptions and figures. Detailed supporting information for verification of welding requirements in accordance with ASME Code requirements is contained in SSAR Chapter 3.

(2) Verifications of the dynamic qualification (e.g., seismic, LOCA, and safety relief valve discharge loads) of seismic Category I mechanical and electrical equipment (including connected instrumentation and controls) described in the design descriptions and figures. Detailed supporting information for dynamic qualification requirements, including qualification records, is contained in SSAR Chapter 3.

(3) Verifications of the environmental qualification of Class 1E electrical equipment described in the design descriptions and figures. Detailed supporting information for environmental qualification requirements is contained in SSAR Chapter 3.

(4) Verifications of the design qualification of motor-operated valves (MOVs) described in the design descriptions and figures. Detailed supporting information for design qualification of MOVs is contained in SSAR Chapter 3.

14.3.1.2.2 Treatment of Individual Items

A licensee is not prohibited from utilizing an item not described in the CDM. However, the as-built facility must be consistent with the rule approving the design, including both tiers of information. The change processes for the certified design are described in the design certification rule for the ABWR.

The term "operate" as utilized in the CDM is intended to refer to the actuation and running of equipment. This is not meant to include the term "operable" in the context of the ongoing reliability and availability of equipment. In developing the ITAAC, the staff recognized that other programs ensure the continued safe operation of a facility after fuel load. For example, the continued operability of a facility after the ITAAC are satisfied is ensured through the Technical Specifications, Startup and Power Ascension Test Programs, as well as various programs such as the maintenance program, quality assurance program, and the in-service inspection and in-service testing program. Also, the operator ensures the facility is operated as designed, through the use of appropriate plant operating and emergency procedures.

The term "exists," when used in the Acceptance Criteria, means that the item is present and meets the design description. Detailed supporting information on what must be present to conclude that an item "exists" and meets the design description is contained in the appropriate sections of the SSAR.

14.3.1.2.3 Implementation of ITAAC

GE developed a three column format for the ITAAC. The design commitments in the first column are derived from the design information in the design descriptions. The inspections, tests, and analyses in the middle column provide the intended means of verifying the design commitment. The acceptance criteria in the third column provide the criteria used to determine whether the design commitment is met.

The licensee is required by 10 CFR Part 52 to perform the required inspections, tests, and analyses for the design, and certify to the NRC that the acceptance criteria have been met. A licensee may utilize the efforts of subordinate vendors, contractors, or consultants. However, the licensee referencing the certified design retains responsibility for ensuring that the ITAAC are met. Additionally, the ITAAC can be satisfied using other programs, such as the pre-operational testing portion of the ITP required by CDM Section 3.5, or the QA program required by 10 CFR Part 50, Appendix B.

The ITAAC may be satisfied at any time prior to fuel load, including prior to issuance of a combined license. However, the primary intent of the ITAAC is to verify that the as-built plant on the final site has been constructed and will perform in accordance with the design certification and applicable regulations.

14.3.1.2.4 Discussion of Matters Related to Operations

Descriptions in the CDM may refer to matters of operation, such as normal valve or breaker alignment during normal operational modes. These descriptions are not intended to require operators to take any particular action. The operational matters referred to in the CDM are governed by existing programs to ensure the ongoing safe operation of a facility, such as plant operating and emergency procedures.

14.3.1.2.5 Interpretation of Figures

The design descriptions include the figures in the CDM, where the figures are provided. They are intended to depict the functional arrangement of the significant SSCs of the ABWR design. An as-built facility referencing the certified design must be consistent with the performance characteristics and functions described in the design descriptions and figures. Any changes to the detailed information in the SSAR must be in accordance with the "50.59-like" change process in the design certification rule for the ABWR, which allows the COL applicant or licensee to make design changes, provided the changes do not impact the information in the CDM.

14.3.1.2.6 Rated Reactor Core Thermal Power

The rated reactor core thermal power for the ABWR is 3926 MW(th).

14.3.1.3 Conclusions

As discussed above, the staff reviewed the definitions and general provisions that are contained in CDM Section 1.0, and the supporting material contained in SSAR Section 14.3.1, in accordance with the requirements in Part 52 and the guidance in SRMs related to design certification applications provided by the Commission. Based on this, the staff concludes that the definitions and general provisions in the CDM are appropriate to support the design descriptions and ITAAC, and are acceptable.

14.3.2 Certified Design Material for Structures and Systems

GE developed design descriptions and ITAAC for the structures and systems of the ABWR design, and these are contained in CDM Section 2.0. General provisions that apply to most of the structures and systems are contained in CDM Section 1.2. Additional CDM material for design issues that apply to many of these structures and systems are contained in CDM Section 3.0. Interface requirements for these systems are provided in the system design descriptions for the in-scope portions of the systems. The interface requirements for the out-of-scope portions of the systems of the design are contained in CDM Section 4.0. GE provided an entry in the CDM for every system of the design to define the full scope of the design.

14.3.2.1 Design Descriptions

The design descriptions address the most safety-significant aspects of each of the systems of the design, and were derived from the detailed design information contained in the SSAR. The design descriptions include the figures associated with the systems. GE's selection criteria and methodology for the system design descriptions are specified in SSAR Section 14.3.2.1. In its review of the material, the staff followed the general guidance for the reviews specified in the SRM related to SECY-90-377, as discussed previously in the introduction to Section 14.3 of this report.

The Tier 1 design descriptions will serve as commitments for the lifetime of a facility. Once completion of ITAAC and the supporting design information demonstrate that the facility has been properly constructed, it then becomes the function of existing programs such as the technical specifications, the in-service inspection and in-service testing program, the quality assurance program, and the maintenance program, to demonstrate that the facility continues to operate in accordance with the certified design and the license. Nevertheless, the Tier 1 design descriptions will remain in effect throughout the plant life to assure that the plant does not deviate from the certified design. In general, a COL applicant or licensee may change the information in the SSAR in accordance with the "50.59-like" change process described in the rule certifying the design, provided that the change does not impact the information in the design descriptions.

GE provided the selection criteria for information in the design descriptions in Section 14.3.2.1. Essentially, GE put the top-level design features and performance standards that were most significant to safety in the design descriptions. The criteria GE utilized in determining the safety significance of SSCs in the design descriptions included the NRC's regulations, whether or not the information pertained to safety-related SSCs, the importance in the SRP, the relative importance based on PRA or severe accident analysis, operating experience, or the technical specifications. GE also included other SSCs based on their importance to safety. Non-safety aspects of SSCs were generally not discussed in the design descriptions. Thus, although a Tier 1 entry was provided for every system that is either fully or partially within the scope of the ABWR design certification, the amount of information provided in CDM Section 2.0 for a given system, if any, was based on the safety significance of the system.

GE provided additional certified design material applicable to the systems of the design in CDM Section 3.0. The design descriptions in CDM Section 3.0 describe the scope and applicability of the additional certified design material to the systems of the design in CDM Section 2.0. Amplifying information on CDM Section 3.0 is provided in SSAR Section 14.3.3, and the staff's review of CDM Section 3.0 is contained in Section 14.3.3 of this report.

The CDM utilizes a system-based structure which is different than the structure of the SSAR. Consequently, developing the CDM design description entries for any one system was based on a review of the multiple SSAR chapters having technical information related to that system. GE illustrated this approach in SSAR Section 14.3.2.1, showing how the many design aspects of any single system in the CDM were derived from multiple chapters of the SSAR.

The staff was particularly interested in ensuring that the assumptions and insights from key safety and integrated plant safety analyses in the SSAR, where plant performance was dependent on contributions from multiple systems of the design, were adequately considered in the CDM. Addressing these assumptions and insights in the CDM ensures that the integrity of the fundamental analyses for the design are preserved in an as-built facility referencing the certified design. These analyses included flooding analyses, overpressure protection, containment analyses, core cooling analyses, fire protection, transient analyses, and radiological analyses.

GE provided information regarding these analyses in the SSAR, and documented the important design information and parameters from the various chapters of the SSAR that are addressed in the CDM in Tables 14.3-1 through 14.3-10. GE provided more detailed cross-references to the CDM for these analyses in a letter dated March 31, 1994. GE also provided cross-references in SSAR Chapter 19 showing how key insights and assumptions from PRA and severe accident analyses are addressed in the CDM. A COL applicant or licensee proposing to change design information in the SSAR that pertained to these analyses via the "50.59-like" change process can use these cross-references when considering whether the proposed change impacts the treatment of these parameters in the CDM.

14.3.2.2 ITAAC

The purpose of the ITAAC is to verify that an as-built facility conforms to the approved plant design and applicable regulations. When coupled in a COL with the ITAAC for site-specific portions of the design, they constitute the verification activities for a facility that must be successfully met prior to fuel load. If the licensee demonstrates that the ITAAC are met and the staff agrees that they are successfully met, then the licensee will be permitted to load fuel.

The scope of the ITAAC is consistent with the SSCs that are in the design descriptions. In general, each system has one or more ITAAC that verify the information in the design descriptions. This is not true in all cases. Reasons for not requiring an ITAAC verification for a Tier 1 design commitment include: (1) the information is only included for context, (2) fulfillment of other ITAAC are sufficient to show verification of the design commitment, (3) a single ITAAC can verify more than one design commitment, or (4) verification of the item can only occur after fuel loading. For the last item, the staff reviewed the power ascension testing program described in SSAR Chapter 14 to ensure that all important design features and commitments that could not be verified prior to fuel load were addressed where appropriate.

The staff reviewed the system ITAAC to ensure that the verifications were consistent with the safety significance of the key design characteristics and performance requirements of the SSC verified by that ITAAC. The certified design descriptions for an SSC contain the significant functions and bases for that SSC. Therefore, the ITAAC have been reviewed to ensure they are necessary and sufficient to provide the NRC with reasonable assurance that the facility should be authorized to load fuel. As a result, the ITAAC verify the significant design features from the design descriptions and the applicable requirements that are necessary and sufficient to authorize fuel loading and subsequent operation.

The staff and industry reached agreement on a three-column format for ITAAC, as discussed below.

Column 1 - Design Commitment

This column contains the text for the specific design commitment that was extracted from the design descriptions discussed above. Any differences in text were minimized, unless intentional. Differences in text were generally intended to better conform the commitments in the design description with the ITAAC format.

Column 2 - Inspections, Tests, and Analyses

This column contains the specific method to be used by the licensee to demonstrate that the design commitment in Column 1 has been met. The method is either by inspection, test, or analysis or some combination of inspections, tests, or analyses.

The SSAR contains detailed supporting information for the CDM about various inspections, tests, and analyses that can, and should be, used to verify the Tier 1 design information and satisfy the acceptance criteria. If questions on interpretation should arise, the material in the SSAR provides the background material and context for the CDM. The SSAR contains information reviewed by the staff which was the basis for the staff's safety determination for the design. Therefore, the information in the SSAR provides an acceptable means of satisfying an ITAAC.

Inspections are defined in CDM Section 1.1, and include visual and physical observations, walkdowns or record reviews. The inspections required for the "Basic Configuration Walkdown" ITAAC invoke the general provisions contained in CDM Section 1.2 for as-built structures and systems.

Tests are defined in CDM Section 1.1, and mean the actuation, operation, or establishment of specified conditions to evaluate the performance or integrity of the as-built SSCs. This includes functional and hydrostatic tests for the systems. The term "as-built" is intended to mean testing in the final as-installed condition at a facility. The term "type tests" is used in this column to mean manufacturer's tests or other tests that are not necessarily intended to be in the final as-installed condition. The results of pre-operational tests can be used to satisfy an ITAAC. In its review, the staff did not rely on the pre-operational tests described in SSAR Section 14.2 or RG 1.68 to substitute for ITAAC. Where testing is specified, appropriate conditions for the test should be established in accordance with the ITP described in CDM Section 3.5, SSAR Section 14.2 and RG 1.68. Conversion of the test results from the test conditions to the design conditions may be required to satisfy the ITAAC.

During its review, the staff emphasized in-situ testing, where possible, of the as-built facility as the preferred means to satisfy the ITAAC. Also, the staff recognized that the results and documentation from facility programs such as the quality assurance program or the ITP may be used to satisfy an ITAAC.

Analyses are defined in CDM Section 1.1, and may refer to detailed supporting information in the SSAR, simple calculations, or comparisons with operating experience or design of similar SSCs. For example, detailed analysis methods of seismic and environmental qualification supporting CDM Section 1.2 are contained in SSAR Chapter 3, and detailed piping design information supporting CDM Section 3.3, are also contained in SSAR Chapter 3.

Column 3 - Acceptance Criteria

This column contains the specific acceptance criteria for the inspections, tests, or analyses described in Column 2 which, if met, demonstrate that the design commitment in Column 1 has been met.

In general, the acceptance criteria were developed to be objective and unambiguous. In some cases, the acceptance criteria were more general because the detailed supporting information in the SSAR did not lend itself to concise verification. For example, the acceptance criteria for the design integrity of piping and structures is that a report "exists" that concludes the design commitments are met. In these cases, the SSAR provides the detailed supporting information on multiple interdependent parameters that must be provided in order to demonstrate that a satisfactory report exists.

Numeric performance values for SSCs were specified as ITAAC acceptance criteria when the design commitment so lent itself, or when failure to meet the stated acceptance criterion would clearly indicate a failure to properly implement the design. The staff did not require that numeric performance values be specified in the design description unless there was a specific reason to include them (e.g., important to be maintained for the life of the facility).

ITAAC Implementation

The ITAAC may be satisfied at any time prior to fuel load, including prior to issuance of a combined license. However, the primary intent of the ITAAC is to verify that the as-built facility on the final site has been constructed and will operate according to the design certification and applicable regulations.

The implementation of a construction verification program, including ITAAC, and other licensee programs, is the responsibility of the licensee. The successful completion of the ITAAC in the combined license will constitute the basis for the NRC's determination to allow fuel loading for the facility.

The licensee will periodically certify to the NRC that the inspections, tests, and analyses have been performed, and that the acceptance criteria have been met. These notifications should document the basis for the successful completion of the ITAAC. In accordance with 10 CFR 52.99, the staff will assure that the required inspections, tests, and analyses have been performed and that the prescribed acceptance criteria have been met. At appropriate intervals, the NRC will publish in the Federal Register, notices of the successful completion of the inspections, tests, and analyses.

14.3.2.3 Staff Review Approach

GE developed the CDM based on the systems of the design rather than on the format of the SSAR and SRP. In order to ensure that the safety-significant design information in the SSAR was adequately reflected in the CDM, the staff adopted a multidisciplinary review approach, rather than the more traditional review approach based on the individual chapters of the SSAR. The staff formed several task groups comprised of various disciplines to ensure that the CDM would provide reasonable assurance that a facility would be built and operated in accordance with the design certification and applicable regulations.

The task groups were composed of various representatives from the technical branches of the staff, depending on the primary area of review by the task group. The task groups were formed based on the following discipline areas: plant systems, reactor systems, electrical, human factors, radiation protection, structural, and instrumentation and controls.

The task groups had primary review responsibilities for systems that were predominantly in their discipline area, and secondary review responsibilities for systems in other discipline areas where appropriate based on the safety significance of the issues. Thus, the groups had overlapping system review responsibilities. All information in the CDM was reviewed by one or more task groups. For example, the reactor core isolation cooling system was reviewed primarily by the Reactor Systems Task Group, and that task group received technical input and comments from the Instrumentation and Controls Task Group. Specialists were designated to provide input to the task groups for selected design issues. Examples of these issues included severe accident issues, testing issues and the ITP, treatment of alarms, displays, and controls, insights from PRA, and functionality of MOVs.

The task groups with primary review responsibility for systems maintained overall responsibility for the reviews of those systems. Overall continuity and consistency of the reviews was maintained through frequent meetings with all task groups and with the projects branch of the staff. Significant policy and technical issues, or issues of concern to multiple task groups, were identified for discussion at periodic senior management meetings between the staff and GE.

The staff developed preliminary draft guidance for use in the reviews of the CDM, and incorporated lessons learned during the course of the reviews into the draft guidance. The draft guidance contained checklists for use in the reviews. The applicability of the issues identified in the checklists to the systems was based on the safety significance of the specific SSCs. The draft guidance also contained standard ITAAC entries that were used to verify selected issues in the appropriate systems of the design. Examples of these standard ITAAC entries are those for the basic configuration of systems, verification of control room and remote shutdown features, and electrical independence. The issues in the checklist and the use of the standard ITAAC entries are discussed in the following sections of this report.

The task groups utilized multiple sources of information to determine the safety-significance of SSCs in the CDM. These sources included the SRP, applicable rules and regulations, GDCs, RGs, USIs and GSIs, NRC generic correspondence, operating experience, NRC inspection programs, facility testing programs, PRA, and insights from ABWR safety and severe accident analyses.

For selected portions of the review, the staff also utilized the regulatory guidance from the Commission related to SECY-90-016, "Evolutionary Light Water Reactor Certification Issues and Their Relationship to Current Regulatory Requirements," as modified by the Commission guidance related to SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor Designs." These issues included staff positions that deviated from or were not embodied in current regulations applicable to the ABWR design. The staff's positions and design-specific requirements are addressed where appropriate in this section of this report, and in the ABWR design certification rule as "applicable regulations."

The staff determined that selected material in the SSAR that supports the CDM, if considered for a change by an applicant or licensee that references the certified ABWR design, would constitute an unreviewed safety question, and therefore, would require NRC review and approval prior to implementation of the change. The material supporting the CDM is discussed where appropriate in this section, in the applicable chapters of this report, and in the ABWR design certification rule.

The task groups utilized a graded approach to the level of detail in its review of the CDM based on the safety-significance of the SSCs. Thus, consistent with the guidance of Part 52 and the SRM related to SECY-90-377, the staff recognized that although many aspects of the design were important to safety, the level of design detail in the CDM and verification of the key features and performance characteristics should be commensurate with the significance of the safety functions to be performed. In addition, the SSAR was reviewed to ensure that the information was consistent with the design description and that the information supporting the Tier 1 material was comprehensive and technically adequate. Thus, the individual task groups reviewed the CDM based on the safety significance of the material, as discussed in the following paragraphs.

14.3.2.3.1 Plant Systems Task Group Review

The Plant Systems Task Group had primary review responsibility for most of the fluid systems in CDM Section 2.0 that were not part of the core reactor systems. The scope of the plant systems review included new and spent fuel handling systems, power generation systems, air systems, cooling water systems, radioactive waste systems and heating, ventilation and air conditioning systems. The group also reviewed selected interface requirements within those systems. The group reviewed issues which affect multiple SSCs such as equipment qualification and protection from fires, floods and tornado missiles, and had secondary review responsibilities for most of the fluid systems and the structures of the design.

The task group primarily utilized the SRP in its review of the CDM to determine the safety significance of SSCs. Other sources included applicable rules and regulations, GDCs, RGs, USIs and GSIs, NRC generic correspondence, PRA, insights from ABWR safety and severe accident analyses, and operating experience. The task group also used the draft review guidance for the design control document as an aid in its review of the systems. For selected portions of the review, the staff also utilized the regulatory guidance from the Commission related to SECY-90-016, "Evolutionary Light Water Reactor Certification Issues and Their Relationship to Current Regulatory Requirements," as modified by the Commission guidance related to SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor Designs." The task group reviewed the Tier 1 submittals (including the design description, figures, and ITAAC) of the design using the guidelines provided in the draft review guidance for the CDM as an aid for establishing consistency and completeness.

The task group reviewed the CDM for treatment of design information proportional to the safety significance of the SSC for that system. Many items were judged to be important to safety, and were thus included in the CDM. The following issues were identified to ensure comprehensive and consistent treatment in the CDM based on the safety significance of the system being reviewed:

(1) System purpose and functions
(2) Location of system
(3) Key design features of the system
(4) Seismic and ASME code classifications
(5) System operation in various modes
(6) Controls, alarms, and displays
(7) Logic
(8) Interlocks
(9) Class 1E electrical power sources and divisions
(10) Equipment to be qualified for harsh environments
(11) Interface requirements
(12) Numeric performance values
(13) Accuracy and quality of figures

Additionally, standard ITAAC entries were utilized to verify selected issues, where appropriate. Examples of these included basic configuration, physical separation, and divisional power supplies. In particular, the general provision for environmental qualification aspects of SSCs invoked by the basic configuration ITAAC was reviewed to ensure appropriate treatment in the CDM.

Environmental qualification (EQ) of safe-shutdown equipment is verified as part of the basic configuration ITAAC for safety-related systems. EQ treatment in the ITAAC is discussed in the General Provisions section of the CDM. Verification includes type tests or a combination of type tests and analyses of Class 1E electrical equipment identified in the Design Description or accompanying figures to show that the equipment can withstand the conditions associated with a design basis accident without loss of safety function for the time that the function is needed.

The task group reviewed integrated plant safety analyses such as fires, floods and missile protection to ensure they were adequately addressed in the CDM. The insights from these analyses that were addressed in the CDM are contained in SSAR Section 14.3. The issues of floods, fires, missiles, pipe failures, and environmental protection are verified by the ITAAC on a system-specific basis, rather than generically. Divisional separation (both physical and electrical) is the primary means of ensuring protection of safety-related equipment from these events. Verification of divisional separation is performed as part of both individual system ITAACs and building ITAACs. Physical and electrical separation is verified in each safety-related system ITAAC and divisional barriers are verified in the reactor and control building ITAACs.

The design features in the CDM were selected to ensure that the integrity of the analyses would be preserved in an as-built facility. For example, 3-hour fire boundaries and divisional separation were shown in the building figures. Also, flooding features such as structure elevations were specified in the site parameters, flood doors were shown on the building figures, and elevations where shown on the buildings to verify that the approximate physical location of components and relative elevations of buildings minimized the effects of flooding. As-built reconciliation reports for fires and floods to ensure consistency with the SSAR analyses are required by the fire protection system ITAAC and selected building ITAAC, respectively.

Other specific issues that were addressed include heat removal capabilities for design-basis accidents and tornado and missile protection. Heat removal capabilities were verified through heat removal requirements for core cooling system heat exchangers and interface requirements for site-specific systems. Tornado and missile protection was provided by inlet and outlet dampers in ventilation systems, and through the structural design of buildings.

The staff was evaluating ACRS comments regarding the need for verification of fires and flooding analyses in the ITAAC for buildings when the advance SER was issued. This was Open Item F14.3.2.3.1-1. GE provided satisfactory resolutions to the ACRS comments in CDM Revision 3 and SSAR Amendment 34. The CDM contained provisions for reconciliation analyses to be conducted for the ABWR design for fires and floods, to ensure that the as-built facility is consistent with the assumptions and analyses for these issues in the design certification. The staff finds this acceptable. This resolved Open Item F14.3.2.3.1-1.

The task group received inputs from other task groups such as the structural, electrical and I&C task groups. The task group also reviewed the ITAAC for consistency with the initial test program described in SSAR Chapter 14. In addition, specialists provided key insights and assumptions from PRA and severe accident analyses, as well as inputs for issues such as treatment of alarms, displays and controls, and functionality of MOVs. A cross-reference from the SSAR to the CDM providing these key insights and assumptions is contained in SSAR Section 19.8.

The issue of containment isolation is addressed by a combination of the system ITAACs and the Leak Detection and Isolation System ITAAC. The containment isolation valves are shown on the system figures. The verification of the design qualification of the motor operated containment isolation valves will be verified by the basic configuration check in the system ITAAC as discussed in the general provisions discussion. In addition, in-situ tests are required for containment isolation MOV and check valves in each system ITAAC. The Leak Detection and Isolation System ITAAC verifies that the containment isolation valves close on receipt of an isolation signal. Actual closure of the containment isolation valves is checked using the manual isolation switches in the main control room (MCR). A separate ITAAC entry verifies that a containment isolation signal is generated for each the process variables that will cause a containment isolation. This precludes multiple cycling of the containment isolation valves during the testing.

The staff decided during the review of the ITAAC that the MCR ITAAC would verify only the minimum inventory as derived from the Emergency Procedure Guidelines, the requirements of RG 1.97, and probabilistic risk assessment insights. Other controls, indications and alarms are identified in the system ITAAC and verified to exist in the MCR. The ability of these controls, indications, and alarms to function will be checked during operation of the system for the functional tests required by the system ITAAC. The operation of the system during the completion of the functional tests required in the system ITAAC will be conducted from the MCR. Therefore, it was decided that the verification that the system can be operated from the MCR need not be a separate ITAAC. The staff also decided that since the operation of the equipment from the control room demonstrates the control function, continuity checks between the remote shutdown panel (RSP) and the equipment demonstrates that the control signal will be received by the component and provides adequate assurance that the equipment can be operated from the RSP. Additionally, the Initial Test Program will adequately cover the technical verifications of the ability to operate plant equipment from the RSP.

14.3.2.3.2 Reactor Systems Task Group Review

The reactor systems task group had primary review responsibility for the reactor systems and core cooling systems in CDM Section 2.0. The group had secondary review responsibilities for those systems that could affect the operation of the reactor and core cooling systems.

The task group primarily utilized the SRP in its review of the CDM to determine the safety significance of SSCs. Other sources included applicable rules and regulations, GDCs, RGs, USIs and GSIs, NRC generic correspondence, PRA, insights from ABWR safety and severe accident analyses, and operating experience. The task group also used guidelines provided in the draft review guidance for the design control document as an aid for establishing consistency and completeness in its review of the systems. For selected portions of the review, the staff also utilized the regulatory guidance from the Commission related to SECY-90-016, "Evolutionary Light Water Reactor Certification Issues and Their Relationship to Current Regulatory Requirements," as modified by the Commission guidance related to SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor Designs."

The task group reviewed the CDM systems in a similar manner as the plant systems task group because the reactor and core cooling systems were primarily fluid systems. Thus, the group examined the systems for comprehensive and consistent treatment of the issues listed in Section 14.3.2.3.1 of this report, based on the safety significance of the respective systems being reviewed. The task group found that many of the systems in this area of review were classified as safety related, and thus many of the characteristics and features of these systems were judged to have safety significance. This is reflected in a higher level of detail in the CDM for these systems.

The task group reviewed the CDM to verify that plant safety analyses, such as for core cooling, transients, overpressure protection, and anticipated transient without scram (ATWS), were adequately addressed. The task group used the tables contained in SSAR Section 14.3 to assess that the important input parameters used in the transient and accident analyses were verified by the ITAAC. The task group also interacted with specialists in PRA and severe accident analyses to ensure important insights and design features from these analyses were incorporated into the CDM. For the severe accident analyses in particular, the basis for the staff's review was the Commission guidance related to SECYs 90-016 and 93-087. For both PRA and severe accident analyses, although large uncertainties and unknowns may have been associated with the event phenomena, design features important for severe accident prevention and mitigation resulting from these analyses were selected for treatment in the CDM. The supporting information regarding the detailed design and analyses remained in the SSAR. For many of the design features, it was impractical to test their functionality. Consequently, the existence of the feature on a figure, subject to a basic configuration walkdown, was considered sufficient CDM treatment.

The staff determined that the detailed supporting information in the SSAR for the nuclear fuel, fuel channel, and control rod CDM, if considered for a change by a COL applicant or licensee that references the certified ABWR design, would constitute an unreviewed safety question. Thus, the staff has concluded that the fuel cycle and control rod design criteria in SSAR Sections 4B and 4C, the first cycle fuel, control rod and core design and the methods used to analyze these components may not be changed without prior NRC review and approval. The specific fuel, control rod, and core designs presented in SSAR Chapter 4 will constitute, based on staff review and approval, an approved design that may be used for the COL first cycle core loading, without further NRC staff review. If any other core design is requested for the first cycle, the COL applicant or licensee will be required to submit for staff review that specific fuel, control rod, and core design analyses as described in SSAR Chapters 6 and 15.

No ITAAC are required for the CDM information in these areas because of the requirement for prior NRC review and approval of any proposed changes to the approved design. Post fuel load testing programs (e.g., startup testing and power ascension testing) verify that the actual core performs in accordance with the analyzed core design.

Examples of the issues that the task group examined for treatment in Tier 1 included net positive suction head for key pumps (standard ITAAC entry specified in the applicable systems), and intersystem LOCA (the design pressure of the piping of the systems that interface with the reactor coolant pressure boundary is specified in the design descriptions of the applicable systems). The task group also reviewed the ITAAC for consistency with the initial test program described in SSAR Chapter 14.

14.3.2.3.3 Electrical Task Group Review

The electrical task group had primary review responsibility for the station electrical systems in CDM Section 2.12. The scope of the ABWR electrical design includes the entire Class 1E portion of the electrical system as well as a major portion of the non-Class 1E electrical system. It also includes portions of the plant lighting system. The group also reviewed selected interface requirements. The group had secondary review responsibilities for some systems using Class 1E power.

In establishing the top level requirements for the electrical design, the staff used the Code of Federal Regulations including the GDC of Appendix A and Parts 50.49, "Environmental Qualification," and 50.63, "Station Blackout," as its main bases. In addition, IEEE nuclear standards were used, as appropriate, to further establish top level requirements. The staff also considered significant lessons learned from operating experience problems and insights gained from the PRA for the ABWR.

> GDC 17, in part, requires that an onsite and an offsite electric power system be provided to permit functioning of structures, systems and components important to safety. It further requires that the onsite electric power system have independence and redundancy and the electric power supplied by the offsite system be supplied by two physically independent circuits.
>
> 10 CFR 50.49 requires that certain electrical equipment be qualified for accident (referred to as harsh) environments.
>
> 10 CFR 50.63 requires that a nuclear power plant be able to withstand and recover from a station blackout event.
>
> IEEE 308 "IEEE Standard Criteria for Class 1E power Systems for Nuclear Power Generating Stations," in conjunction with other related IEEE standards, establish specific design criteria for nuclear power plant electrical systems and equipment.

The staff's review of the ABWR standard plant was conducted to ensure, in part, that the certified design contains top level design, fabrication, testing, and performance requirements for SSCs important to safety. Design descriptions and ITAAC were established to verify that these top level (Tier 1) requirements (or design commitments) are met when the plant is built.

<u>Class 1E Electrical Systems</u>

The ABWR Class 1E electrical systems include: (1) the Class 1E electrical power distribution system, (2) the emergency diesel generators, (3) the Class 1E direct current power supply, and (4) the Class 1E vital ac and Class 1E instrument and control power supplies. Using the above regulations, IEEE standards, operating experience, and PRA as its bases, GE established top-level design commitments for the Class 1E electrical systems of the ABWR to be included in the design descriptions and verified by ITAAC.

The top-level design commitments for the Class 1E electrical systems include design aspects related to:

(1) Equipment qualification for seismic and harsh environment
(2) Redundancy and independence
(3) Capacity and capability
(4) Electrical protection features
(5) Displays/controls/alarms

Equipment Qualification

To ensure that the seismic design requirements of GDC 2 and the environmental qualification requirements of 10 CFR 50.49 have been adequately addressed, "basis configuration" ITAAC were established for applicable systems to verify these design aspects of electrical equipment important to safety.

The design description identifies that Class 1E equipment is seismic Category 1 and equipment located in a harsh environment is qualified. A "basic configuration" ITAAC was developed to include these areas.

Redundancy and Independence

To ensure that the Class 1E electric systems meet the single failure requirements of GDC 17 (and other GDC), ITAAC were established to verify the redundancy and independence of the Class 1E portion of the electrical design.

For the electrical systems, ITAAC verified the Class 1E divisional assignments and independence of electric power by both inspections and tests. The independence is established by both electrical isolation and physical separation. Identification of the Class 1E divisional equipment is included to aid in demonstrating the separation. (The detailed requirements are specified in the SSAR. For example, separation distances and identification are outlined in the SSAR.) These attributes are verified all the way to the electrically powered loads by a combination of the electrical system ITAAC and the ITAAC of the individual fluid, I&C, and HVAC systems which also cover the electrical independence and divisional power supply requirements.

Capacity and Capability

To ensure that the electrical systems have the capacity and capability to supply the safety-related electrical loads, ITAAC were established to verify the adequate sizing of the electrical system equipment and its ability to respond (e.g., automatically in the times needed to support the accident analyses) to postulated events. This includes the Class 1E portion and the non-Class 1E portion to the extent that it is involved in supporting the Class 1E system.

ITAAC are included to analyze the as-built electrical system and installed equipment (diesel generators, transformers, switchgear, batteries, etc.) to verify its ability to power the loads. In addition, the ITAAC also includes tests to demonstrate the operation of the equipment.

To ensure that the Class 1E portions of the electrical power system have the capability to respond to postulated events including LOCA, loss of normal preferred power, and degraded voltage conditions, ITAAC were established to verify the initiation of the Class 1E equipment necessary to mitigate the event.

ITAAC are included to analyze the as-built electrical power system for its response to a LOCA, loss of voltage, combinations of LOCA and loss of voltage, and degraded voltage. In addition, tests are included to demonstrate the actuation of the electrical equipment in response to postulated events.

Electrical Protection Features

To ensure that the electrical power system is protected against potential electrical faults, ITAAC were established to verify the adequacy of the electrical circuit protection included in the design. Operating experience and NRC Electrical Distribution System Functional Inspections (EDSFIs) have indicated some problems with the short circuit rating of some electrical equipment and breaker and protective device coordination.

ITAAC are included to analyze the as-built electrical system equipment for its ability to withstand and clear electrical faults. ITAAC are also included to analyze the protection feature coordination to verify its ability to limit the loss of equipment due to postulated faults.

Displays, Alarms and Controls

To help ensure that the electrical power system is available when required, ITAAC are included to verify the existence of monitoring and controls for the electrical equipment. The minimum set of displays, alarms, and controls is based on the emergency procedure guidelines. In some cases, additional displays, alarms, and controls are specified based on special considerations in the design and/or operating experience.

ITAAC are included to inspect for the ability to retrieve the information (displays and alarms), and to control the electrical power system in the main control room and/or at locations provided for remote shutdown.

Other Electrical Equipment Important to Safety

In addition to the Class 1E systems addressed above, other aspects of the electrical design were deemed to be important to safety and the top-level design commitments were included in the CDM.

Offsite Power

To ensure that the requirements of GDC 17 for the adequacy and independence of the preferred offsite power sources within the ABWR scope were met, ITAAC were developed to verify the capacity and capability of the offsite sources to feed the Class 1E divisions, and the independence of those sources.

ITAAC are included to inspect the direct connection of the offsite sources to the Class 1E divisions and to inspect for the independence/separation of the offsite sources. Lightning protection and grounding features are inspected as part of the configuration ITAAC.

In addition, the design description includes "interface" requirements for the portions of the offsite power outside of the ABWR scope, however no ITAAC are included for the interfaces. The interfaces define the requirements that the offsite portion of the design (that is out-of-scope) must meet to support and not degrade the in-scope design.

Containment Electrical Penetrations

To ensure the containment electrical penetrations (both those containing Class 1E circuits and those containing Non Class 1E circuits) do not fail due to electrical faults and potentially breach the containment, ITAAC were developed to verify that all electrical containment penetrations are protected against postulated currents greater than their continuous current rating.

ITAAC are included to inspect and analyze the electrical penetrations and their protection.

Combustion Turbine Generator

To ensure the availability of the combustion turbine generator (CTG) as an alternate AC source for station blackout events, ITAAC were developed to verify its inclusion in the design and its independence from other AC sources. In addition, the PRA has indicated that the CTG is relatively important from a risk perspective.

ITAAC are included to inspect and test the CTG and its auxiliaries.

Lighting

To ensure that portions of the plant lighting remain available during power failures, ITAAC were developed to verify the continuity of power sources for the lighting systems.

ITAAC are included to inspect and test the lighting and its power sources.

Electrical Power For Non-Safety Plant Systems

To ensure that electrical power is provided to support the non-safety plant systems, Design Descriptions cover portions of the non-Class 1E electrical systems. A basic configuration ITAAC verifies the functional arrangement and the Tier 1 design commitments for these areas.

14.3.2.3.4 Human Factors Task Group Review

The Human Factors task group had primary review responsibility for the main control panel, remote shutdown panel, and local control panels, described in CDM Section 2.0. The group also reviewed CDM Section 3.1, "Human Factors Engineering." CDM Section 3.1 is discussed further in Section 14.3.3.1 of this report. The task group provided input to other task groups on the minimum inventory of alarms, controls, and indications for the control room and the RSS.

The basis for the task group's review in this area was a human factors engineering (HFE) program review model (PRM) developed by the staff. The staff's certification review in the control room design area was based on a design and implementation process plan. The staff informed the Commission of the development of the DAC in this area in SECY-92-299, "Development of Design Acceptance Criteria (DAC) for the Advanced Boiling Water Reactor (ABWR) in the Areas of Instrumentation and Controls (I&C) and Control Room Design," dated August 27, 1992. In addition, the task group utilized the SRP in its review of the CDM. Other sources included applicable rules and regulations, RGs, USIs and GSIs, and operating experience.

The staff developed the HFE PRM, contained in Appendix J of this report, to serve as a technical basis for the review of the design process and DAC proposed by GE for certification of the ABWR control room and remote shutdown station design. The HFE PRM is (1) based upon currently accepted HFE practices, (2) well-defined, and (3) validated through experience with the development of complex, high-reliability systems in other industrial and military applications. The review model identifies the important HFE elements in a system development, design, and evaluation process that are necessary and sufficient requisites to successful integration of human factors in complex systems. The review model also identifies aspects of each HFE element that are key to a safety review, and describes acceptance criteria by which the HFE elements can be evaluated. The HFE PRM has eight program elements, each of which contain both general and more specific acceptance criteria.

Part 52 requires applicants for design certification to meet the TMI requirements in 10 CFR 50.34(f)(2)(iii) for providing a control room design that reflects state-of-the-art human factors principles. GE did not develop a final control room and RSS design before design certification because this is an area of rapidly changing technology. Instead, GE provided the processes and acceptance criteria in CDM Section 3.1 and the detailed supporting information in SSAR Chapter 18 by which the details of the design in this area would be developed, designed, and evaluated. In lieu of having a completed control room design for review, the staff concluded that it could make its safety determination if GE submitted for certification an acceptable process for the design of the control room. In addition, GE must have submitted a description of a minimum inventory of displays, controls, and alarms necessary to accomplish the emergency procedure guidelines (EPGs) and critical operator actions identified through GE's PRA analysis.

The processes and design acceptance criteria in CDM Section 3.1, "Human Factors Engineering," apply to the human factors design of the control room and the RSS systems of the ABWR design. The detailed supporting information for the human factors aspects of the ABWR control room and RSS design are provided in SSAR Chapter 18, "Human Factors," and together with the associated DAC in CDM Section 3.1, are evaluated in Chapter 18 of this report. GE provided amplifying information regarding the processes and CDM selection criteria in this area in SSAR Section 14.3.3.1. The implementation of the process and the design is the responsibility of the COL applicant or licensee.

The staff requested that the minimum inventory of displays, controls, and alarms be developed through a task analysis of the operator actions necessary to carry out the EPGs and PRA critical actions. The staff's evaluation of the resulting minimum inventory encompassed a multidisciplinary effort consisting of human factors, I&C, PRA, and plant, reactor, and electrical system engineering. The criteria used to determine acceptability of the inventory included assuring that: (1) the scope of these items in the EPGs and PRA effort were adequately considered, (2) the task analysis was detailed and comprehensive, (3) RG 1.97, category I variables for accident monitoring were included, and (4) important system displays and controls described in the Tier 1 system design descriptions necessary for transient mitigation were included.

The minimum inventory list for the control room was included in the CDM Section 2.7.1, "Main Control Room Panels." The controls and indicators required on systems to remotely shutdown the reactor are contained in CDM Section 2.2.6, "Remote Shutdown System." The items required for operation of the RSS are shown with an "R" on the figures for the individual systems. Detailed supporting information is contained in Chapter 7 of the SSAR. The individual systems that contained the sensors for the displays, controls, and alarms were reviewed to ensure that standard ITAAC entries were used to verify their function. The design processes and acceptance criteria specified in the DAC for I&C equipment contained in CDM Section 3.4, particularly the verification and validation aspects of the I&C DAC, will verify proper operation of the I&C aspects of the equipment. Similarly, the design processes and DAC for HFE contained in CDM Section 3.1, particularly the verification and validation aspects of the HFE DAC, will verify proper design of the equipment for human factors aspects.

14.3.2.3.5 Radiation Protection Task Group Review

The Radiation Protection Task Group had primary review responsibility for the area radiation monitoring system, containment atmospheric monitoring system, and emergency response facilities in CDM Section 2.0; the additional material in CDM Section 3.2, "Radiation Protection," applicable to multiple systems of the design; and selected site parameters. CDM Section 3.2 is discussed further in Section 14.3.3.2 of this report. The group had secondary review responsibility for all other ITAACs which addressed the plant radiation protection design or systems relied upon in the design-basis accidents (DBAs) dose assessment. These ITAACs included buildings, ventilation and filtration systems, primary containment, drywell bypass, post-accident sampling system, and site parameters (atmospheric dispersion).

The task group primarily utilized the SRP in its review of the CDM to determine the safety significance of SSCs. Other sources included applicable rules and regulations, GDCs, RGs, USIs and GSIs, NRC generic correspondence, and operating experience. The task group also used the draft review guidance for the design control document as an aid for consistency in its review of the systems.

The task group relied heavily on the material in CDM Section 3.2 during its review of the design. The design processes and acceptance criteria in this section were developed because GE did not provide sufficient information to stipulate the source terms needed to verify the design of the shielding, ventilation, and airborne radioactivity monitoring systems. Therefore, GE extracted the most important acceptance criteria for these design features from Chapter 12 of the SRP and put them into the DAC in CDM Section 3.2. A COL applicant or licensee must meet these criteria in the design of the plant, and the staff can audit the facility's design documentation to ensure that the criteria are met. The DAC are general criteria which apply to the design of shielding and ventilation systems throughout the plant. Therefore, there are no references to the DAC in the ITAACs for the buildings and systems.

The group reviewed the ITAAC for the area radiation monitoring system to ensure that the system provides information on radiation dose rates in the plant during normal operation and accidents and provides alarms to warn plant personnel of changes in those dose rates. The group reviewed the ITAAC for the containment atmospheric monitoring system to ensure that the system provides information on radiation dose rates and gas concentrations during accidents and provides alarms to warn plant personnel of high levels of these parameters. The group reviewed the ITAAC for emergency response facilities to ensure that adequate facilities are provided for the technical support center (TSC) and operational support center including space, data retrieval and communications equipment, and a ventilation system to provide radiation protection.

The group reviewed several ITAAC for which the group had secondary review responsibility. The review of these ITAAC was focused on verifying design features and assumptions upon which the radiological dose consequence assessment of the design basis accidents (DBAs) in this SER is based. The following discussion provides examples of some of the important design features and assumptions that are addressed in the CDM. The maximum MSIV closure time and maximum MSIV leakage rates will be verified by the ITAAC for the nuclear boiler system. The maximum primary containment leakage rate will be verified by the ITAAC for the primary containment system. The minimum radioiodine removal efficiency of the charcoal adsorbers in the standby gas treatment system (SGTS) filter trains and the maximum time for the SGTS to draw a specified negative pressure in the secondary containment will be verified by the ITAAC for the SGTS. The minimum radioiodine removal efficiency of the charcoal adsorbers in the control room and TSC ventilation system filter trains will be verified by the ITAAC for the HVAC systems. Capability of the main steam system to maintain structural integrity in an safe-shutdown earthquake (SSE) will be verified by the ITAAC for the turbine main steam system. Capability of the off-gas system to withstand an internal hydrogen explosion will be verified by the ITAAC for the off-gas system. In addition, the meteorological dispersion values assumed in the accident analyses were identified as bounding parameters for a site in CDM Table 5.0, ABWR Site Parameters. Also, the radiological analysis table in SSAR Section 14.3 was used to ensure that GE had addressed in the CDM the most important, though not necessarily all, of the key parameters in the accident dose analyses.

14.3.2.3.6 Structural Task Group Review

The Structural Task Group had primary review responsibility for building structures, chemical engineering systems, site parameters, piping DAC, reactor pressure vessel (RPV) system, and the legend for figures. The piping DAC contained in CDM Section 3.3 is discussed in Section 14.3.3.3 of this report. The task group had secondary review responsibilities for other systems as they related to MOVs, check valves, hydrostatic tests, and seismic and safety classification of systems, and for other structural aspects of systems. The task group was composed of reviewers with experience in structural, mechanical, materials, and chemical engineering. In addition, the task group was augmented by a specialist in MOVs, check valves, and pumps, a specialist in seismic and safety classification, and a specialist in chemical engineering.

The task group primarily utilized rules and regulations to review the top level commitments in the CDM. Other sources included RGs, SRP guidelines, and PRA insights from ABWR safety and severe accident analyses and operating experience. For selected portions of the review, the staff also adhered to policy discussions by the Commission in the SRM related to SECY-90-016, "Evolutionary Light Water Reactor Certification Issues and Their Relationship to Current Regulatory Requirements," as modified by the Commission guidance in the SRM related to SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor Designs." In addition, the task group reviewed the Tier 1 submittals (including the design description, figures, and ITAAC) of the design using the guidelines provided in the draft review guidance for the CDM as an aid for establishing consistency and completeness.

The task group reviewed the design description for those assigned systems to ensure that the certified design was consistent with the NRC regulations and policy decisions as discussed in SECY-93-087. The task group reviewed the Tier 1 material to assess whether a conclusion could be reached that the ITAAC were necessary and sufficient to provide reasonable assurance that the facility has been constructed and will be operated in conformity with the license, the provisions of the Atomic Energy Act, and the Commission's rules and regulations.

The task group reviewed the design descriptions, figures, ITAAC, and the SSAR for consistency. The task group reviewed the CDM for all the systems to ensure consistency with the seismic and safety classification described in Section 3.2 of the SSAR. The task group ensured that the seismic classification of the system as described in the design description and the ASME Code Class boundaries of the system as depicted on the figures was consistent with SSAR Section 3.2.

The task group reviewed the ITAAC for consistency with the preoperational tests specified in Chapter 14 of the SSAR. The task group reviewed the tests identified in the ITAAC tables to determine whether those tests have been appropriately included in SSAR Chapter 14 and also whether the preoperational tests have been adequately incorporated into ITAAC. The task group also reviewed tests in Chapter 14 of the SSAR or in the ITAAC that would require an analysis to convert preoperational test conditions to accident conditions to ensure that the methodology for performing the analysis was specified adequately. In addition, the task group reviewed all systems ITAAC to ensure that selected issues were adequately and consistently treated in the CDM through the use of the standard ITAAC entries for basic configuration, hydrostatic test, MOVs, and check valves.

The task group used the following general approach in reviewing the design descriptions, figures, and ITAAC and for establishing what information should reside in each tier. The certified design (design description) should contain top level design, fabrication, testing, and performance requirements for SSCs important to safety. ITAAC are established, in part, to verify that these top-level (Tier 1) design, fabrication, testing and performance requirements are met when the plant is built.

Although the establishment of what specific information was to be included in the design description was essentially a matter of judgement, the draft review guidance provided some guidance for consistency in certain areas regarding what information should be in which tier as well as whether an inspection, test or analysis was required to be performed. The draft review guidance also provided a basis for the staff's judgement in selecting which tier the information should reside and why an ITAAC was deemed necessary. These areas include component welding, equipment seismic qualification, pumps, valves, and piping systems. The basis for selecting these areas included its importance to safety as well as its past experience with construction and operating problems.

Design descriptions and ITAAC were developed and grouped by systems and building structures. These Tier 1 requirements for systems and building structures are typically verified by inspections, tests, and analyses specified in the system ITAAC. For example, system-specific performance tests are typically conducted to demonstrate that the system can perform its intended function. For building structures, the structural capability is typically verified by performing an analysis to reconcile the as-built data with the structural design bases for each safety-related building.

For components, the verification of design, fabrication, testing, and performance requirements are partially addressed in conjunction with the specific system ITAAC. For example, a test is typically performed to verify the ability of a motor-operated valve to close under design-basis fluid conditions. However, performance tests are not practical for verifying certain component design requirements such as its seismic design or safety classification. Therefore, ITAAC have been developed to verify certain areas where performance tests are not practical. These areas include seismic design qualification and fabrication (i.e., welding) of components. The ITAAC for seismic design qualification and fabrication of components are established on a generic basis in the general provisions for verifying the basic configurations of systems rather than on an individual component basis.

The Tier 1 treatment of the design qualification and fabrication of components was reviewed to ensure that the issues were verified by ITAAC as discussed below:

(1) Fabrication of Components

A basic configuration check (system) is required in each individual system ITAAC. The configuration check includes an inspection of the welding quality for all ASME Code Class 1, 2, and 3 piping systems. A hydrotest is also required in each system ITAAC for ASME Code Class 1, 2, and 3 piping systems to verify that, in the process of fabricating the overall piping system, the welding and bolting requirements for ensuring the pressure integrity have been met. The methods to be used by the COL applicant or licensee to verify the acceptability of the welds are discussed in the SSAR in the sections applicable to the specific component or structure.

(2) Design Qualification of Components

(a) Safety Classification - The safety classification of SSCs are described in each system's design description. The functional drawings identify the boundaries of the ASME Code classification that are applicable to the safety class. The piping DAC includes a verification of the design report to ensure that the appropriate code design requirements for the system's safety class have been implemented.

(b) Mechanical and Electrical Equipment (including I&C) - A basic configuration check (system) is required in each individual system ITAAC. The configuration check includes an inspection of the as-built equipment (including anchorages) and a review of the qualification records to verify that the equipment in its as-built condition is seismically qualified. The material in SSAR

Section 3.10.1 provides detailed supporting information for the CDM regarding the methods to be used by the COL applicant or licensee for the dynamic qualification of equipment. This material, if considered for a change by an applicant or licensee that references the certified ABWR design, would constitute an unreviewed safety question, and therefore, would require NRC review and approval prior to implementation of the change. This material supporting the CDM is discussed further in Section 3.10 of this report.

(c) Valves - The verification of the design qualification of valves is performed in conjunction with the basic configuration check for mechanical equipment as discussed above. Specifically, for MOVs, a special inspection is required as a part of the basic configuration check to verify the records of vendor tests that demonstrate the ability of MOVs to function under design conditions. In addition, in-situ tests are required for MOVs and check valves in each system ITAAC. These tests will be performed during the initial test program. The material in SSAR Section 3.9.6.2.2 provides detailed supporting information for the CDM regarding the methods to be used by the COL applicant or licensee for the design, qualification, and testing of MOVs to demonstrate their design basis capability. This material, if considered for a change by an applicant or licensee that references the certified ABWR design, would constitute an unreviewed safety question, and therefore, would require NRC review and approval prior to implementation of the change. This material supporting the CDM is discussed further in Section 3.9.6 of this report.

(d) Piping - The verification of the overall piping design including the effects of high-energy line breaks and the application of leak-before-break (as applicable) is performed in conjunction with the piping DAC. The as-built piping system is required to be reconciled with the design commitments. The material in SSAR Section 3.12 provides detailed supporting information for the CDM regarding the analysis methods and design criteria to be used by the COL applicant or licensee to complete the piping design. This material, if considered for a change by an applicant or licensee that references the certified ABWR design, would constitute an unreviewed safety question, and therefore, would require NRC review and approval prior to implementation of the change. This material supporting the CDM is discussed further in Section 3.12 of this report.

Review of the ABWR Structural Design Integrity

The scope of structural design covers the major structural systems in the ABWR plant including the RPV, ASME Code Class 1, 2, and 3 piping systems, and major building structures (primary containment, reactor building, control building, turbine building, service building, and radwaste building). The RPV, piping systems, and primary containment are included because they provide the defense-in-depth principle for nuclear plants. The major building structures house those systems and components that are important to safety.

In establishing the top level requirements for structural design, the staff used the General Design Criteria (GDC) of 10 CFR Part 50, Appendix A, as its basis. The primary general design criteria pertaining to the major structural system design are GDC 1, "Quality Standards and Records," GDC 2, "Design Bases for the Protection Against Natural Phenomena," GDC 4, "Environmental and Dynamic Effects Design Basis," GDC 14, "Reactor Coolant Pressure Boundary," GDC 16, "Containment Design," and GDC 50, "Containment Design Basis."

GDC 1 requires, in part, the need for structures, systems and components important to safety to be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed.

GDC 2 requires, in part, the need to design structures, systems, and components important to safety to withstand the effects of natural phenomena such as earthquakes, tornados, hurricanes, and floods without loss of capability to perform their safety functions, including the appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena.

GDC 4 requires, in part, the need to protect structures, systems, and components important to safety from dynamic effects including the effects of missiles, pipe whipping, and discharging fluids that may result from equipment failures and from events and conditions outside the nuclear power unit.

GDC 14 requires, in part, the need for the reactor coolant pressure boundary to be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.

GDC 16 requires, in part, the need for the reactor containment to provide an essentially leak-tight barrier against uncontrolled release of radioactivity to the environment.

GDC 50 requires, in part, the need for the reactor containment structure including access openings and penetrations to be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any loss-of-coolant accident.

Using the above GDC as its basis, the staff established the following top- level attributes to be verified by ITAAC:

(1) pressure boundary integrity (GDC 14, 16 and 50)
(2) normal loads (GDC 2)
(3) seismic loads (GDC 2)
(4) suppression pool hydrodynamic loads (GDC 4)
(5) flood, wind, and tornado (GDC 2)
(6) rain and snow (GDC 2)
(7) pipe rupture (GDC 4)
(8) codes and standards (GDC 1)

In addition, to ensure that the final as-built plant conforms to the certified design, GE provided ITAAC to reconcile the as-built plant with the structural design basis. A summary of the top-level structural design requirements for the major structural systems that are verified by the structures and systems in CDM Section 2.0 and the piping design information in CDM Section 3.3 is provided below.

Pressure Boundary Integrity

To ensure that the applicable requirements of GDC 14, 16, and 50 have been adequately addressed, ITAAC were established to verify the pressure boundary integrity of the RPV, piping, and primary containment for the ABWR. GDC 16 and 50 apply to the primary containment and GDC 14 applies to the RPV and the reactor coolant pressure boundary piping systems. The pressure integrity for these major structural systems are needed to ensure the defense-in-depth principle.

For the RPV and piping, hydrostatic tests performed in conjunction with the ASME Boiler and Pressure Vessel Code, Section III are required by ITAAC. For the primary containment, a structural integrity test is required by ITAAC to be performed on the pressure boundary components of the primary containment in accordance with the ASME Boiler and Pressure Vessel Code, Section III. Because the requirements of GDC 14, 16, and 50 do not apply to the reactor, control, turbine, service, and radwaste buildings, ITAAC were not required to verify the pressure integrity for these other buildings.

Normal Loads

To ensure that the applicable requirements of GDC 2 have been adequately addressed, ITAAC were established to verify that the normal and accident loads have been appropriately combined with the effects of natural phenomena.

For piping systems, ITAAC require an analysis to reconcile the as-built piping design with the design-basis loads (which include the appropriate combination of normal and accident loads). For the RPV, the fabrication is performed primarily in the vendor's shop where adherence to design drawings is tightly controlled. Therefore, ITAAC for the as-built reconciliation of normal loads with accident loads for the RPV were deemed to be inappropriate. Instead, ITAAC verify that the ASME Code-required reports exist to document that the RPV has been designed, fabricated, inspected, and tested to Code requirements to ensure adequate safety margin.

Similarly, for safety-related buildings, ITAAC require an analysis for reconciling the as-built plant with the structural design basis loads (which include the combination of normal and accident loads with the effects of natural phenomena). The analysis results are to be documented in a structural analysis report, the scope and contents of which are described in the SSAR. The staff determined that the design of certain structures did not require verification by ITAAC, based on their safety significance. In particular, these ITAAC apply only to safety-related structures and are not applicable to the service and turbine buildings.

Seismic Loads

To ensure that the applicable requirements of GDC 2 have been adequately addressed, ITAAC were established to verify that the safety-related systems and structures have been designed to seismic loadings. Component qualification for seismic loads is addressed by ITAAC that were established for verifying the basic configuration of systems.

As discussed above for normal loads on piping systems and the RPV, ITAAC require an analysis to reconcile the as-built piping design with the design basis loads (which include seismic loads). For the RPV, ITAAC for the as-built reconciliation of seismic loads for the RPV were deemed to be inappropriate as previously discussed. Instead, ITAAC verify that the ASME Code-required reports exist for the RPV ensuring that the RPV has been designed, fabricated, inspected, and tested to ASME Code requirements.

For safety-related buildings, ITAAC require an analysis for reconciling the as-built plant with the structural design-basis loads (which include seismic loads). The analysis results are to be documented in a structural analysis report, as discussed above. These ITAAC apply only to safety-related structures and are not applicable to the service and turbine buildings. However, because the leakage path for fission products includes components within the turbine building, the turbine building is required to withstand the effects of a safe-shutdown earthquake. Therefore, ITAAC were established to verify that, under seismic loads, the collapse of the turbine building will not impair the safety-related functions of any structures or equipment located adjacent to or within the turbine building.

For non-seismic Category I SSCs, the need for ITAAC to verify that their failure will not impair the ability of near-by safety-related SSCs to perform their safety-related functions was assessed. Because the design detail and as-built and as-procured information for many non-safety-related systems (e.g., field-run piping and balance-of-plant systems) are not required for design certification and the spatial relationship between such systems and seismic Category I SSCs cannot be established until after the as-built design information is available, the non-seismic to seismic (II/I) interaction cannot be evaluated until the plant has been constructed. Accordingly, the design criteria for ensuring acceptable II/I interactions and a commitment for the COL applicant to describe the process for completion of the design of balance-of-plant and non-safety related systems to minimize II/I interactions and proposed procedures for an inspection of the as-built plant for II/I interactions have been specified as a COL action item in the SSAR.

Suppression Pool Hydrodynamic Loads

To ensure that the applicable requirements of GDC 4 have been adequately addressed, ITAAC were established to verify that the safety-related systems and structures have been designed to suppression pool hydrodynamic loadings, which include safety relief valve discharge and loss-of-coolant accident (LOCA) loadings. Component qualification for suppression pool hydrodynamic loads is addressed by ITAAC established for verifying the basic configuration of systems.

As discussed above for seismic loads on piping systems and the RPV, ITAAC require an analysis to reconcile the as-built piping design with the design- basis loads (which include suppression pool hydrodynamic loads). For the RPV, ITAAC verify that the ASME Code-required reports exist to ensure that the RPV has been designed, fabricated, inspected, and tested to ASME Code requirements.

For the reactor building and primary containment including the internal structures, ITAAC require an analysis for reconciling the building as-built configuration with the structural design basis loads (which include suppression pool hydrodynamic loads). The as-built analysis results are to be documented in a structural analysis report as discussed above. The effects of suppression pool hydrodynamic loads do not extend beyond the reactor building, and, thus, ITAAC are not required to verify these loadings for the other ABWR building structures.

ITAAC also require the verification of the horizontal vent system, water volume, and the safety-relief valve discharge line quencher arrangement to ensure adequacy of the suppression pool hydrodynamic loads used for design.

Flood, Wind, Tornado, Rain, and Snow

To ensure that the applicable requirements of GDC 2 have been adequately addressed, ITAAC were established to verify that the safety-related systems and structures have been designed to withstand the effects of natural phenomena other than those associated with seismic loadings. The effects include those associated with flood, wind, tornado, rain, and snow.

These loadings do not apply to the RPV, the ASME Code Class 1, 2, and 3 piping systems and components, nor the primary containment because they are all housed within the safety-related buildings. For safety-related buildings, ITAAC require an analysis for reconciling the as-built plant with the structural design basis loads (which include the flood, wind, tornado, rain, and snow loads). Based on their safety significance, these ITAAC apply only to safety-related structures and are not applicable to the service and turbine buildings.

For flooding, ITAAC also require inspections to verify that water-tight doors exist, penetrations (except for water-tight doors) in the divisional walls are at least 2.5 m above the floor, and safety-related electrical, instrumentation, and control equipment are located at least 20 cm above the floor surface. In addition, for safety-related buildings, ITAAC require that external walls below flood level are equal to or greater than 0.6 m to protect against water seepage, and penetrations in the external walls below flood level are provided with flood protection features.

Pipe Break

To ensure that the applicable requirements of GDC 4 have been adequately addressed, ITAAC were established to verify that the safety-related SSCs have been designed to the dynamic effects of pipe breaks. Component qualification for the dynamic effects of pipe breaks is addressed by ITAAC established for verifying the basic configuration of systems.

For the RPV, ITAAC that verify the basic configuration of the RPV system require an inspection of the critical locations that establish the bounding loads in the LOCA analyses for the RPV to ensure that the as-built areas not exceed the postulated break areas assumed in the LOCA analyses.

In addition, ITAAC have been established to verify by inspections of as-built, high-energy pipe break mitigation features and of the pipe break analysis report that safety-related SSCs be protected against the dynamic and environmental effects associated with postulated high-energy pipe breaks. ITAAC to verify pipe break loads are not required for the turbine, service, and radwaste buildings either because they are not safety-related structures or there are no high-energy lines located within the structure.

Codes and Standards

To ensure that the applicable requirements of GDC 1 have been adequately addressed, ITAAC were established to verify that appropriate codes and standards were used in the design and construction of safety-related systems and components. In general, the staff considered those codes and standards endorsed by the regulations under 10 CFR 50.55a in determining which codes and standards were appropriate for Tier 1 verification. The ASME Boiler and Pressure Vessel Code, Section III for Code Class 1, 2, and 3 systems and components was established as the code for the design and construction of ABWR piping systems and the RPV. For safety-related building designs, the staff based its safety findings on audits of ABWR design calculations which relied on specific codes and standards. These codes and standards are contained in SSAR Sections 3.8.1, 3.8.2, 3.8.3, 3.8.4, and 3.8.5, and were identified in Section 3.8 of this report as material that, if considered for a change by an applicant or licensee that references the certified ABWR design, would constitute an unreviewed safety question, and therefore, would require NRC review and approval prior to implementation of the change.

Inspections will be conducted as a part of ITAAC to verify that ASME Code-required documents exist that demonstrate that the RPV, piping systems and containment pressure boundaries have been designed and constructed to their appropriate Code requirements. For other ASME Code components and equipment, the verification of Code compliance will be performed in conjunction with the quality assurance programs and by the authorized inspection agency as required by the ASME Boiler and Pressure Vessel Code.

As-built Reconciliation

To ensure that the final as-built plant structures are built in accordance with the certified design as required by 10 CFR Part 52, structural analyses will be performed which reconcile the as-built configuration of the plant structures with the structural design bases of the certified design. The structural analyses will be documented in structural analysis reports. Structural analysis reports will be verified in conjunction with ITAAC for the primary containment and the reactor, control, radwaste, and turbine buildings. The detailed supporting information on what is required for an acceptable analysis report is contained in SSAR Chapter 3.

Similarly for piping systems, an as-built analysis will be performed using the as-designed and as-built information. ITAAC will verify the existence of acceptable final as-built piping stress reports that conclude the as-built piping systems are adequately designed.

For the RPV, the key dimensions of the RPV system will be verified in conjunction with the basic configuration check of the system. The key dimensions of the RPV system and the acceptable variations of the key dimensions are provided in the certified design description.

For component qualification, tests, analyses, or a combination of tests and analyses will be performed for seismic Category I mechanical and electrical equipment (including connected instrumentation and controls) to demonstrate that the as-built equipment and associated anchorages are qualified to withstand design basis dynamic loads without loss of safety function. These test and analyses will be performed as a part of ITAAC to verify the basic configuration of the system in which the equipment is located.

14.3.2.3.7 Instrumentation and Controls (I&C) Task Group Review

The I&C Task Group's primary review responsibilities included a review of the CDM for I&C systems involving core protection and control, other miscellaneous I&C

systems, the additional I&C material in CDM Section 3.4 applicable to multiple systems of the design, and selected interface requirements. The material in CDM Section 3.4 is discussed further in Section 14.3.3.4 of this report. The group's secondary review responsibilities included ESF systems, reactivity control systems, and other systems using I&C equipment.

The figures in CDM Section 3.4 depict both safety-related and non-safety-related systems of the design. The block concept was used for developing the system control interface diagrams that were needed for depicting the configuration of the I&C system architecture. The I&C design described in the SSAR was to the level of control functional blocks, and therefore, the configuration in the CDM was to the same level.

The CDM entries were reviewed to confirm that the safety-related I&C system met the protection systems requirements of 10 CFR 50.55a(h), as well as the quality standards and records requirements of GDC 1, the protection against natural phenomenon requirements of GDC 2, the environmental and dynamic effects requirements of GDC 4, the instrumentation and control requirements of GDC 13, the control room requirements of GDC 19, the protection system design requirements of GDC 20, the protection system reliability and testability requirements of GDC 21, the protection system independence requirements of GDC 22, the protection system failure modes requirements of GDC 23, the protection system requirements for reactivity control malfunctions of GDC 25, and the protection against anticipated operational occurrences requirements of GDC 29. To meet the criteria of 10 CFR 50.55a(h), "Criteria for Protection Systems for Nuclear Generating Stations," and IEEE Standard 279-1971, the ITAAC entries were reviewed considering the following design issues:

(1) General functional requirements for the system

(2) Single failure criterion

(3) Quality of components and modules (hardware and software)

(4) Equipment qualification

(5) Channel integrity and channel independence

(6) Classification of equipment

(7) Isolation devices

(8) Single random failure

(9) System inputs

(10) Capability for sensor checks, tests and calibration

(11) Channel bypasses, operating bypasses, indication of bypasses, and access to means for bypassing

(12) Completion of protective action once initiated

(13) Manual initiation

(14) Information read-out

(15) Identification

Standard ITAAC entries for several attributes of the I&C system were developed and used for basic configuration, divisionalized power supply, electrical isolation and physical separation (independence), and control room and remote shutdown system configuration. For those systems reviewed that were not safety-related systems, appropriate criteria from the SRP applicable to those systems were used.

For the microprocessor and digital control technology aspects of the I&C system design of the ABWR, GE did not provide complete design information in the SSAR. This was because the technology in this area is rapidly evolving and it is, therefore, important that the certified design description and ITAAC not "lock in" a design which could be obsolete at the time of construction. The process to complete the design, with appropriate acceptance criteria, is specified in CDM Section 3.4, with detailed supporting information in SSAR Chapter 7. The issues discussed in that material include the SSLC system, hardware and software development, electromagnetic compatibility, instrument setpoint methodology, environmental qualification of I&C equipment, and I&C system diversity and defense-in-depth considerations. Since the additional CDM information in CDM Section 3.0 apply to both safety- and non-safety-related I&C systems, the staff relied heavily on the information contained in those references in its reviews of the I&C systems.

The CDM Section 3.0 and SSAR contain criteria which describe the method to develop plans and procedures that will guide the design process throughout the lifecycle stages. The ITAAC provides the acceptance criteria for verifying the design through the stages while the SSAR adds the set of guidelines and standards that will provide more detailed criteria for the development of the design. The CDM has been written to incorporate the most important and general aspects (top-level requirements) from the standards. The set of standards and criteria in the SSAR encompass the guidance for generating the plans that will be used in the computer software and hardware design process for the computer design throughout the lifecycle.

The certified design description and design development process continue for the lifetime of the plant. Any safety-related software that is changed or added after plant startup is required to either be developed using the certified design development process described in the computer CDM, or the licensee must submit a design process (together with the design bases) description that will produce software of the same or higher quality than the original certified design process, consistent with the CDM. The licensee will be required to use the approved software change procedure (SCP) based upon the certified design development process for the operation stage of the lifecycle.

14.3.2.4 Approval of the CDM for Structures and Systems

The staff performed a multidisciplinary review, utilizing several task groups, of the SSCs of the ABWR, in accordance with 10 CFR Part 52 and the guidance provided in SRMs related to design certification applications provided by the Commission. This review included information contained in multiple CDM and SSAR submittals to the staff, as discussed in the background portion of Section 14.3 of this report.

Based on the task group reviews, the staff concludes that the top-level design features and performance characteristics of the SSCs important to safety in the ABWR are appropriately described in the design descriptions of the CDM, and are acceptable.

Further, these top-level design features and performance characteristics can be adequately verified by the ITAAC provided by GE. Therefore, the staff concludes that the ITAAC in the CDM are necessary and sufficient to provide reasonable assurance that if the inspections, tests, and analyses are performed and the acceptance criteria met, the SSCs important to safety in a facility that references the design have been constructed and will operate in accordance with the design certification and applicable regulations.

14.3.2.5 DFSER Issues

Section 2.0, "Tier 1 Material for ABWR Systems," of the "Tier 1 Design Certification Material for the GE ABWR," was under staff review at the time the DFSER was issued. The staff stated that the final evaluation would be provided in the FSER. This was DFSER Open Item 14.1.2-1.

The DFSER contained preliminary comments on the second stage submittal of the CDM, titled "Tier 1 Design Certification Material for the GE ABWR Design - Stage 2 Submittal," dated March 31, 1992. These preliminary comments were documented in many sections of the DFSER based on their relationship to the detailed design information in various SSAR sections. The preliminary comments provided in the DFSER were intended to discuss the philosophy of development of the Tier 1 design certification material, to establish early staff positions on the material, and to provide an indication of the status of development of the material.

GE provided resolutions to all of the staff's comments in various revisions to the Tier 1 CDM, and provided revised supporting design information in various SSAR amendments, as discussed in the background part of Section 14.3 of this report. In these submittals, GE addressed all of the comments of the staff, the ACRS, and an independent review group. Based on the revised material in the CDM and SSAR submittals, the CDM development process, criteria, and methodology described in SSAR Section 14.3, and the review process discussed in this section of this report, the staff concluded that these issues were adequately addressed where appropriate in the CDM. This resolved DFSER Open Item 14.1.2-1.

The following is a list of issues identified in the DFSER that specifically cited the Tier 1 material, and that were resolved as discussed above. This list of issues considered for treatment in the CDM is not all-inclusive, nor were all issues listed necessarily incorporated into the final CDM. The staff considered many issues for treatment in the CDM, using general approach and criteria discussed in this section of this report. Other specific issues considered for treatment in the CDM may also be discussed where applicable in other sections of this report.

<u>ITAAC Open Items In DFSER</u>

Item Number	Description of Item
2.6-1	Additional Site Parameters from Tier 1
3.2.1-3	ITAAC-plant specific walkdown
3.4.1-2	ITAAC-flood protection
3.5.1.1-1	ITAAC-protect SSCs from internally-generated missiles
3.5.1.2-2	ITAAC-protection of safety-related (SR) equipment from missiles

ITAAC Open Items In DFSER

Item Number	Description of Item
3.5.1.4-1	ITAAC-missiles generated from natural events
3.5.2-1	ITAAC-protect S/R SSCs from failure of non-S/R SSCs
3.5.2-2	ITAAC-protect SSCs from externally-generated missiles
3.6.1-1	ITAAC-protection of safety equipment from DBA
3.8.6-1	ITAAC-generic building design concerns (11 items)
3.8.6-2	ITAAC-SW design concerns (2 items)
3.8.6-3	ITAAC-containment design concerns (4 items)
3.8.6-4	ITAAC-containment structures design concerns (2 items)
3.8.6-5	ITAAC-reactor vessel pedestal design concerns (3 items)
3.8.6-6	ITAAC-reactor building design concerns (6 items)
3.8.6-7	ITAAC-control building design concerns (4 items)
3.8.6-8	ITAAC-radwaste building design concerns (2 items)
3.8.6-9	ITAAC-yard structures-stack systems design concerns (3 items)
3.9.6.4-1	ITAAC-generic MOV sizing
3.11.3-1	ITAAC-equipment qualification (EQ) radiation concern
4.4-2	ITAAC-LPMS consistent with RG 1.133
4.6-1	ITAAC-control rod drive system
5.2.2-1	ITAAC-safety relief valve and fuel
5.2.5-1	ITAAC-reactor coolant system leakage detection
5.4.1-1	ITAAC-recirculation flow control
5.4.6-1	ITAAC-reactor core isolation cooling

ITAAC Open Items In DFSER

Item Number	Description of Item
5.4.7-1	ITAAC-residual heat removal
6.2.1.7-1	ITAAC-containment design
6.2.3.1-1	ITAAC-functional of secondary containment
6.2.4-1	ITAAC-standby gas treatment system
6.2.4.1-4	ITAAC-containment isolation system
6.2.6-9	ITAAC-containment leak testing
6.3.6-1	ITAAC-high pressure core flooder
6.4-1	ITAAC-control room habitability
6.4-2	ITAAC-control room environmental design
7.1.3.3-1	ITAAC/DAC-instrument setpoints, safety system logic and control, EQ, computer development
7.1.4-1	ITAAC/DAC-neutron monitoring systems
7.2.2.3-1	ITAAC-timeout predetermined safe states
7.2.8-1	ITAAC-safety hazards, sneak circuit, timing analyses
7.4.1-1	ITAAC-systems required for safe shutdown
7.4.2-1	ITAAC-use of remote shutdown panel
7.6.1-1	ITAAC-interlock systems
7.7.1-1	ITAAC-key features of the control system
8.2.1.4-1	ITAAC-interfaces
8.2.2.1-1	ITAAC-physical separation (circuits and transformers)
8.2.2.1-2	ITAAC-circuit separation
8.2.2.2-1	ITAAC-physical separation (power, instrumentation, etc.)
8.2.2.3-1	ITAAC-electrical independence
8.2.2.4-1	ITAAC-testing of the offsite power system

ITAAC Open Items In DFSER

Item Number	Description of Item
8.2.2.6-1	ITAAC-capacity and capability of the offsite power system
8.2.2.7-1	ITAAC-electrical grounding
8.2.3.1-1	ITAAC-independence between offsite circuits and Class 1E
8.2.3.2-1	ITAAC-independence during parallel operations
8.2.3.3-1	ITAAC-LOCA during parallel operations
8.2.3.3-2	ITAAC-LOOP during parallel operations
8.2.3.3-3	ITAAC-diesel generator (DG) protective relaying when DG is operating
8.2.3.3-4	ITAAC-synchronizing interlocks
8.2.3.4-1	ITAAC-independence during ops/fail of non-safety load
8.2.3.4-2	ITAAC-configuration offsite connection
8.2.3.4-3	ITAAC-separation of offsite and onsite Class 1E
8.3-1	ITAAC-onsite Class 1E design
8.3.1.2-1	ITAAC-safe shutdown with one division
8.3.2.1-1	ITAAC-conduits to open tray separation
8.3.2.2-1	ITAAC-separation of neutron monitoring raceways
8.3.2.3-1	ITAAC-separation of dc emergency lighting raceways
8.3.2.4-1	ITAAC-separation between Class 1E penetrations
8.3.2.4-2	ITAAC-separation between Class 1E & non-Class 1E
8.3.2.4-3	ITAAC-separation of non-Class 1E from Class 1E
8.3.2.5-1	ITAAC-separation/protection of cables outside cabinets and panels

ITAAC Open Items In DFSER

Item Number	Description of Item
8.3.2.6-1	ITAAC-separation of cables inside cabinets/panels
8.3.2.7-1	ITAAC-separation of cables approaching/exiting cabinets
8.3.2.8-1	ITAAC-independence/physical separation of equipment
8.3.2.9-1	ITAAC-ID power, instrumentation, control equipment
8.3.2.9-3	ITAAC-ID neutron monitoring, scram solenoid
8.3.3.1-1	ITAAC-protection of electrical penetrations
8.3.3.2-1	ITAAC-design and qualification of electrical equipment
8.3.3.3-1	ITAAC-seismic qualification of light bulbs
8.3.3.4-1	ITAAC-submergence
8.3.3.5-2	ITAAC-protection of redundant Class 1E (environment)
8.3.3.6-1	ITAAC-associate circuits
8.3.3.7-1	ITAAC-diesel generator protective relaying bypass
8.3.3.8-1	ITAAC-thermal overload
8.3.3.10-1	ITAAC-protective relay
8.3.3.11-1	ITAAC-fault interrupting capacity
8.3.3.12-1	ITAAC-control of design parameters for MOV
8.3.3.13-1	ITAAC-separation of raceways
8.3.3.14-1	ITAAC-electrical protection for scram and MSIV
8.3.3.15-1	ITAAC-safety buses grounding
8.3.3.16-1	ITAAC-control of access to Class 1E equipment
8.3.4-1	ITAAC-electrical independence
8.3.4.1-1	ITAAC-interconnections

ITAAC Open Items In DFSER

Item Number	Description of Item
8.3.4.2-1	ITAAC-CVCF power supplies
8.3.4.4-1	ITAAC-isolation between safety and non-safety buses
8.3.5-1	ITAAC-lighting system under design basis accident
8.3.5-2	ITAAC-lighting requirements
8.3.6.1-1	ITAAC-control of the electrical design process
8.3.7-2	ITAAC-testing surveillance
8.3.8.1-1	ITAAC-non-safety dc power system
8.3.8.2-1	ITAAC-capacity of the Class 1E 125v dc battery supply
8.3.8.4-1	ITAAC-class 1E ac standby power system
8.3.8.5-1	ITAAC-constant voltage/constant frequency (CVCF) capacity
8.3.8.6-1	ITAAC-battery charger
8.3.8.7-1	ITAAC-distribution system
8.3.9.1-1	ITAAC-station blackout (SBO) compliance
8.3.9.2-1	ITAAC-SBO coping capability
8.3.9.3-1	ITAAC/DAC
9.1.1-1	ITAAC-change new fuel storage interface to ITAAC
9.1.2-1	ITAAC-change spent fuel storage interface to ITAAC
9.1.2-2	ITAAC-review fuel storage facility ITAAC and T1
9.1.3-2	ITAAC-fuel pool cooling and cleanup system
9.1.4-1	ITAAC-light load handling system
9.1.5-1	ITAAC-reactor building crane capability under safe shutdown earthquake
9.2.4-2	ITAAC-sanitary and potable water system

ITAAC Open Items In DFSER

Item Number	Description of Item
9.2.5-1	ITAAC-ultimate heat sink
9.2.8-1	ITAAC-makeup water system (preparation)
9.2.9-1	ITAAC-makeup water condensate system
9.2.10-2	ITAAC-makeup water (purified) system
9.2.11-2	ITAAC-reactor building cooling water system
9.2.12-1	ITAAC-heating, ventilation, and air conditioning (HVAC) normal cooling water
9.2.13-1	ITAAC-HVAC emergency cooling water system
9.2.14-1	ITAAC-turbine building cooling system
9.2.15-2	ITAAC-reactor service water system
9.2.16-2	ITAAC-turbine service water system
9.3.1-3	ITAAC-compressed air systems
9.3.2.2-3	ITAAC-post accident sampling system (PASS)
9.3.5-1	ITAAC-standby liquid control system (SLCS)
9.3.8-1	ITAAC-radioactive drain transfer system
9.3.9-1	ITAAC-hydrogen water chemistry
9.3.10-1	ITAAC-oxygen injection
9.3.11-1	ITAAC-zinc injection system
9.4-1	ITAAC-HVAC systems
9.5.1.3-1	ITAAC-fire protection system
9.5.4.1-2	ITAAC-DG and auxiliary system, fuel oil storage and transfer
9.5.5-1	ITAAC-put interfaces into ITAAC
10.2.1-1	ITAAC-turbine generator
10.2.2-1	ITAAC-turbine disk test data

ITAAC Open Items In DFSER	
Item Number	Description of Item
10.2.2-2	ITAAC-design bases for turbine design overspeed
10.3.1-1	ITAAC-main steam (7 items)
10.4.1-1	ITAAC-main condenser
10.4.2-1	ITAAC-main condenser evacuation system (7 items)
10.4.3-1	ITAAC-turbine gland seal system (7 items)
10.4.4-1	ITAAC-turbine bypass (7 items)
10.4.5-1	ITAAC-circulating water system (10 items)
10.4.7-1	ITAAC-condenser/feedwater (7 items)
11.0-1	ITAAC-radwaste system
12.3.5.3-1	ITAAC and Tier 1 radiation design submittal
13.3-3	ITAAC and Tier 1 for technical support center and operations support center
13.6.3.6-1	ITAAC-verify HVAC bulletproof features
14.1.1.5.2-1	ITAAC-roadmap of key analyses
14.1.1.5.3-1	ITAAC-certain systems may not have ITAAC
14.1.2-1	ITAAC-staff evaluation of system ITAAC
14.1.3.1-1	ITAAC-staff evaluation of generic ITAAC
14.1.3.3.3.6-1	ITAAC-structural design of small bore piping
14.1.3.3.3.9-1	ITAAC-buried piping design
14.1.3.3.4.1-1	ITAAC-confirmatory analysis on computer model adequacy
14.1.3.3.4.3-1	ITAAC-piping benchmark program
14.1.3.3.4.4-1	ITAAC-small bore piping decoupling criteria

ITAAC Open Items In DFSER	
Item Number	Description of Item
14.1.3.3.5.2-1	ITAAC-60 year life cycle factor of 1.5
14.1.3.3.5.7-1	ITAAC-environmental effects in fatigue design, Cl. 1
14.1.3.3.5.7-2	ITAAC-method of including environmental effects of fatigue
14.1.3.3.5.8-1	ITAAC-environmental effect in fatigue design, Cl. 2
14.1.3.3.5.10-1	ITAAC-methodology to address thermal striping
14.1.3.3.5.13-1	ITAAC-inertial and seismic motion effects
14.1.3.3.5.17-1	ITAAC-modal damping for composite structures
14.1.3.3.5.18-1	ITAAC-minimum temperature for thermal analyses
14.1.3.3.6-1	ITAAC-pipe support criteria (8 items)
14.1.3.3.7-1	ITAAC-high energy line break criteria
14.1.3.3.9.1-1	ITAAC-fatigue cumulative usage factor of 1.0
14.1.3.3.9.13-1	ITAAC-Tier 1 piping design description
14.1.3.8-1	ITAAC-reliability assurance program
14.1.3.9-1	ITAAC-welding
14.1.4-1	ITAAC-interface requirements
14.1.5-1	ITAAC-site parameters
14.2.12.7-1	ITAAC-design certification material initial test program
18.9.1-1	ITAAC-design description
19.1.2.2.2-2	ITAAC-fire barriers/separation
19.1.2.2.2-3	ITAAC-interface design for fires
19.1.2.4.2-1	ITAAC-PRA for internal, external events
19.1.5.2-3	ITAAC-interfacing piping

ITAAC Open Items In DFSER

Item Number	Description of Item
19.1.5.4-3	ITAAC-reliability of interfacing systems
19.1.5.6.3-1	ITAAC-human reliability analyses

ITAAC Confirmatory Items

19.1.6.3.2-3	ITAAC-seismic capacity of equipment
19.1.6.4-4	ITAAC-fire barrier installation, smoke mitigation
19.1.6.4-5	ITAAC-fire for COL design
5.2.4-1	ITAAC-add discussion of PSE and 89 code
7.2.2.1-1	ITAAC-verify maximum transmission distance
7.2.2.1-2	ITAAC-include accuracy in setpoint methods
2.2.5-2	ITAAC-Tier 1-eliminate test jumpers and lifted leads
7.2.3-1	ITAAC-verify bypasses are annunciated
7.2.8-1	ITAAC-Tier 1-software metrics to track error rate
7.2.8-4	ITAAC-equipment to be tested for low range EMI
7.7.1.5-1	ITAAC-reactor protection system trip ID in computer
7.10.2-1	ITAAC-follow EPRI for operating experience
8.2.3.4-1	ITAAC-independence safety vs non-safety systems
12.2.3-1	ITAAC/DAC include former interfaces
14.1.3.3.3.8-2	ITAAC-verification of seismic/non-seismic interactions
9.2.2.8-4	ITAAC-tests and analyses in CDM Table 3.6

The following issues were incorrectly classified in the DFSER as ITAAC COL Action Items, but were also resolved based on the revised CDM material, supporting SSAR information, the discussion in SSAR Section 14.3, and the discussion in this section of this report.

ITAAC COL Action Items

Item Number	Description of Action Item
9.5.7-1	ITAAC-DG lube oil system design criteria
9.5.8-1	ITAAC-DG combustion air system flow capacity

14.3.3 Additional Certified Design Material

This section of the ABWR CDM provides additional certified design material for design and construction activities that are applicable to more than one system. There are five entries in this CDM section, and these are discussed in the appropriate paragraphs that follow. The first four entries describe design related processes and associated DAC for the ABWR, and the fifth entry describes the ITP for a facility referencing the certified design. The design description for each entry describes its scope and applicability to the ABWR design. Amplifying information on CDM Section 3.0 is contained in SSAR Section 14.3.3. The material in this section of the CDM applies to the individual systems of the ABWR design contained in CDM Section 2.0, and the staff's review of the material in CDM Section 2.0 is contained in Section 14.3.2 of this report. The staff's safety evaluation for each design area where the DAC are used is contained in the section of this report applicable to the area.

Design Acceptance Criteria (DAC)

Design and engineering information for some areas of the design was not provided by GE at a level of detail customarily reviewed by the staff in making a final safety determination. GE provided less detailed information in these areas because GE believed they were either areas of rapidly changing technology and it would have been detrimental to freeze the details of the design many years before an actual plant was ready to be constructed, or because GE believed they were areas for which GE did not have sufficient as-built or as-procured information to complete the final design. Areas of rapidly changing technology included control room and RSS design (human factors) and advanced instrumentation and controls. Areas dependent on as-built or as-procured information included piping design and radiation shielding, ventilation, and airborne monitoring design. The staff provided its views on the DAC to the Commission in SECY-92-053, "Use of Design Acceptance Criteria During 10 CFR Part 52 Design Certification Reviews," dated February 19, 1992.

The design information and appropriate design methodologies, codes, and standards provided in the SSAR, together with the design descriptions and DAC, are sufficiently detailed to provide an adequate basis for the staff to make a final safety determination regarding the design, subject only to satisfactory design implementation and verification of the DAC by the COL applicant or licensee. The DAC are a set of prescribed limits, parameters, procedures, and attributes upon which the NRC relies, in a limited number of technical areas, in making a final safety determination in support of the ABWR design certification. The acceptance criteria for the DAC are objective; that is, they are inspectable, testable, or subject to analysis using pre-approved methods, and must be verified as a part of the ITAAC performed to demonstrate that the as-built facility conforms to the certified design. Thus, the acceptance criteria for DAC are specified together with the related ITAAC in the Tier 1 material, and both are part of the design certification. The DAC and the ITAAC, when met, ensure that the completed design and as-constructed plant conforms to the design certification. The material in the SSAR for each of the DAC areas includes, as appropriate, sample calculations or other supporting information to illustrate methods that are acceptable to the staff for meeting Tier 1 DAC commitments.

The structure of each area where DAC are used is the same as for the other areas of the design that are verified by ITAAC. The structure consists of three parts: the Tier 1 design description, the corresponding DAC, and the Tier 2 supporting information in the SSAR for the DAC. The staff has based its safety findings for the areas where DAC are used on the Tier 2 information specified in the SSAR, including applicable design methodologies, codes and standards, contingent on verification that the design has been properly implemented according to the Tier 1 design descriptions and the corresponding DAC.

For the two areas of rapidly changing technology, control room and RSS design (human factors) and instrumentation and controls design, the design descriptions and DAC delineate the process and requirements that a COL applicant or licensee must implement to develop the design information required in each area. Acceptance criteria are specified in the CDM for the development process at various stages of detailed design and subsequent construction and testing. The COL applicant or licensee is required to develop the procedures and test programs necessary to demonstrate that the DAC requirements are met at each stage. Similar to ITAAC, the COL applicant or licensee will certify to the NRC that the design through that stage is in compliance with the certified design. The NRC will review and inspect the work to confirm that the COL applicant or licensee has adequately implemented the commitments of the DAC at these stages. The process is referred to as a phased DAC because it consists of a set of sequential steps or stages that require successful completion. A COL applicant or licensee is not required to certify that each phase is completed sequentially. However, if the staff determines that a DAC was not successfully met, the design process may be required to be repeated to meet the DAC, possibly requiring a change to the as-built system design.

14.3.3.1 Human Factors Engineering DAC

The human factors aspects of the ABWR control room and remote shutdown system (RSS) design are provided in SSAR Chapter 18, "Human Factors," and together with the associated DAC in CDM Section 3.1, "Human Factors Engineering," are evaluated in Chapter 18 of this report. GE did not develop a final control room and RSS design before design certification because this is an area of rapidly changing technology. Instead, GE provided the processes and acceptance criteria by which the details of the design in this area would be developed, designed, and evaluated. GE provided amplifying information regarding the processes in this area in SSAR Section 14.3.3.1. The material in CDM Section 3.1 applies to the human factors design of the control room and the RSS. The implementation of the process and the design is the responsibility of the COL applicant or licensee.

Complete detailed human-system interface (HSI) design information was not available for staff review. The basis for the staff's review in this area was a HFE Program Review Model (PRM) developed by the staff. The staff's certification review in the control room design area was based on a design and implementation process plan. The staff informed the Commission of the development of the DAC in this area in SECY-92-299, "Development of Design Acceptance Criteria (DAC) for the Advanced Boiling Water Reactor (ABWR) in the Areas of Instrumentation and Controls (I&C) and Control Room Design," dated August 27, 1992.

The staff developed the HFE PRM, contained in Appendix J of this report, to serve as a technical basis for the review of the design process and DAC proposed by GE for certification of the ABWR control room and remote shutdown station design. The HFE PRM is (1) based upon currently accepted HFE practices, (2) well-defined, and (3) validated through experience with the development of complex, high-reliability systems in other industrial and military applications. The review model identifies the important HFE elements in a system development, design, and evaluation process that are necessary and sufficient prerequisites to successful integration of human factors in complex systems. The review model also identifies aspects of each HFE element that are key to a safety review, and describes acceptance criteria by which the HFE elements can be evaluated. The HFE PRM has eight program elements, each of which contain both general and more specific acceptance criteria.

The CDM describes the process to develop the HSI design information for the control room and RSS based on human factors systems analyses and human factors principles. The design effort will be directed by a multi-disciplinary HFE design team comprised of personnel with expertise in HFE and other technical areas relevant to the HSI design, evaluation and operations. The HSI design team shall develop a program plan to establish methods for implementing the HSI design through a process of human factors system analyses as discussed in CDM Figure 3.1, "Human-System Interface Design Implementation Process." The details of implementation of each stage of the development process are described in CDM Section 3.1, together with the related acceptance criteria. Detailed supporting information is contained primarily in SSAR Chapter 18, Appendix 18E.

The staff conducted a complete and thorough review of the ABWR CDM to ensure that the general criteria of the eight program elements in the HFE PRM were appropriately addressed in the Tier 1 CDM. The Tier 2 SSAR material contains more detailed guidelines and applicable guidance documents. The staff also conducted a review of the SSAR material to ensure that the specific acceptance criteria in the HFE PRM were appropriately addressed. The staff reviewed the CDM and SSAR Section 14.3.3.1 in accordance with the requirements in Part 52 and the guidance provided in SRMs related to design certification applications provided by the Commission. This review included information contained in multiple submittals to the staff as listed in the background part of Section 14.3 of this report.

The material in SSAR Chapter 18 provides design information and defines design processes that are acceptable for use in meeting the acceptance criteria in the CDM. However, the SSAR information may be changed by a COL applicant or licensee referencing the certified design in accordance with a "50.59-like" process. The staff's evaluation of the ABWR design for the control room is based on the design processes and acceptance criteria material in the DAC and the SSAR, especially those defined in SSAR Section 18E. Consequently, the staff indicated in Section 18 of this report that any proposed changes to SSAR Section 18E constitutes an unreviewed safety question and, therefore, must be submitted to the NRC for review and approval prior to implementation.

Conclusions

On the basis of the above, the staff concludes that the top-level design processes, features and performance characteristics of the human factors aspects of SSCs important to safety in the ABWR are appropriately described in the design descriptions of the CDM, and are acceptable.

Further, these top-level design processes, features and performance characteristics can be adequately verified by the DAC provided by GE. Therefore, the staff concludes that the DAC in the CDM are necessary and sufficient to provide reasonable assurance that if the inspections, tests, and analyses are performed and the acceptance criteria met, the human factors aspects of SSCs important to safety in a facility that references the design have been designed, constructed and will operate in accordance with the design certification and applicable regulations.

14.3.3.2 Radiation Protection DAC

The radiation protection aspects of the ABWR design are provided in SSAR Chapter 12, "Radiation Protection," and together with the associated DAC in CDM Section 3.2, "Radiation Protection," are evaluated in Chapter 12 of this report. GE did not provide the complete design information in this design area before design certification because the radiation shielding design and the calculated concentrations of airborne radioactive material were dependent upon as-built and as-procured information of plant systems and components. Therefore, GE was not able to describe the ABWR radiation source terms (i.e., the quantity and concentration of radioactive materials contained in, or leaking from plant systems) in sufficient detail to allow the staff to verify the adequacy of the shielding design, ventilation system designs, or the design and placement of the airborne radioactivity monitors. Instead, GE provided the processes and acceptance criteria by which the details of the design in this area would be developed, designed, and evaluated. GE provided amplifying information regarding the processes in this area in SSAR Section 14.3.3.2. This material in CDM Section 3.2 applies to the radiological shielding and ventilation design of the reactor building, turbine building, control building, service building, and radwaste building. The implementation of the process and the design is the responsibility of the COL applicant or licensee.

The acceptance criteria in the DAC are taken from the acceptance criteria in the applicable section of Chapter 12 of the SRP. The analysis methods and source term assumptions specified in the DAC are consistent with approved methods and assumptions listed in the SRP. The SRP is the basis for the staff's safety review of the ABWR design. Therefore, demonstrating that the final design meets these DAC with the methods and assumptions specified in Tier 1 ensures that the as-built ABWR design meets the applicable acceptance criteria of the SRP and the associated regulations and staff technical positions. The staff informed the Commission of the development of the DAC in this area in SECY-92-196, "Development of Design Acceptance Criteria (DAC) for the Advanced Boiling Water Reactor (ABWR)," dated May 28, 1992.

The DAC in the Tier 1 information address the verification of the plant radiation shielding design and the plant airborne concentrations of radioactive materials (e.g., the ventilation system and airborne monitoring system designs). The DAC require the COL applicant to calculate radiation levels and airborne radioactivity levels within the plant rooms and areas to verify the adequacy of these design features during plant construction (concurrently with the verification of the ITAAC). The plant rooms and areas to which the DAC apply are given in the figures in CDM Section 3.2. Detailed supporting information is contained in SSAR Chapter 12.

The criteria in CDM Table 3.2a, Items 1 and 2, ensure that the radiation shielding design (either that provided for by the plant structures, or design permanent or temporary shielding) is adequate to ensure that the maximum radiation levels in plant areas are commensurate with the area's access requirements so radiation exposures to plant personnel can be maintained as low as reasonably achievable (ALARA) during normal plant operations and maintenance. Item 4 in Table 3.2a ensures that adequate shielding is provided for those areas of the plant that may require occupancy to permit an operator to aid in the mitigation of or the recovery from an accident. Item 4 of Table 3.2a ensures that the contribution to the radiation dose from gamma shine (particularly from the turbine building) to a member of the public (off site) will be a small fraction of the EPA dose limit in 40 CFR Part 190.

The criteria in CDM Table 3.2b, Item 1, ensures that the plant provides adequate containment and ventilation flow rates to control the concentrations of airborne radioactivity to levels commensurate with the access requirements of areas in the plant. Item 2 in Table 3.2b ensures that once the concentrations of airborne radioactivity are determined per Item 1 above, the required airborne monitors are provided in the appropriate locations in the plant.

The staff conducted a complete and thorough review of the GE ABWR CDM material to ensure that the SRP guidelines for radiation protection design were appropriately addressed in both the Tier 1 CDM and the SSAR. The staff's evaluation included the analysis methods, design procedures, acceptance criteria, and related ITAAC that are to be used for the completion and verification of the ABWR radiation protection design. The SSAR information contains more detailed guidelines and applicable documents. The staff reviewed the CDM and SSAR Section 14.3.3.2 in accordance with the requirements in Part 52 and the guidance provided in SRMs related to design certification applications provided by the Commission.

Conclusions

On the basis of the above, the staff concludes that the top-level design processes, features and performance characteristics of the radiation protection aspects of SSCs important to safety in the ABWR are appropriately described in the design descriptions of the CDM, and are acceptable.

Further, these top-level design processes, features and performance characteristics can be adequately verified by the DAC provided by GE. Therefore, the staff concludes that the DAC in the CDM are necessary and sufficient to provide reasonable assurance that if the inspections, tests, and analyses are performed and the acceptance criteria met, the radiation protection aspects of SSCs important to safety in a facility that references the design have been designed, constructed and will operate in accordance with the design certification and applicable regulations.

14.3.3.3 Piping Design DAC

The piping design aspects of the ABWR design are provided in SSAR Chapter 3, "Structures, Components, Equipment, and Systems," and together with the associated DAC in CDM Section 3.3, "Piping Design," are evaluated in Section 3.12 of this report. GE did not provide the complete design information in this design area before design certification because the piping design was dependent upon as-built and as-procured information. Instead, GE provided the processes and acceptance criteria by which the details of the design in this area would be developed, designed, and evaluated. GE provided amplifying information regarding the processes in this area in SSAR Section 14.3.3.3. The material in CDM Section 3.3 applies to ABWR piping systems classified as nuclear safety-related, and to non-nuclear safety systems as specified in the Tier 1 material for the individual systems in CDM Section 2.0. The implementation of the process and the design is the responsibility of the COL applicant or licensee.

The staff used the SRP guidelines to evaluate the piping design information in the ABWR CDM and SSAR and performed a detailed audit of the piping design criteria, including sample calculations. The staff evaluated the adequacy of the structural integrity and functional capability of safety-related piping systems. The review was not limited to the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Class 1, 2, and 3 piping and supports, but included buried piping, instrumentation lines, the interaction of non-seismic Category I piping with seismic Category I piping, and any safety-related piping designed to industry standards other than the ASME Code. The staff's evaluation included the analysis methods, design procedures, acceptance criteria, and related ITAAC that are to be used for the completion and verification of the ABWR piping design. The staff's evaluation included both CDM and SSAR information regarding the applicable codes and standards, analysis methods to be used for completing the piping design, modeling techniques, pipe stress analyses criteria, pipe support design criteria, high-energy line break criteria, and leak-before-break (LBB) approach applicable to the ABWR. The staff informed the Commission of the development of the DAC in this area in SECY-92-196, "Development of Design Acceptance Criteria (DAC) for the Advanced Boiling Water Reactor (ABWR)," dated May 28, 1992.

The material in CDM Section 3.3 provides the design process to develop the piping for the nuclear safety-related (seismic Category I) systems of the ABWR design. Piping systems that must remain functional during and following an SSE are designated as seismic Category I and are further classified as ASME Code Class 1, 2, or 3. The piping systems and their components are designed and constructed in accordance with the ASME Code requirements identified in the individual systems of the ABWR design. The CDM ensures that the piping systems will be designed to perform their safety-related functions under all postulated combinations of normal operating conditions, system operating transients, postulated pipe breaks, and seismic events. The material in the CDM section also addresses the consequential effects of pipe ruptures such as jet impingement, potential missile generation, and pressure and temperature effects.

GE specified three ITAAC in the CDM to ensure the design process for piping systems was as described in the design description. The first ITAAC specified in the CDM requires that an ASME Code certified stress report exists to ensure that the ASME Code Class 1, 2, or 3 piping systems are designed to retain their pressure integrity and functional capability under internal design and operating pressures and design basis loads. The specific contents and requirements of the certified stress report are contained in the ASME Code. As used in this report, an ASME Code certified stress report is the design document required by ASME Code, Section III, Subarticle NCA-3550. A certified piping stress report provides assurance that requirements of the ASME Code, Section III for design, fabrication, installation, examination, and testing have been met and that the design complies with the design specifications.

The second ITAAC requires that a pipe break analysis report exists that documents that SSCs that are required to be functional during and following an SSE have adequate high-energy pipe break mitigation features, or alternatively, that a leak-before-break report exists for those sections of piping systems qualified for leak-before-break design. As discussed in the design description, the pipe break analysis report specifies the criteria used to postulate pipe breaks and the analytical methods used to perform pipe breaks and confirms the adequacy of the results of the pipe break analyses. This verification provides assurance that the high-energy line break analyses have been completed and meet the following certified design commitments. For postulated pipe breaks, the Pipe Break Analysis Report shall confirm that: (1) piping stresses in the containment penetration area shall be within their allowable stress limits, (2) pipe whip restraints and jet shield designs shall be capable of mitigating pipe break loads, (3) loads on safety-related SSCs shall be within their design load limits, and (4) SSCs are protected or are qualified to withstand the environmental effects of postulated failures. The Pipe Break Analysis Report shall conclude that, for each postulated piping failure, the reactor can be shut down safely and maintained in a safe, cold shutdown condition without offsite power. Detailed information that supports this ITAAC is contained in SSAR Chapter 3.

The third ITAAC requires that an as-built piping stress report exists that documents the results of an as-built reconciliation analysis confirming that the final piping system has been built in accordance with the ASME Code certified stress report. The report provides an overall verification that the as-constructed piping system is consistent with the certified design commitments. Although similar to the first ITAAC, this verification also provides assurance that modification of any document used for construction from the corresponding document used for design analysis has been reconciled with the certified stress report discussed above. This documentation may become part of the certified stress report.

As discussed in the advance SER, GE stated that it intended to provide, in a future SSAR amendment, amplifying information in the SSAR to support the piping DAC. This was Confirmatory Item F14.3.3.3.1-1. GE provided this amplifying information in Amendment 34. The staff finds this acceptable. This resolved Confirmatory Item F14.3.3.3-1.

The staff conducted a complete and thorough review of the GE ABWR CDM material to ensure that the SRP guidelines for piping design were appropriately addressed in both the Tier 1 CDM and the SSAR. The staff's evaluation included the analysis methods, design procedures, acceptance criteria, and related ITAAC that are to be used for the completion and verification of the ABWR piping design. The Tier 2 SSAR material contains more detailed guidelines and applicable documents. The staff reviewed the CDM and SSAR Section 14.3.3.3 in accordance with the requirements in Part 52 and the guidance provided in SRMs related to design certification applications provided by the Commission. This review included information contained in multiple submittals to the staff as listed in the introductory part of Section 14.3 of this report.

Selected material in SSAR Chapter 3 provides design information and defines design processes that are acceptable for use in meeting the piping DAC in the CDM. However, the SSAR information may be changed by a COL applicant or licensee referencing the certified design in accordance with a "50.59-like" process. The staff's evaluation of the ABWR design for piping systems is based on the design processes and acceptance criteria material in the DAC and the SSAR. Consequently, the staff indicated in Section 3.12 of this report that any proposed changes to selected aspects of these piping design processes described in the appropriate SSAR sections constitutes an unreviewed safety question and, therefore, must be submitted to the NRC for review and approval prior to implementation.

Conclusions

On the basis of the above, the staff concludes that the top-level design processes, features and performance characteristics of the piping design aspects of SSCs important to safety in the ABWR are appropriately described in the design descriptions of the CDM, and are acceptable.

Further, these top-level design processes, features and performance characteristics can be adequately verified by the DAC provided by GE. Therefore, the staff concludes that the DAC in the CDM are necessary and sufficient to provide reasonable assurance that if the inspections, tests, and analyses are performed and the acceptance criteria met, the piping design aspects of SSCs important to safety in a facility that references the design have been designed, constructed and will operate in accordance with the design certification and applicable regulations.

14.3.3.4 Instrumentation and Controls (I&C) DAC and Other I&C ITAAC

The I&C aspects of the ABWR design are provided in SSAR Chapter 7, "Instrumentation and Control Systems," and together with the associated DAC and other multi-system I&C related ITAAC in CDM Section 3.4, "Instrumentation and Control," are evaluated in Chapter 7 of this report. GE did not develop a final design for I&C before design certification because this is an area of rapidly changing technology. Instead, GE provided the processes and acceptance criteria by which the details of the design in this area would be developed, designed, and evaluated. GE provided amplifying information regarding the processes in this area in SSAR Section 14.3.3.4. The material in CDM Section 3.4 applies to the design of both safety related and non-safety related I&C systems of the ABWR. These I&C systems are described in CDM Section 2.0, and the staff's review of these systems is described in Section 14.3.2 of this report. The implementation of the process and the design is the responsibility of the COL applicant or licensee.

The staff used the SRP guidelines to review the I&C design information in the ABWR CDM and SSAR to confirm that both the safety-related and non-safety-related I&C systems met the appropriate acceptance criteria of the SRP. The staff also used the Commission guidance contained in a SRM of July 15, 1993, related to SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light Water Reactor (ALWR) Designs," dated April 2, 1993. The staff informed the Commission of the development of the DAC in this area in SECY-92-299, "Development of Design Acceptance Criteria (DAC) for the Advanced Boiling Water Reactor (ABWR) in the Areas of Instrumentation and Controls (I&C) and Control Room Design," dated August 27, 1992.

CDM Section 3.4 has multiple entries addressing three key issues associated with the I&C design. These issues include the design of the SSLC system, the development and qualification processes for I&C systems, and design features that provide diverse backup as protection against common-mode failures in the SSLC. These issues and their relationships to other systems of the design are illustrated in the figures in CDM Section 3.4, which contain a block diagram showing the SSLC logic and control, a depiction of the integrated hardware and software development process for I&C systems, and a diagram showing the interfaces of the SSLC system with other I&C systems in the design. Detailed supporting information for the CDM is contained in SSAR Chapter 7.

CDM Section 3.4A, "Safety System Logic and Control," contains material for the SSLC. The SSLC integrates the automatic decision-making and trip logic functions, and manual initiation functions associated with the safety actions of the safety-related systems. Safety-related trip logic and monitoring of plant protection system resides in SSLC equipment. SSLC equipment comprises microprocessor-based, software-controlled signal processors that perform signal conditioning, setpoint comparison, trip logic, system initiation and reset, self-test, calibration, and bypass functions. The signal processors associated with a particular safety-related system are an integral part of that system and do not belong to SSLC.

CDM Section 3.4B, "I&C Development and Qualification Processes," contains the DAC for the I&C area of the design. The DAC are contained in four subsections that describe (1) design processes and acceptance criteria to be used for safety-related systems using programmable microprocessor-based control equipment, (2) a program to assess and mitigate the effects of electromagnetic interference on I&C equipment, (3) a program to establish setpoint for safety-related instrument channels, and (4) a program to qualify safety-related I&C equipment for in-service environmental conditions.

The subsection of CDM Section 3.4B titled, "Hardware and Software Development Process," describes hardware and software development processes to be used in the design, testing, and installation of I&C equipment. The following discussion addresses the considerations made in reviewing the entries for this subsection of the CDM.

The primary function of this development process is to implement the functional instrumentation and control requirements described in the CDM and the SSAR for the systems which comprise the ABWR. The decomposition of the functional system (SSLC, RPS, ARI, etc.) requirements to specific computer hardware and software components to perform the various tasks is accomplished using the structured design process described below.

The CDM includes the description of the design process to be followed for hardware and software development, design commitments, the inspections, tests, and analysis to be performed to verify that the design is consistent with the commitments, and the appropriate acceptance criteria against which the design will be judged. This ITAAC describes attributes of the process to be used to develop the software as well as attributes of the final software product. The ITAAC for software and hardware describes the following design stages within the design process:

(1) Planning
(2) Design definition
(3) Software design
(4) Software coding
(5) Integration
(6) Validation
(7) Change control

The CDM and SSAR contain criteria which describe the method to develop plans and procedures that will guide the design process throughout the lifecycle stages. The ITAAC provides the acceptance criteria for verifying the design through the stages listed above, while the SSAR adds the set of guidelines and standards that will provide more detailed criteria for the development of the design. The CDM has been written to incorporate the most important and general aspects (top-level requirements) from the standards. The set of standards and criteria in the SSAR encompass the guidance for generating the plans that will be used in the computer software and hardware design process for the computer design throughout the lifecycle. These plans are described below.

The software QA (SQA) plan describes the software-specific activities that are to be performed and controlled in addition to the approved QA plan (in accordance with 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants") for the total ABWR design. The SQA plan establishes the criteria under which the other software development plans will be generated. The software management plan (SMP) establishes the organization and authority structure for the design, the procedures to be used, and the interrelationships between major activities. The software configuration management plan (CMP) provides the means to identify software products, control and implement changes, and record and report change implementation status. The software development plan (SDP) describes a development process, tools documentation, and products developed according to the software lifecycle. The verification and validation plan (V&VP) describes the method to ensure that the requirements of each phase or stage of the design process (lifecycle) are fully and accurately implemented into the next phase. The software safety plan (SSP) describes the safety and hazards analyses that will be performed. The software operation and maintenance plan (SOMP) includes the procedures required to ensure that the software will be operated correctly and that the quality of the software is maintained. GE has combined these plans into a software management plan, a configuration management plan, and a verification and validation plan.

The ITAAC activities completed by the COL applicant will be inspected by the NRC to verify conformance with the requirements at several stages during the digital control system design process or stage of the lifecycle. The documents which demonstrate satisfactory implementation of the ITAAC will be available for inspection during the NRC audit at the completion of each of the above stages. The stages or phases described by GE are shown in Figure 3.4d. The NRC audit and the COL applicant conformance review points are shown in Figure 7.1-2 of the ABWR FSER. These stages correspond with the phases described by GE in the CDM. The actual stages, including the conformance review and audit points, will be determined for each of the software products to be developed when design implementation is scheduled to begin.

At each stage, the design development must be verified by the COL applicant to be in accordance with the certified design process and the detailed design developed (through that stage) to be in conformance with the certified design. Upon completion of ITAAC activities for each stage, the COL applicant will certify to the NRC that the stage has been completed and the design and construction completed up through that stage is in compliance with the certified design. Although not required, the COL applicant should satisfactorily complete ITAAC activities at each stage prior to proceeding to the next stage of the design development process. Failure to successfully complete the ITAAC at a stage, as determined by the conformance review or the NRC audit, may require repeating an earlier stage ITAAC or changing the system design. The NRC staff will identify any open issues which require resolution for each stage of the ITAAC. Significant open issues which are not resolved could result in the NRC staff concluding that the ITAAC had not been satisfactorily completed.

The subsection of Section 3.4B titled "Electromagnetic Compatibility" describes the process to ensure that I&C equipment is able to function properly when subjected to an electromagnetic environment. An electromagnetic compatibility (EMC) compliance plan to confirm the level of immunity to electrical noise is included in the design, installation, and testing of I&C equipment. The plan is structured on the basis that EMC of I&C equipment is verified by factory testing and site testing of both individual components and interconnected systems to meet electromagnetic compatibility requirements.

The subsection of Section 3.4B titled "Instrument Setpoint Methodology" describes the process to ensure that setpoints for initiation of safety-related functions are determined, documented, installed, and maintained. The process (the instrument setpoint methodology) establishes a program for specifying requirements for documenting the bases for selection of trip setpoints, accounting for instrument inaccuracies, response testing, and replacement of instrumentation.

The subsection of Section 3.4B titled "Equipment Qualification" describes the process to ensure that qualification of safety-related I&C equipment is able to complete its safety-related function under the environmental conditions that exist up to and including the time the equipment has finished performing that function. An equipment qualification program is established that ensures qualification specifications consider conditions that exist during normal, abnormal, and design-basis accident events in terms of their cumulative effect on equipment performance for the period up to the end of equipment life.

CDM Section 3.4C, "Diversity and Defense-In-Depth Considerations," addresses the concern that software design faults or other initiating events common to redundant, multidivisional logic channels of I&C protection systems could disable significant portions of the plant's safety functions at the moment when these functions are needed to mitigate an accident, and addresses the diverse backup features that are provided for the primary automatic logic. Diversity is provided in the form of hard-wired backup for reactor trip, diverse display of important process parameters, defense-in-depth arrangement of equipment, and other equipment diversity.

The staff conducted a complete and thorough review of the GE ABWR CDM material to ensure that the SRP guidelines and Commission guidance for I&C design were appropriately addressed in both the CDM and the SSAR. The staff's evaluation included the analysis methods, design procedures, acceptance criteria, and related ITAAC that are to be used for the completion and verification of the ABWR I&C design. The SSAR material contains more detailed guidelines and applicable guidance documents. The staff reviewed the CDM and SSAR Section 14.3.3.4 in accordance with the requirements in Part 52 and the guidance provided in SRMs related to design certification applications provided by the Commission. This review included information contained in multiple submittals to the staff as listed in the background part of Section 14.3 of this report.

Selected material in SSAR Section 7.2 provides detailed design information and defines design processes that are acceptable for use in meeting the I&C DAC in the CDM. However, the SSAR information may be changed by a COL applicant or licensee referencing the certified design in accordance with a "50.59-like" process. The staff's evaluation of the ABWR design for I&C systems is based on the design processes and acceptance criteria material in the DAC and the SSAR. Consequently, the staff indicated in Section 7.2 of this report that any proposed changes to the appropriate SSAR sections constitutes an unreviewed safety question and, therefore, must be submitted to the NRC for review and approval prior to implementation.

Conclusions

On the basis of the above, the staff concludes that the top-level design processes, features and performance characteristics of the I&C aspects of SSCs important to safety in the ABWR are appropriately described in the design descriptions of the CDM, and are acceptable.

Further, these top-level design processes, features and performance characteristics can be adequately verified by the DAC provided by GE. Therefore, the staff concludes that the DAC in the CDM are necessary and sufficient to provide reasonable assurance that if the inspections, tests, and analyses are performed and the acceptance criteria met, the I&C aspects of SSCs important to safety in a facility that references the design have been designed, constructed and will operate in accordance with the design certification and applicable regulations.

14.3.3.5 Initial Test Program (ITP)

This section of the CDM consists of a high level commitment to an ITP and a description of the program and major program documents (i.e., a site-specific startup administrative manual, test specifications, and test procedures). The ABWR SSAR Chapter 14.2 contains a complete description of the ITP, and the staff's evaluation of the ITP is contained in Section 14.2 of this report.

The staff reviewed this CDM Section for consistency with the guidelines contained in the SRP and RG 1.68, "Initial Test Program for Water-Cooled Nuclear Power Plants." RG 1.68 describes the general scope and depth of testing that is acceptable to the staff for conduct of preoperational and startup testing as part of the ITP.

The key facets of the ITP are described in the Tier 1 CDM to ensure that subsequent changes in the conduct of the ITP cannot be initiated unilaterally by the COL applicant. This ITP is described in Tier 1 because of the essential role of a test program in the verification that SSCs have been constructed and will perform satisfactorily in service. The Tier 1 description requires that the ITP be performed under suitably controlled conditions and processes. The development of test procedures, conduct of the tests, and safe execution of the test program, are important considerations in ensuring that as-built facility is in accordance with the design certification and applicable regulations. Thus, the staff will have the confidence that the ITP will be implemented effectively, so that the appropriate testing methodologies, and associated programmatic controls for testing plant systems will be ensured.

A corresponding ITAAC for this design description is not required for several reasons:

(1) The Tier 1 certified design material consists of a high level commitment to an ITP, and a description of the program and major program documents that constitute an acceptable ITP (i.e., a site-specific startup administrative manual, test specifications, and test procedures). The specific testing necessary to verify design features and performance aspects of the design is delineated in the system-specific ITAAC.

(2) The ITP covers a broader spectrum of time than the ITAAC. While ITP pre-operational testing shall be completed prior to fuel load, the ITP startup and power ascension testing will be conducted after fuel load. As the ITP involves testing post-fuel load, it is not appropriate to define associated ITAAC entries as Part 52 specifies that the ITAAC will be completed prior to fuel load.

In summary, the top-level ITP commitments in the CDM ensure that suitable controls are imposed over the pre-operational and start-up testing programs, which provide reasonable assurance that the facility can be operated without undue risk to the public. The staff concludes that the ITP information in the CDM is acceptable.

14.3.4 Interface Requirements

The requirements for interfaces for a design are contained in 10 CFR 52.47(a)(1)(vii-ix). An applicant for design certification is required to provide (1) the interface requirements to be met by those portions of the plant for which the application does not seek certification, (2) justification that compliance with the interface requirements is verifiable through inspection, testing , or analysis, and the method to be used for verification of interface requirements, and (3) a representative conceptual design for those portions of the plant for which the application does not seek certification. The staff evaluated these interface requirements and the ABWR design in the appropriate sections of this report.

GE defined the interface between the systems of the design and the site-specific systems to be at the walls of the turbine building, reactor building, and control building, as depicted in Figure 1.2-1 of this report. This section of the CDM specifies interface requirements for those portions of the certified design that interface with site-specific portions of the design, and specifies the systems that are completely or partially out of scope of the certified design. The interface requirements define the design attributes and performance characteristics that must be met by the site-specific, out-of-scope portion of the plant so that the site-specific portion of the design is in conformance with the certified design. The site-specific portions of the design are those portions of the design that are dependent on characteristics of the site, such as the design of the ultimate heat sink.

The review of the appropriate inspections, tests, and analyses to demonstrate compliance with the interface requirements for the site-specific portion of the design is accomplished in the review of an application and for a combined license under Subpart C of 10 CFR Part 52.

GE provided information discussing the interface requirements in the CDM and in SSAR Section 14.3.4. GE provided acceptable interface requirements in CDM Section 4.0, and in the appropriate systems in CDM Section 2.0. This information was based on the information in the various sections of the SSAR, and is evaluated by the staff in the appropriate sections of this report. In CDM Section 4.0, GE stated that the development of ITAAC for the interface requirements will be similar in nature to the development of ITAAC in CDM Section 2.0. The staff concludes that this is an acceptable justification that compliance with the interfaces is verifiable through ITAAC, and the process described in SSAR Section 14.3 provides an acceptable methodology for verification of the interface requirements. GE provided acceptable representative conceptual designs in the SSAR that enabled the staff to complete its review of the design, as discussed in the appropriate sections of this report.

Therefore, based on the above discussion, the staff concludes that the interface information provided by GE meets the requirements contained in 10 CFR 52.47(a)(1)(vii - ix), and is acceptable.

14.3.5 Site Parameters

The requirements for site parameters for a design are contained in 10 CFR 52.47(a)(1)(iii). An applicant for design certification is required to provide the site parameters used in the design, and an analysis and evaluation of the design in terms of these parameters. The site parameters are specified in both the CDM Section 5.0 and Chapter 2 of the SSAR, and the analysis and evaluation of the design is contained in the various sections of the SSAR. The staff evaluated these parameters and the design in the appropriate sections of this report.

Site parameters are specified in this section of the CDM for establishing the bounding parameters to be used in the selection of a suitable site for a facility referencing the ABWR certified design. Because they were used in bounding evaluations of the certified design, they define the requirements for the design that must be met by a site to ensure that a facility built on the site remains in conformance with the design certification. The demonstration that the site parameters are met at a given site is accomplished in conjunction with an application and issuance of a combined license under Subpart C of 10 CFR Part 52.

GE provided information discussing the site parameters in the CDM and in SSAR Section 14.3.5. GE provided acceptable site parameters postulated for the certified design in CDM Section 5.0 and in the appropriate sections of the SSAR. The appropriate sections of the SSAR information also provided an acceptable analysis and evaluation of the design in terms of these parameters, and the staff found the design acceptable in the related sections of this report. Therefore, the staff concludes that the site parameter information provided by GE meets the requirements of 10 CFR 52.47(a)(1)(iii), and is acceptable.

14.3.6 Summary

The staff reviewed the GE ABWR CDM and SSAR Section 14.3 in accordance with the requirements in Part 52 and the guidance provided in SRMs related to design certification applications provided by the Commission. This review included information contained in multiple submittals to the staff, as listed in the background portion of Section 14.3 of this report.

Based on the staff's review of the material in the CDM, and a review of the selection methodology and criteria for the development of the CDM contained in SSAR Section 14.3, the staff concludes that the top-level design features and performance characteristics of the ABWR SSCs important to safety are appropriately described in the CDM, and are acceptable.

Further, these top-level commitments can be adequately verified by the ITAAC and additional certified design material provided by GE. Therefore, in the appropriate parts of Section 14.3 of this report, the staff concludes that the CDM are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed, and the acceptance criteria met, a facility referencing the certified design will be constructed and will operate in conformity with the design certification and applicable regulations.

The staff also concludes in the appropriate parts of Section 14.3 of this report that the interface requirements and site parameters provided by GE for the ABWR meet the requirements for design certification applications in 10 CFR 52.47, and are acceptable.

14.3.7 DFSER Issues

14.3.7.1 Treatment of Non-Traditional Items in ITAAC

GE incorporated into the SSAR any insights into the design that were obtained from non-traditional items such as PRA and severe accident issue resolutions. Additionally, the staff followed the Commission's guidance in its review of the evolutionary designs for the resolution of non-traditional issues such as those discussed in SECY-90-016, "Evolutionary Light Water Reactor Certification Issues and Their Relationships to Current Regulatory Requirements," and SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs." In 10 CFR 52.47, the Commission specified that the ITAAC must provide reasonable assurance that "a plant which references the design is built and will operate in accordance with the design certification." Therefore, by verifying key aspects and features of the design, the ITAAC implicitly confirm the implementation of these non-traditional items and the safety findings contained in the safety evaluation report.

The staff requested GE to develop a cross-reference of key aspects, analyses, and features of the design from the SSAR to the CDM in order to document how these issues had been incorporated into the CDM. Specifically, the cross- references were to show how key aspects of the accident analyses, PRA, and severe accident issue resolutions were included in the CDM. This was DFSER Open Item 14.1.1.5.2-1.

GE provided the cross-references in SSAR Amendment 33 and updated the cross-references in Amendment 34. In those submittals, GE provided cross-references for key safety and integrated plant analyses in SSAR Section 14.3, and cross- references for PRA and severe accident analyses in SSAR Section 19.8. GE also provided more detailed cross-references for these analyses in a letter dated March 31, 1994.

The treatment of non-traditional items in the CDM for the ABWR design is discussed further in Section 14.3.2 of this report. Based on the discussion in Section 14.3.2, the staff found the cross-references, and the treatment of non-traditional items in the CDM acceptable. This resolved DFSER Open Item 14.1.1.5.2-1.

14.3.7.2 Relationship of the Design Description to the ITAAC

GE proposed that certain systems could have Tier 1 design descriptions, but may not require any corresponding ITAAC to verify the design for those systems. Examples of these systems were the fuel service equipment, the internal pump maintenance facility, and the fuel cask cleaning facility. The staff was reviewing this proposal at the time the DFSER was issued. This was DFSER Open Item 14.1.1.5.3-1.

GE adopted a graded approach to the level of detail in the development of the CDM, based on the safety significance of the ABWR structures, systems, and components. In SSAR Section 14.3, GE provided the process it used in the development of the CDM, based on the design presented in the appropriate sections of the SSAR. GE applied various selection criteria to the information in multiple chapters of the SSAR to determine the level of design information for a given system in the CDM.

GE provided its selection criteria and methodology for the ITAAC in SSAR Section 14.3.2.2. In general, each ABWR system with information in the design description has one or more ITAAC, based on the its safety significance. A single ITAAC may verify one or more provisions in the design description. Other aspects of systems may be satisfied by ITAAC contained in other systems or other sections of the CDM. For example, the piping design information in CDM Section 3.3 provides acceptance criteria for seismic Category I and ASME Code Class 1, 2, and 3 components. Additionally, since Part 52 requires that the ITAAC be satisfied prior to fuel loading, there are no ITAAC to verify any information dependent on post-fuel load conditions (e.g., nuclear fuel, fuel channels, and control rods). This information will be verified by the ITP as part of start-up and power ascension testing.

In Section 14.3 of this report, "Basis for Approval of the CDM," the staff discussed its graded approach to the review of the CDM based on the safety significance of the SSCs. Thus, consistent with the guidance of Part 52 and the SRM related to SECY-90-377, the staff recognized that although many aspects of the design were important to safety, the level of design detail in the CDM and verification of the key features and performance characteristics should be commensurate with the importance of the safety functions to be performed.

The relationship of design descriptions and ITAAC for the ABWR design is discussed further in Section 14.3.2 of this report. Based on the discussion in Section 14.3.2, the staff found the treatment of design descriptions and ITAAC in the CDM acceptable. This resolved DFSER Open Item 14.1.1.5.3-1.

14.3.7.3 System Design Descriptions and ITAAC

Section 2.0, "Tier 1 Material for ABWR Systems," of the "Tier 1 Design Certification Material for the GE ABWR," was under staff review at the time the DFSER was issued. The staff stated that the evaluation would be provided in the FSER. This was DFSER Open Item 14.1.2-1.

GE provided revised CDM information and supporting information in various SSAR submittals to the staff as discussed in the background part of this report. In these submittals, GE addressed all of the comments of the staff, the ACRS, and an independent review group. As discussed in Section 14.3 of this report, "Basis for Approval of the CDM," the staff reviewed the GE ABWR CDM and selection, methodology and criteria for the development of the CDM contained in SSAR Section 14.3 in accordance with the requirements in Part 52 and the guidance provided in SRMs related to design certification applications provided by the Commission.

System design descriptions and ITAAC for the ABWR design are discussed further in Section 14.3.2 of this report. Based on the discussion in Section 14.3.2, the staff found the treatment of system design descriptions and ITAAC in the CDM acceptable. This resolved DFSER Open Item 14.1.2-1.

14.3.7.4 Equipment Qualification (EQ)

Section 3.1, "Equipment Qualification," of the "Tier 1 Design Certification Material for the GE ABWR," was under staff review at the time the DFSER was issued, and the staff stated that the evaluation would be provided in the FSER. This was DFSER Open Item 14.1.3.1-1.

GE provided revised CDM information in a submittal dated August 31, 1993, and supporting information in Amendment 32. GE eliminated the proposed CDM Section 3.1, and put the required equipment qualification information in CDM Section 1.2, General Provisions. GE provided supporting information on equipment qualification in the SSAR. The equipment qualification of safety-significant portions of the design will be verified as part of the basic configuration walkdown of individual SSCs.

The basic configuration walkdown for equipment qualification is discussed in greater detail in Sections 14.3.1.2 and 14.3.2.3.1 of this report. Based on the discussion in Sections 14.3.1.2, and 14.3.2.3.1 the staff found the treatment of equipment qualification in the CDM acceptable. This resolved DFSER Open Item 14.1.3.1-1.

14.3.7.5 Reliability Assurance Program

Section 3.8, "Reliability Assurance Program," of the "Tier 1 Design Certification Material for the GE ABWR," was under staff review at the time the DFSER was issued. The staff stated that the evaluation would be provided in the FSER. This was DFSER Open Item 14.1.3.8-1.

When the DFSER was written the staff's position on a reliability assurance program (RAP) was that a high-level commitment to a RAP applicable to design certification (D-RAP) was required. GE committed to provide the required D-RAP commitments in the CDM in a letter dated July 12, 1994. The staff finds GE's commitments to D-RAP acceptable, subject to incorporation in the CDM. DFSER Open Item 14.1.3.8-1 becomes Confirmatory Item F14.3.7.5-1.

14.3.7.6 Welding

Section 3.9, "Welding," of the "Tier 1 Design Certification Material for the GE ABWR," was under staff review at the time the DFSER was issued. The staff stated that the evaluation would be provided in the FSER. This was DFSER Open Item 14.1.3.9-1.

GE provided revised CDM information in a submittal dated August 31, 1993, and supporting information in Amendment 32. GE eliminated the proposed CDM Section 3.9, and put the required welding information in CDM Section 1.2, General Provisions. GE provided supporting information on welding in the SSAR. The welding aspects of safety-significant portions of the design will be verified as part of the basic configuration walkdown of individual SSCs.

The basic configuration walkdown for welding is discussed in greater detail in Section 14.3.1.2 of this report. Based on the discussion in Section 14.3.1.2, the staff found the treatment of welding in the CDM acceptable. This resolved DFSER Open Item 14.1.3.9-1.

14.3.7.7 Interface Requirements

Section 4.0, "Interface Tier 1 Material," of the "Tier 1 Design Certification Material for the GE ABWR," was under staff review at the time the DFSER was issued. The staff stated that the evaluation would be provided in the FSER. This was DFSER Open Item 14.1.4-1.

The staff provided comments on interfaces during a meeting with GE on July 27, through 29, 1993. These comments are documented in a meeting summary dated August 10, 1993. GE provided information to resolve these comments in revised Tier 1 material dated August 31, 1993, and provided supporting information in SSAR Amendment 32. In those submittals, GE modified the CDM Section 4.0 to provide consistent treatment of interface requirements for systems, clarify the in-scope and out-of-scope portions of the design, and modify the SSAR listing of systems so that it was consistent with the CDM.

Interfaces are discussed further in Section 14.3.4 of this report. Based on the discussion in Section 14.3.4, the staff found the treatment of interface requirements in the CDM acceptable. This resolved DFSER Open Item 14.1.4-1.

14.3.7.8 Site Parameters

Section 5.0, "Site Parameters," of the "Tier 1 Design Certification Material for the GE ABWR," was under staff review at the time the DFSER was issued. The staff stated that the evaluation would be provided in the FSER. This was DFSER Open Item 14.1.5-1.

The staff provided comments on site parameters in a letter to GE dated July 9, 1993. These comments were discussed during a meeting with GE on July 27 through 29, 1993, and were documented in a meeting summary dated August 10, 1993. GE provided information to resolve these comments in revised Tier 1 material dated August 31, 1993, and provided supporting information in SSAR Amendment 32. In those submittals, GE modified the CDM Section 5.0 to provide consistent treatment of site parameters with the site parameters and design basis analyses contained in the SSAR.

Site parameters are discussed further in Section 14.3.5 of this report. Based on the discussion in Section 14.3.5, the staff found the treatment of site parameters in the CDM acceptable. This resolved DFSER Open Item 14.1.5-1.

15 TRANSIENT AND ACCIDENT ANALYSIS

The staff reviewed the transient and accident analysis for the advanced boiling water reactor (ABWR) in accordance with Standard Review Plan (SRP) Section 15. Conformance with the acceptance criteria, except as noted for each of the sections, formed the basis for deciding if the design of the facility for each of the areas reviewed was acceptable.

Three groups of design-basis events are evaluated in this section: anticipated operational occurrences, anticipated operational occurrences involving common-mode software failure, and accidents. A conservative model of the reactor is used for the analysis of events in each group and all appropriate systems whose operations (or postulated misoperations) would affect the event are included. Anticipated operational occurrences (AOOs) are expected to occur during the life of the plant and are analyzed to ensure that they will not cause damage to either the fuel or to the reactor coolant pressure boundary. AOOs involving common-mode software failures have a lower probability of occurrence and are discussed in Section 15.2 of this report. Design-basis accidents (DBAs) are not expected to occur but are postulated because their consequences would include the potential for the release of significant amounts of radioactive material. These postulated accidents are analyzed to determine the extent of fuel damage expected and to ensure that reactor coolant pressure boundary damage, beyond that assumed initially to be the DBA, will not occur and that the radiological dose is maintained within 10 CFR Part 100 guidelines.

For loss-of-coolant accidents (LOCAs), the acceptance criteria for the emergency core cooling system (ECCS) specified in 10 CFR 50.46 are as follows:

- The peak cladding temperature must remain below 1204 °C (2200 °F).
- Maximum cladding oxidation must nowhere exceed 17 percent of the total cladding thickness before oxidation.
- Total hydrogen generation must not exceed 1 percent of the hypothetical amount that would be generated if all the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.
- The core must be maintained in a coolable geometry.
- Calculated core temperatures after successful initial operation of the ECCS must be maintained acceptably low, and decay heat must be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

The staff evaluation of GE Nuclear Energy (GE) LOCA analysis is given in Sections 15.4.4 and 6.3 of this report.

To demonstrate the adequacy of the plant's engineered safety features (ESFs), GE calculated the offsite consequences that could result from the occurrence of each of several DBAS and presented the results of these computations in the standard safety analysis report (SSAR).

15.1 Anticipated Operational Occurrences

AOOs which include infrequent and moderate frequency events are those transients expected to occur during normal or planned modes of plant operation. The acceptance criteria for these transients are based on GDC 10, 15, and 20. GDC 10 specifies that the reactor core and associated control and instrumentation systems shall be designed with appropriate margin to ensure that specified acceptable fuel-design limits are not exceeded during any condition of normal operation, including the effects of AOOs. GDC 15 specifies that sufficient margin shall be included to ensure that design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including AOOs. GDC 20 specifies that a protection system be provided that automatically initiates appropriate systems to ensure specified acceptable fuel design limits are not exceeded during any condition of normal operation including AOOs.

Acceptance criteria contained in Chapter 15 of the SRP for AOOs are as follows:

- Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design values according to American Society of Mechanical Engineers (ASME) Code, Section III, Article NB-7000. For ABWR, which has a design pressure of 8722 kPa (1250 psig), the pressure should not exceed 9584 kPa (1375 psig) during any AOO.
- Fuel-cladding integrity should be maintained by ensuring that the reactor core is designed with appropriate margin during any conditions of normal operation, including the effects of AOOs. For BWRs, the minimum value of the critical power ratio reached during the transient should be such that 99.9 percent of the fuel rods in the core would not be expected to experience boiling transition during core-wide transients. This limiting value of the minimum critical power ratio (MCPR), called the safety limit for the ABWR, is 1.07.
- An incident that occurs with moderate frequency should not generate a more serious plant condition unless other faults occur independently.

- An incident that occurs with moderate frequency in combination with any single active component failure, or operator error, should not result in loss of function of any barrier other than the fuel cladding. A limited number of fuel-rod-cladding perforations is acceptable. (See the discussion of Three Mile Island (TMI) Item II.K-3.44, NUREG-0737, in Chapter 20 of this report.)

For ABWR transient analysis, GE used the ODYNA computer code (proprietary) to simulate pressurization events and the REDYA computer code (proprietary) for other transient events.

The ODYNA code (proprietary) is designed to simulate selected transient conditions for the ABWR. Major features of this code are one dimensional description of kinetics and thermal-hydraulics of the core; reactor internal pumps (RIPs); option for two- and eight-node steamline model; and system models consisting of the core, bypass, and upper plenum. The core thermal-hydraulics is calculated using a five-equation formulation consisting of mass and energy balances for the vapor and liquid phases and momentum balance for the mixture. The code also contains models for ECCS, boron injection, and detailed control system for turbine control and pressure relief. The code includes modifications to the upper plenum model to allow for subcooled liquid.

The REDYA code (proprietary) is designed to simulate selected transient conditions for the ABWR. Major features of this code are point kinetics description of the core; lumped pressure calculation from a volume consisting of the core, bypass, and upper plenum; and multiple RIPs. Additionally, the code has models for internal separators; options for two- and eight-node steamline models; wide range of turbine control, pressure relief, and rod motion; reactor protection system (RPS) options; and boron injection models.

Both ODYNA and REDYA codes are similar to previous models, ODYN (NEDO-24154) and REDY (NEDO-10802), which the staff approved for transient analysis of operating BWRs. The ODYNA and REDYA versions have been revised to reflect the recirculation model and ECCS unique for the ABWR. The REDYA code also incorporated some model improvements already included in the ODYN code (e.g., safety/relief valve model and steamline model).

The recirculation model was the major modification in both ODYNA and REDYA codes. This new recirculation model consists of three groups of RIPs. RIP characteristics, initial conditions, and pump trip and runback functions are assumed to be identical for all RIPs in the same group. The performance of this recirculation model has been qualified against plant startup data obtained from European plants with similar RIPs. The events simulated for qualification purposes included pump coastdown during a trip of all pumps (data from two plants), during a trip of one pump, and during momentary voltage drop. GE compared the calculated results and plant data to verify the adequacy of the codes. GE also conducted model-to-model comparison and comparison with other test data (e.g., Peach Bottom turbine trip test) to ensure that the models perform correctly.

The staff, with the technical assistance of Brookhaven National Laboratory, audited ODYNA and REDYA during January 1992 at GE offices in San Jose, California. The staff audited three major areas: (1) formulation and models, (2) quality assurance (QA) procedures, and (3) verification and validation. The staff concluded that the modifications to the ODYN and REDY codes for the ABWR were adequately justified. It found the changes to ODYN and REDY to be acceptable. However, there was no documentation to verify that the coding changes to implement new models were independently checked. As stated in the DFSER, GE was to inform the staff, in writing, to confirm that the implementation of the code modifications had been independently verified as correct. This was DFSER Confirmatory Item 15.1-1. (Previously DSER SECY-91-355 Open Item 134.)

In the letter dated February 26, 1993, GE submitted the necessary information describing its verification process. The staff reviewed the documents governing GE QA requirements, and found that the code modifications were performed in accordance with the applicable design control provisions contained within the GE QA program and the GE design control procedures. Therefore, the staff has concluded that GE properly controlled the coding changes for ODYNA and REDYA. Thus, DFSER Confirmatory Item 15.1-1 is resolved.

The staff concludes that the ODYNA and REDYA computer codes are acceptable for design analysis of the ABWR.

In conducting its analyses of anticipated transients, GE used conservative assumptions with regard to initial power, scram reactivity, reactivity coefficients, and power profiles. It used conservative time delays to trip for each scram signal in the analyses. GE analyzed the following transients:

- decrease in core coolant temperature
- increase in reactor pressure
- decrease in reactor coolant system flow rate

- reactivity and power distribution anomalies
- increase in reactor coolant inventory
- decrease in reactor coolant inventory
- anticipated transient without scram (ATWS) (See Section 15.5 of this report.)

(1) Decrease in Core Coolant Temperature

Transients analyzed in this group included loss of feedwater heaters, feedwater control failure, runout of one feedwater pump, feedwater controller failure during maximum demand, opening of turbine control and bypass valves, pressure regulator failure in the open direction, inadvertent opening of a safety/relief valve, and inadvertent residual heat removal (RHR) shutdown cooling operation.

For transients categorized under Decrease in Core Coolant Temperature, the most severe transient is feedwater controller failure during maximum demand (runout of two feedwater pumps). The resultant minimum critical-power ratio (MCPR) reached is 1.07, and the peak vessel pressure is 1262 kPa (168 psi) below the ASME Code limit.

For the loss of feedwater heating transient, GE assumed a drop of 38 °C (68°F) in feedwater temperature, although a drop of 66 °C (119°F) occurred at a domestic BWR following an electrical component failure. In a letter dated January 10, 1992, GE stated that the drop of 66 °C (119°F) was a unique condition for that particular BWR design. The feedwater temperature will not drop as far in the ABWR design during this transient because the ABWR is designed so that no single failure of equipment can cause a temperature drop of more than 38 °C (68°F). The staff agrees with GE that the 66 °C (119 °F) drop was caused by a plant-specific design feature not present in the ABWR. Therefore, the 38 °C (68 °F) drop analyses is acceptable.

Inadvertent opening of a safety relief valve will cause a decrease in reactor coolant inventory and result in a mild depressurization event that will have only a slight effect on fuel thermal margins. Changes in surface heat flux are calculated to be negligible indicating an insignificant change in the MCPR. Thus, the transient response is acceptable and is bounded by the more severe feedwater controller failure transient.

GE initially inappropriately categorized the inadvertent RHR shutdown cooling operation event as an accident rather than an AOO. This was identified as DSER (SECY-91-355) Open Item 135, which was a significant deviation from the SRP. GE recategorized this event as a moderate frequency event (an AOO) and applied the appropriate acceptance criteria in the SSAR. The reanalysis of the event shows that the AOO acceptance criteria are met. This is acceptable and resolved DSER SECY-91-355 Open Item 135.

(2) Increase in Reactor Pressure

Transients in this group included generator load rejection and turbine trip with and without turbine bypass, inadvertent main steam isolation valves (MSIV) closure, loss of condenser vacuum, loss of auxiliary power transformer, loss of all grid connections, loss of all feedwater flow, and failure of RHR shutdown cooling.

The transient resulting in the highest system pressure was a generator load rejection without turbine bypass, which would result in a peak system pressure of about 1138 kPa (151 psi) below the allowable maximum pressure of 9584 kPa (1375 psig). In the overpressure protection report, the most limiting transient is an inadvertent MSIV closure with failure of the position switch scram (see Section 5.2.2 of this report). The reactor pressure increase for the transient turbine trip without bypass is bounded by the MSIV closure transient and is acceptable.

(3) Decrease in Reactor Coolant System Flow Rate

Transients in this group included trip of three RIPs, trip of all RIPs, runback of RIPs, and failure of the recirculation flow control to decrease flow.

Traditionally, loss of all forced circulation has been classified as a moderate frequency event and subject to the associated acceptance criteria. The ABWR will use motor-generator (MG) sets to power six of the ten RIPs. On loss of offsite power, the inertia of the MG provides a longer flow coastdown period. Analysis of this event shows that AOO acceptance criteria are met. However, the staff asked GE to consider the possibility of failures of other systems or components that could result in the loss of forced circulation without the availability of MG set coastdown. GE identified loss-of-flow transients both with and without offsite power available. The case with offsite power available was found more limiting because it did not result in an immediate reactor scram.

The most severe transient in this group is the simultaneous trip of all RIPs with offsite power available which is discussed in Section 15.2 of this report.

(4) Reactivity and Power Distribution Anomalies

Transients in this group included rod withdrawal error, abnormal startup of one RIP, fast runout of RIPs, and control rod misoperations.

The startup of an idle RIP is categorized under reactivity anomalies. This event is not a limiting transient and neither primary pressure boundary nor fuel damage criteria are exceeded.

(a) Rod Withdrawal Error at Low Power

GE examined the design of the rod control system to ascertain if a single failure can lead to the uncontrolled withdrawal of a control rod during refueling and during startup and low-power operation. During refueling operations, interlocks ensure that all control rods are inserted while fuel is being handled over the core. When no fuel is being handled, a maximum of one rod may be withdrawn. However, the control system is designed (see Section 4.3.2 of this report) so that the core is subcritical with the highest worth rod withdrawn. Finally, a control rod cannot physically be removed (from the top) without removing the four fuel assemblies that surround the rod. Therefore, GE has not provided an analysis of control rod removal error during refueling. This is in accord with approvals for current BWRs and is acceptable.

GE claims that the uncontrolled withdrawal of a rod during reactor startup is prevented by the rod block control system function of the rod control and information system (RCIS). This system enforces the banked position withdrawal sequence and precludes rod withdrawals other than those permitted in normal operation SRP Section 15.4.1 states that this transient need not be considered if single failures cannot cause the sequence. The single failure evaluation of the RCIS is given in Chapter 7 of this report. However, in accordance with staff requirements on current BWRs, GE also analyzed the erroneous withdrawal of a high worth control rod and found that the results fall well within the MCPR and other fuel criteria limits. This analysis is acceptable for the postulated event.

(b) Rod Withdrawal Error at Power

The rod withdrawal error transient can result from either a procedural error by the operator so that a gang of control rods is withdrawn continuously or from a malfunction of the automated rod withdrawal control logic during automated operation in which a gang of control rods is withdrawn continuously.

In the ABWR, the automated thermal limit monitor (ATLM) and the multi-channel rod block monitor (RBM) subsystem logic issues a rod block signal used in the RCIS logic to enforce rod blocks. This feature acts to prevent fuel damage by ensuring that the MCPR and maximum average planar linear heat generation rate (MAPLHGR) do not violate the fuel thermal operating and safety limits. The operating thermal limits rod block function will block rod withdrawal when the operating thermal limit is reached.

The rod block algorithms and set point are based on online core information (e.g., core flow and local power range monitor (LPRM) readings used to calculate the fuel status relative to limits). In response to staff questions and DSER (SECY-91-135) Open Item 137, GE presented (enclosures to its letter dated January 10, 1992, "GE Response to Agenda Item 12 Discussed During . . . . Meeting on November 20 and 21, 1991") a description of the algorithms, their development and bases, and examples of the functions (correlating MCPR or MAPLHGR with LPRM set points) programmed into the microprocessor-based ATLM for a given fuel and core design.

The functions for both MCPR and MAPLHGR are developed statistically with a database developed by analyzing control rod withdrawals for a wide range of initial power and flow conditions for a core design. These cases are used to provide a statistically derived bounding (95/95 probability and confidence) function for the rod block set point. The method is similar to the approach previously approved for BWR/5 rod block set point analysis. The functions will be updated for each fuel and core design used in the ABWR. This resolved DSER SECY-91-355 Open Item 137.

The RBM algorithms are simple relationships between power (LPRM readings) and operating limits (MCPR and MAPLHGR) and are reasonable and acceptable. The statistical approach is similar to previous staff-approved rod block methodology and is appropriate and conservative for use as proposed for the ABWR.

(5) Increase in Reactor Coolant Inventory

GE analyzed inadvertent startup of the high-pressure core flooder (HPCF) pump. (Feedwater flow control failure to maximum demand is covered in Category (1).)

The transient which could cause unplanned addition to coolant inventory is the inadvertent actuation of the HPCF system. The HPCF system actuation has little effect because its flow is small compared to the recirculation flow. Because the HPCF full flow is a small contributor to total core flow, the increase in total coolant inventory is also small. GE's analysis shows that the consequences of this small inventory increase has little effect on fuel thermal margins and reactor system pressure. In accordance with SRP Section 15.5.1, this is acceptable.

(6) Decrease in Reactor Coolant Inventory

The anticipated operational occurrence of the inadvertent opening of a safety relief valve is covered in Category (1) above.

GE indicated that non-safety-grade equipment is credited for the high water level 8 trip, use of turbine bypass valves, and recirculation pump trip on load/turbine trip.

The staff questioned the appropriateness of GE taking credit for equipment that is not safety grade in the transient analysis as stated in DSER (SECY-91-355) Open Item 138. GDC 1 through 4 require that components important to safety be designed to be commensurate with the quality standards, and GDC 21 requires that the protection system be designed for high functional reliability. GE listed in a table the redundancy, isolation, environmental, seismic, periodic testing, and QA requirements for the equipment.

Even though the equipment discussed above will not be categorized as safety grade, it is of high quality and has sufficient redundancy to ensure its operability. To ensure an acceptable level of performance for the ABWR, GE committed to identify the above equipment in the ABWR technical specifications (TS) with regard to availability, set points and surveillance testing. This was DFSER TS Item 15.1-1. GE included the level 8 trip, the RIP trip and the turbine bypass in the proposed ABWR TSs. This is acceptable and TS Item 15.1-1 is resolved.

By letter dated August 23, 1989, GE informed Gulf States Utilities Company of a condition that could be reportable under 10 CFR Part 21, applicable to the River Bend Station. This condition involves a slow closure of one main turbine control valve. This low probability event, which was not previously considered, results from a turbine control valve that GE assumes to close as a result of an unspecified failure in the turbine control circuit or in the servo-mechanism hardware. According to GE, if the valve closes in less than 2.3 seconds, a reactor scram will be initiated as a result of high neutron flux and no safety limits will be exceeded. However, if the valve closes in more than 2.3 seconds, the reactor scram will be initiated by high reactor pressure. During this slow-closure case, the MCPR safety limit may be exceeded if the maximum combined flow limiter is set for less than 113 percent of rated steam flow. GE based the consequences of this postulated event on its assessment of a generic BWR/6 analysis. This was identified as DSER (SECY-91-355) Open Item 139. The staff, however, requested that GE address the event for ABWR applicability. In response, GE performed an ABWR-specific analysis for the slow closure of one turbine control valve with the remaining three control valves remaining open. In this case, the neutron flux increase will not reach the high neutron flux scram set point. Since the available turbine bypass capacity will be high enough to bypass all steam flow not passing through the remaining three turbine control valves, the reactor power settles back to its steady state. (The total steam flow through three control valves will increase to about 85 percent, and the remaining 15 percent of flow will pass through the slow-opening control valve and the bypass valves.) During this transient, the peak fuel surface heat flux will not exceed 104 percent of its initial value. The MCPR remains above the safety limit and is acceptable. Therefore, DSER Open Item 139 was resolved.

15.2 Trip of All Reactor Internal Pumps and Pressure Regulator Down-Scale Failure

For the postulated trip of all of the RIPs with offsite power available, GE postulated a common-mode failure of the adjustable speed drives. GE estimated that a fraction of low burnup fuel rods will achieve boiling transition during this event although test results indicate no fuel failures would occur. The staff classified this postulated event in the special category of anticipated transients involving a common-mode software failure and established a special acceptance criterion for the radiological dose calculation. The staff will not require that fuel failure be assumed in dose calculations for fuel rods that are under approximately 600 °C (1111 °F) for less than 60 seconds. This time and temperature criterion is based on test data for fuel that has achieved up to 20 gigawatts days per metric ton (GWD/MtU) (18 GWD/t) burnup; thus, it may be applied only to fuel with burnup of less than 20 GWD/MtU (18 GWD/t). For fuel with greater burnup, the dose calculations must assume fuel failure for all fuel rods that achieve transition boiling. In the equilibrium cycle, the higher burnup fuel accounts for about 45 percent of the total fuel bundles. The power generated by these bundles is usually 20 percent less than that of the hottest bundles, and less than 0.2 percent of these rods are expected to enter transition boiling. Because none of the hottest fuel rods exceed the time and temperature failure criterion, the radiological dose requirements limit of 10 percent of 10 CFR Part 100 are satisfied.

For the pressure regulator down-scale failure to occur, all three channels would have to suffer a common-mode failure before the pressure regulator would go either up or down the scale. If the pressure regulator failed down-scale, the steam control valves would close causing the reactor pressure and reactivity to increase. When reanalyzing this postulated event, GE proposed to assume that any fuel rods that achieve transition boiling fail for the purposes of the radiological dose calculation. The staff includes this postulated event in the special category of anticipated transients involving a common-mode software failure because of the uncertainty that such an event will occur during the plant lifetime. GE originally categorized this event as an accident. The staff believes that it is more appropriate to apply a special classification for such an event. The staff required that GE demonstrate that this special event will not exceed the limits of 10 percent of 10 CFR Part 100, which the staff considers appropriate for an event of such postulated frequency.

According to GE analysis, during this event, it is estimated that less than 0.2 percent of fuel rods enter transition boiling and the requirement that the limit of 10 percent of 10 CFR Part 100 not be exceeded is met.

The staff will treat the above two postulated events as special cases, applicable only for the ABWR. This is due to the unique design features of the ABWR instrumentation and control systems, which reduce the frequency of such events; therefore, allowing these events to be recategorized as special cases. This resolved DSER (SECY-91-355) Open Item 136.

15.3 Accidents

GE analyzed RIP seizure and shaft break accidents. In the unlikely event of the pump motor shaft of 1 of the 10 RIPs stops instantaneously, a very rapid decrease of pump flow will result from the large hydraulic resistance introduced by the stopped rotor or shaft and cause pump seizure or shaft break. Consequently, core inlet flow and core cooling capability decreases. However, GE's analysis shows that with only 1 out of 10 RIPs seized, the core flow decrease is small (< 10 percent), so the event is mild. The RIP seizure and shaft break do not result in any fuel failure. This satisfies the dose limit criteria of 10 CFR Part 100 and is acceptable.

GE's analyses of the mislocated fuel bundle accident, misoriented fuel bundle accident, rod ejection accident, and control rod drop accident are discussed below.

(1) Mislocated Fuel Bundle Accident

Three errors must occur for this event to take place: (a) a bundle must be misloaded into a wrong location in the core; (b) the bundle, which was supposed to be loaded where the mislocation occurred, also is put in an incorrect location or discharged; and (c) the misplaced bundles are overlooked during the core verification process after core loading. A fuel loading error not detected by in-core instruments after fueling operations may result in an undetected reduction in thermal margin during power operations. However, GE evaluated the probability and consequences of a misplaced fuel bundle accident in equilibrium, first cycle, and subsequent cycle cores for current operating reactors and concluded that no fuel failure will occur and no radioactive material will be released from the fuel. The staff approved this analysis for operating plants and it is also applicable for the ABWR. This satisfies the criteria of 10 CFR Part 100 as required by the SRP for this event and is acceptable.

(2) Misoriented Fuel Bundle Accident

GE notified the staff by a 10 CFR Part 21 report (GE letter dated June 19, 1992, from S.J. Stark, "10 CFR Part 21, Reportable Condition, Rotated C or S-Lattice Fuel Assembly") that a fuel misorientation event may lead to fuel damage in BWR/6 designs. The staff required GE to evaluate the applicability of the issue for the ABWR and discuss its evaluation in the SSAR. This was DFSER Open Item 15.3-1.

For this event the fuel bundle is located correctly, but is rotated by 90 ° or 180 °, resulting in non-uniform adjacent water gaps. While this does not result in exceeding specified acceptable fuel design limits for some fuel bundle configurations during normal operation, for other configurations the limit might be exceeded. GE prepared a generic probabilistic risk model (GE letter dated September 30, 1992, from J.F. Klapproth, "Rotated Bundle Event Licensing Basis Change"), based on experience from recommended procedures for verifying fuel loading instituted in 1981, this model applies to current operating reactors and the ABWR, to demonstrate that the event may be classified as an accident. Since the staff had not completed the review of the GE probabilistic study, GE chose to reanalyze the misoriented fuel bundle event for a core loading with a bundle very similar to the reference fuel bundle design. GE made only slight modifications to the radial enrichment distribution to reduce the delta R factor. The analysis reflected that the energy capabilities were equivalent to the reference bundle design and the 15 percent thermal margin requirement was maintained. The infinite lattice void coefficient was also unchanged. This assures that the fuel safety limits are not exceeded for a misorientation event and is acceptable to the staff. Therefore, DFSER Open Item 15.3-1 is resolved.

(3) Rod Ejection Accident

The rod ejection accident is caused by a major break on the fine motion control rod drive (FMCRD) housing, outer tube, or associated control rod drive (CRD) pipe lines. The consequences of a rod ejection accident are similar to those of the rod drop accident in that the fuel enthalpy criteria may be violated if the speed of the ejected rod and/or the reactivity added are large enough. The same criterion of 280 cal/gm (1172 E+3 joules/kg) is applied to the rod ejection accident.

A redundant brake mechanism is installed in the FMCRD system (two brakes in each FMCRD) to prevent severe consequences resulting from this accident. Even if this accident does happen, the brake effectively terminates this event and prevents any severe consequences.

The radiological consequences are bounded by the analysis of the control rod drop accident. Therefore, the plant response is acceptable.

(4) Rod Drop Accident

The locking piston CRD mechanism in current BWRs cannot detect separation of the control rod from the drive mechanism during normal rod movements. Therefore, a latch mechanism is provided on the control rod to restrict the control rod free-fall distance to acceptable limits to prevent damage to the nuclear system process barrier by the rapid reactivity increase that would result from a free fall of a control rod (rod drop accident) from its fully inserted position to the position where the drive mechanism is withdrawn.

In contrast to the locking piston CRD, the FMCRD is designed to detect the separation of the control rod from the drive mechanism. Two redundant and separate Class 1E switches are provided to detect the separation of either the control rod from the hollow piston or the hollow piston from the ball nut. Actuation of either of these switches cause an immediate rod block and initiate an alarm in the control room, thereby reducing the probability of a rod drop accident from occurring. The radiological consequences of the control rod drop accident are provided in Section 15.4.1 of this report.

15.4 Radiological Consequences of Accidents

(Throughout this chapter there are statements indicating that 10 CFR Part 100 guidelines and criteria have been met. This means that the design basis accident being evaluated would result in a fission product release that would result in an exposure to an individual located in the plant's exclusion area for a period of two hours immediately after the release, of no more than 250 mSv (25 rem), or a whole body dose of no more than 3000 mSv (300 rem), or a total thyroid exposure from iodine of no more than 3000 mSv (300 rem). Further, a person located in the plant's low population zone for the duration of the release would receive no more than a whole body dose of 250 mSv (25 rem) or a total iodine dose to the thyroid of 3000 mSv (300 rem). These dose limits are listed in 10 CFR 100.11.)

In Chapter 15 of the SSAR, GE assessed the radiological consequence of the following six reactor DBAs (1) failure of small lines carrying primary coolant outside containment, (2) main steamline failure outside containment, (3) LOCA, (4) fuel handling accident, (5) spent fuel cask drop accident, and (6) reactor water cleanup system failure outside containment. GE concluded that the ABWR design using (1) reactor accident source term assumptions given in Technical Information Document (TID)-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," (2) certain ESF systems in the ABWR design, and (3) the bounding sets of atmospheric relative concentration values (χ/Q) which determine certain distances to the exclusion area boundary (EAB) and the low-population zone (LPZ) for a given site, are sufficient to provide reasonable assurance that the radiological consequences of such DBAs will be within the dose reference values established in 10 CFR Part 100 and the dose limits given in GDC 19 of Appendix A to 10 CFR Part 50. (A maximum whole body dose of 50 mSv (5 rem) or its equivalent to any part of the body for the duration of an accident).

To verify GE's conclusion, the staff independently assessed the radiological consequences of the above six DBAs and reactor control rod drop accident. In its assessments, the staff used assumptions and methods described in the SRP Section 15 and in Regulatory Guide (RG) 1.3 (Rev. 2), "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Boiling Water Reactors," and the bounding atmospheric relative concentration values for EAB and LPZ proposed by GE. The major parameters and assumptions used in the staff's radiological consequence assessments are listed in Tables 15.2 through 15.9 of this report.

In its evaluation of the radiological consequences in the DFSER, the staff used an EAB of 800 m (0.5 mi) and an LPZ ranging from 1500 (0.9 mi) to 4800 m (3 mi), with Pasquill F stability and persistent (greater than 95 percent of time) and a wind velocity of 1 m/sec (3 ft/sec). The staff also stated that the median values of EAB and LPZ of current operating reactor sites are 800 m (0.5 mi) and 4800 m (3 mi), respectively. In the DFSER, the staff accepted and used, in its radiological consequence assessments, the atmospheric relative concentration values corresponding to these distances as proposed by GE. GE calculated these values using the acceptable regulatory methodology given in RG 1.3.

Subsequently, as agreed to by the staff, GE proposed in Amendment 32 to the SSAR the use of the bounding atmospheric relative concentrations for EAB and LPZ for the ABWR design rather than median EAB and LPZ distances used by the staff in the DFSER. Table 15.5 of this report and in Table 15.6-13 of the SSAR list the bounding atmospheric relative concentration values proposed by GE and accepted by the staff in its radiological consequence assessments. The staff's recalculated offsite doses resulting from DBAs using the bounding atmospheric relative concentration values are listed in revised Table 15.1 of this report.

The atmospheric relative concentration for the 2-hour EAB was determined by the most limiting DBA (fuel-handling accident) not to exceed the dose acceptance criteria specified in the SRP Section 15.7.4 (750 mSv (75 rem) for the thyroid and 60 mSv (6 rem) for the whole-body doses).

The bounding atmospheric relative concentration values for LPZ (the 8-hour time period from 0 to 8 hours, the 16-hour period from 8 to 24 hours, the 3-day period from 1 to 4 days, and the 26-day period from 4 to 30 days) were also determined by the most limiting DBA (LOCA) not to exceed the dose reference values given in 10 CFR Part 100 (3000 mSv (300 rem) for the thyroid and 250 mSv (25 rem) for the whole-body). Two-hour LPZ atmospheric relative concentration and an annual average (8760 hours) concentration values were obtained by logarithmic interpolation of the calculated LPZ bounding atmospheric relative concentrations. In determination of these χ/Q values, GE followed the guidelines provided in RG 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," (Rev. 1). The bounding atmospheric relative concentration values for the EAB, 2-hour LPZ, and annual average (8760-hours) are specified in Table 5.0, "ABWR Site Parameters," of the ABWR Certified Design Material (Tier 1 Design Control Document) and in Table 2.0-1, "Envelope of ABWR Standard Plant Site Design Parameters," of the SSAR.

The staff will independently assess χ/Q values for potential accident consequence assessments on a site-specific basis using onsite meteorological data (hourly cumulative frequency distributions) submitted by a COL applicant in accordance with RG 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants." In its evaluation, the staff will use the guidance provided in (1) RG 1.23, "Onsite Meteorological Programs," (2) RG 1.145, (3) RG 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluent in Routine Releases from Light-Water-Cooled Reactors," and (4) SRP Section 2.3.4, Short-Term Dispersion Estimates. As discussed in Section 2.3.4 of this report, this was DFSER COL Action Item 2.3.4-1.

If site-specific atmospheric relative concentration values are greater than the bounding values (e.g., less dispersion) used in this report, a COL applicant may have to augment ABWR ESF systems to meet the relevant requirements of 10 CFR Part 100 and GDC 19. However, this condition is not expected to arise frequently since the bounding values for atmospheric relative concentrations envelope most of the current operating reactor site meteorological characteristics.

Table 15.1 Revised radiological consequences of design-basis accidents

	EAB, mSv (rem)		LPZ, mSv (rem)	
Postulated Accident	Thyroid	Whole Body	Thyroid	Whole Body
Loss of coolant				
Containment leakage				
00- 02 hours	331 (33)	40 (4)	38 (4)	3 (0.3)
02- 08 hours			198 (20)	11 (1)
08- 24 hours			244 (24)	15 (1.5)
24- 96 hours			417 (42)	6 (0.6)
96-720 hours			248 (25)	2 (0.2)
Exfiltration			172 (17)	6 (0.6)
Total containment leakage	331 (33)	40 (4)	1317 (132)	43 (4)
Emergency core cooling systems component leakage	20 (2)	10 (1)	72 (7)	<10 (1)
Main steam isolation valve leakage	125 (13)	10 (1)	1302 (130)	<10 (1)
Total	476 (48)	60 (6)	2691 (269)	63 (6)
Main stealine failure outside containment				
With concomitant iodine spike	110 (11)	<10 (1)	13 (1)	<10 (1)
With preaccident iodine spike	270 (27)	<10 (1)	30 (3)	<10 (1)
Rod drop accident	6 (0.6)	<10 (1)	20 (2)	<10 (1)
Fuel-handling accident	750 (75)	<10 (1)	20 (2)	<10 (1)
Small line break accident	46 (5)	<10 (1)	<10 (1)	<10 (1)
Reactor water cleanup line break				
With concomitant iodine spike	92 (9)	<10 (1)	13 (1)	<10 (1)
With preaccident iodine spike	225 (23)	<10 (1)	30 (3)	<10 (1)

Note: EAB = exclusion area boundary
LPZ = low population zone.

Table 15.2 Assumptions used to compute rod drop accident doses

Parameter	Value
Power level	4005 MWt (4.2 E+6 BTU/sec)
Peaking factor	1.55
Number of fuel rods perforated	770
Number of fuel rods melted	6
Condenser leak rate	1.0%/day
Fraction of fission product inventory release to coolant from melted fuel rods	
Iodines	50%
Noble gases	100%
Fraction of fission product inventory released to coolant from perforated fuel rods	
Iodines	10%
Noble gases	100%
Iodine fraction released to condenser	10%
Iodine fraction available for release from condenser after plate-out and partitioning	10%
Atmospheric diffusion values	
0-2 hour, exclusion area boundary	1.37E-3 sec/m^3 (3.88 E-S sec/ft^3)
0-8 hour, low population zone	1.56E-4 sec/m^3 (4.42 E-6 sec/ft^3)

Table 15.3 Assumptions used to evaluate the main steamline and small line break accidents outside the containment

Parameter	Value
Mass of primary coolant released before main steam isolation valve closure	
Steam mass released	1.29E+4 kg (2.8E+4 lb)
Water mass released	2.2E+4 kg (4.84E+4 lb)
Mass of primary coolant released through small line	5.5E+3 kg (1.2E+4 lb)
Mass of primary coolant flashed	2.3E+3 kg (5E+3 lb)
Fraction of iodine in the primary coolant released	100%
Fraction of noble gases in the primary coolant released	100%
Primary coolant concentration (dose equivalent I-131)	
Technical specification limit, normal long-term operation	7.4E-3 mBq (0.2 μCi/gm)
Technical specification limit, normal short-term operation	1.5E-1 mBq (4.0 μCi/gm)
Atmospheric diffusion values	
0-2 hour, exclusion area boundary	1.37E-3 sec/m^3 (3.88 E-5 sec/ft^3)
0-8 hour, low population zone	1.56E-4 sec/m^3 (4.42 E-6 sec/ft^3)

Table 15.4 Assumptions used to evaluate the loss-of-coolant accident

Parameter	Value
Power level	4005 MWt (4.2E+6 BTU/sec)
Fraction of core inventory released	
Noble gases	100%
Iodines	50%
Iodines initial plate-out fraction	50%
Iodine chemical species	
Elemental	91%
Particulate	5%
Organic	4%
Suppression pool decontamination factor	
Noble gas	1
Organic iodine	1
Elemental iodine	2
Particulate	2
Pool bypass	50%
Iodine dose conversion factors	ICRP-30*
Primary containment leakage	0.5%/day
Main steam isolation valve leakage (total)	3.96 m^3/hr (140 ft^3/hr)
Standby gas treatment system	
Filter efficiency	97%
Flow rate	113 m^3/min (4000 ft^3/min)
Drawdown time	20 minutes
Emergency core cooling system (ECCS) system leakage	3.785 L/min (1.0 gpm)
Iodine core inventory in suppression pool	0.5 fraction
Suppression pool water volume	3.785E+6 m^3 (1E+6 gal)
Iodine partition factor for ECCS leak	0.1
Primary containment free volume	1.33E+4 m^3 (4.7E+5 ft^3)
Secondary containment free volume	8.5E+4 m^3 (3E+6 ft^3)
Secondary containment mixing efficiency	50%

* ICRP-30 International Commission on Radiation Protection Publication 30.

Table 15.5 Atmospheric dispersion (χ/Q) values used in accident evaluations

Time Period	χ/Q Value (sec/m^3) (Sec/ft^3)
0-02 hour EAB	1.37E-3 (3.88 E-5)
0-08 hour LPZ	1.56E-4 (4.42 E-6)
8-24 hour LPZ	9.61E-5 (2.72 E-6)
1-04 day LPZ	3.36E-5 (9.51 E-7)
4-30 day LPZ	7.42E-6 (2.10 E-7)

Note: EAB = exclusion area boundary
LPZ = low population zone.

Table 15.6 Method to evaluate iodine removal in main steamlines and main condenser

Parameter	Value
Source term	Regulatory Guide 1.3
Main steam isolation valve leakage rate	3.96 m^3/hr (140 ft^3/hr)
Leakage duration	30 days
Condenser volume	9.85E+8 cc (3.47 E+4 ft^3)

	Deposition Temperature (K) (F)	Velocities (cm/sec)	Resuspension Rate (in./sec)	Fixation Rate (L/sec)	Fixation Rate (L/sec)
Elemental and particulate iodine	300 (81)	3.2E-2	1.2 E-2	3.14E-6	4E-6
	400 (261)	5.0E-3	2.0 E-3	7.05E-6	8E-6
	500 (441)	1.0E-3	3.9 E-4	8.10E-6	1E-5
	560 (549)	6.2E-4	2.4 E-4	9.20E-6	2E-5
Organic iodine	300 (81)	1.4E-3	5.5 E-4	9.5E-8	4E-6
	400 (261)	3.5E-4	1.4 E-4	2.0E-7	8E-6
	500 (441)	1.0E-5	3.9 E-6	3.0E-7	1E-5
	560 (549)	1.3E-6	5.2 E-7	3.6E-7	2E-5

Component	Inside Diameter (cm) (in.)	Length (cm) (ft)	Thickness (cm) (in.)
Main steamline	64 (25)	4773 (157)	2.5 (1.0)
Drain line	6.7 (2.6)	610 (240)	1.1 (.4)

Table 15.7 Iodine releases* GBq (curies)

Hours	Inlet to Main Steamlines	Inlet to Main Condenser	Outlet From Main Condenser
00 - 002	4.8E+05 (1.3E+04)	1.6E+04 (4.2E+03)	8.9E+02 (2.4E+01)
02 - 008	1.3E+06 (3.5E+04)	1.7E+04 (4.6E+03)	3.5E+03 (9.5E+01)
08 - 024	1.8E+06 (4.9E+04)	1.7E+04 (4.6E+03)	1.8E+04 (4.8E+02)
24 - 096	6.7E+06 (1.8E+05)	5.9E+05 (1.6E+04)	2.0E+05 (5.4E+03)
96 - 720	2.1E+07 (5.6E+05)	1.7E+06 (4.6E+05)	1.1E+06 (2.9E+04)

*Based on total MSIV leakage rate.

Table 15.8 Assumptions used in computing fuel handling accident doses

Parameter	Value
Power level	4,005 Mwt
Peaking factor	1.55
Number of fuel rods damaged	124(1) 5,120(2)
Number of fuel rods in cask	1,116
Filter iodine removal efficiencies:	
Organic	99%
Elemental	99%
Shutdown Times	24 hours(1) 7 days(2) 120 days(3)
Inventory released from damaged rods:	
Iodine and noble gases	10%
Kr-85	35%
Iodine fraction:	
Organic	0.25
Elemental	0.75
Atmospheric diffusion values	(sec/m^3) (sec/ft^3):
0-2 hour, exclusion boundary	1.37E-3 (3.88 E-5)
0-8 hour, low population zone	1.56E-4 (4.42 E-6)

(1) single fuel assembly dropping
(2) heavy object dropping
(3) fuel cask dropping

Table 15.9 Assumptions and estimates of the radiological consequences to control room operators after a loss-of-coolant accident

Parameter	Value	
Control room free volume	7,000 m^3 (2E+5 ft^3)	
Recirculation rates		
Filtered intake	1.8 m^3/sec	
Unfiltered intake	0.0	
Filtered recirculation	0.8 m^3/sec	
Filter efficacy	95%	
Unfiltered control room infiltration rate (assumed)	4.7E-3 m^3/sec (1.0 E-1 ft^3/min)	
Duration of accident	30 days	
Breathing rate of operators in control room for course of accident	3.47 E-4 m^3/sec (1.22 E-2 ft^3/sec)	
Meteorology (wind speeds for all sectors		
00 - 008 hours	4.0 E-3 sec/m^3 (1.1 E-4 sec/ft^3)	
08 - 024 hours	2.4 E-3 sec/m^3 (6.8 E-5 sec/ft^3)	
24 - 096 hours	1.5 E-3 sec/m^3 (4.3 E-5 sec/ft^3)	
96 - 720 hours	6.6 E-4 sec/m^3 (1.9 E-5 sec/ft^3)	
Iodine protection factor	27	
Iodine dose conversion factors*	ICRP-30**	
Iodine reduction factor for dual air intake	4	
Control room operator occupational factors		
00 - 008 hours	1	
08 - 024 hours	1	
24 - 096 hours	0.6	
96 - 720 hours	0.4	
Doses to control room operators	Thyroid dose* mSv (rem)	Whole-body dose*** mSv (rem)
00 - 008 hours	10 (1)	10 (1)
08 - 024 hours	30 (3)	3 (0.3)
24 - 096 hours	100 (10)	1 (0.1)
96 - 720 hours	130 (13)	1 (0.1)
Total	270 (27)	15 (1.5)

* Unweighted dose equivalent.
** ICRP-30 International Commission on Radiation Protection Publication 30.
*** Unweighted dose equivalent - red bone marrow.

The bounding atmospheric relative concentrations eliminate the need for either the COL applicant with the certified ABWR design or the staff to assess the offsite radiological consequence assessments for DBAs for the reactor site proposed if a COL applicant can demonstrate that (1) its atmospheric relative concentrations at the proposed ABWR reactor site are less than the bounding values used in this report, and (2) its design characteristics of main steamlines, steam drain lines, and condenser can remove equal or greater amounts of iodine than that claimed in the staff assessment in this report.

15.4.1 Control Rod Drop Accidents

In SSAR Section 15.4.9.6, GE states that the radiological consequences of a control rod drop accident need not be considered because such an accident is extremely unlikely with the improved design of the ABWR. The ABWR design employs the FMCRD system, which has several new features not found in current BWR locking piston CRDs.

In SSAR Section 15.4.9.2, GE states that for the rod drop accident to occur, it is necessary for such highly unlikely events as failures of both Class 1E separation-detection devices or the failure of the rod block interlock, and the failure of the latch mechanism to occur simultaneously with the occurrence of a stuck rod on the same FMCRD. GE further states that, therefore, there is no basis to postulate the occurrence of this event because of the low probability of such simultaneous occurrence of these multiple independent events.

The staff considered past licensing reviews, such as Clinton, Perry, and River Bend, and concluded that a control rod drop accident for the ABWR design results in radiological consequences less than a small fraction of the dose reference values specified in 10 CFR 100.11 even with conservative assumptions. SRP Section 15.4.9 (III) states that a specific calculation of the radiological consequences for this accident is not necessary unless unusual plant or site features are present, or the applicant's calculation shows an unusually large amount of fuel damage. However, the staff specifically evaluated this accident because it is the first application involving the ABWR standard design with hypothetical site boundaries. This evaluation should establish a reference for comparison of future applications incorporating the ABWR design.

To evaluate the radiological consequence of this accident, the staff postulated that the highest worth control rod becomes decoupled from its drive mechanism at a fully inserted position in the core. The drive mechanism is withdrawn, but the decoupled control rod is assumed to be stuck in place. At a later moment, the control rod suddenly falls free and drops out of the core. This results in the insertion of a large positive reactivity into the core, causing a localized power excursion. An automatic safety feature would terminate this excursion with required operator action. The rod pattern control function of the RCIS limits the worth of any control rod by regulating the withdrawal sequence.

The staff estimated that such a rod drop would cause no more than 770 fuel rods to reach the threshold for cladding damage, with 6 fuel rods melting. Table 15.2 of this report lists the assumptions used for estimating this fuel damage which are consistent with those given in Appendices A (Physics and Thermal-Hydraulics) and B (Radiological Assumption) of RG 1.77, "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors," (Rev. 0). The computed doses are listed in Table 15.1 of this report and are well within the dose reference values of 10 CFR 100.11. Therefore, the staff concludes that the ABWR design is adequate to control the release of fission products following a postulated control rod drop accident.

15.4.2 Failure of Small Lines Carrying Primary Coolant Outside Containment

GDC 55 contains provisions to ensure isolation of all pipes carrying reactor coolant that penetrate the containment building. Exempted from these specifications are small-diameter pipes (instrument lines) that must be continuously connected to the primary coolant system in order to perform necessary functions. The design must include methods of mitigating the consequences of a rupture of an instrument line because the lines cannot be automatically isolated. GE submitted a radiological analysis for an instrument line failure in SSAR Section 15.6.2.5, and the results of the analysis are provided in SSAR Table 15.6.3.

GE postulated that a small steam or liquid line breaks inside or outside the primary containment and that a small instrument line, instantaneously and circumferentially, breaks at a location where it may not be able to be automatically isolated and where detection is not automatic or apparent.
GE estimated that 5,448 kg (12,000 lbs) of primary coolant would be released through the break before it is isolated. GE estimated that 2,270 kg (5,000 lbs) of the 5,448 kg released would flash to steam and be available for release.

While conducting past licensing reviews, such as Clinton, Perry, and River Bend, the staff determined that a small line break accident is expected to result in radiological consequences less than a small fraction of the dose reference values specified in 10 CFR 100.11. Furthermore, the staff believes that these postulated breaks are subsumed by the design-basis LOCA radiological consequences, as stated in SRP Section 15.6.2. However, the staff did perform a specific evaluation of this accident because this application is the first involving the ABWR standard design with hypothetical site boundaries. This evaluation should establish a reference for comparison of future applications incorporating the ABWR design.

The assumptions used for the evaluation are listed in Table 15.3 of this report. The computed doses are listed in Table 15.1 and are well within the dose reference values of 10 CFR 100.11. Therefore, the staff concludes that the ABWR design is adequate to control the release of fission products following a postulated small line break accident.

15.4.3 Main Steamline Failure Outside Containment

GE postulated one of the four main steamlines to rupture between the outer isolation valve and the turbine control valves. GE analyzed this hypothetical accident in SSAR Section 15.6.4.5 and concluded that no more than 34,800 kg (76,770 lb) of reactor coolant would be lost through the break before automatic isolation and less than 12,870 kg (28,373 lb) of that would be lost as steam. The results of the GE analysis are provided in SSAR Table 15.6.7.

The staff accepted GE's estimated value and calculated the potential radiological consequences assuming 34,000 kg (74,957 lb) of reactor coolant are released directly to the environment. The staff assumed that 100 percent of the iodine and noble gases present in the released coolant are released to the atmosphere within 2 hours as stated in RG 1.5, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Steamline Break Accident for Boiling Water Reactors," (Rev. 0). Other assumptions are listed in Table 15.3 of this report.

In Section 3.6 of this report, the staff evaluated the ABWR design for protection against postulated piping failures in fluid systems outside the containment. The staff's evaluation of the protection provided against adverse environmental effects (excluding radiation effects) resulting from postulated piping failure is discussed in Section 3.11 of this report.

The staff assumed two reactor coolant conditions for the evaluation. In Case 1, the lost coolant was contaminated with radioactive iodine at the limits included in the standard technical specifications (STS) (i.e., 7.4E-3 mBq (0.2 μCi) of dose-equivalent I-131 per gram) for BWRs at continued full-power operation. In Case 2, a concentration of 1.5E-1 mBq (4.0 μCi) of dose-equivalent I-131 per gram was assumed (the limits in the STSs above which the reactor is required to be shut down). The SRP acceptance criteria are the dose reference values of 10 CFR 100.11 for Case 2, and less than 10 percent of these values for Case 1. Dose-equivalent I-131 is defined as any mixture of iodine isotopes yielding the same inhalation thyroid dose as the stated amount of pure I-131. The staff also considered the amounts of 13 noble gas isotopes that also would be released.

The major parameters and assumptions used in the staff's radiological consequence assessment are given in Table 15.3 of this report. The calculated doses are listed in Table 15.1 and are within the acceptance criteria of SRP Section 15.6.4. Therefore, the staff concludes that the ABWR design is adequate to control the release of fission products following a postulated steamline break accident.

15.4.4 Loss-of-Coolant Accidents

In SSAR Section 15.6.5, GE selected and analyzed a hypothetical design-basis LOCA and concluded that certain bounding sets of atmospheric relative concentration values in conjunction with the use of ESF systems provided in the ABWR design are sufficient to provide reasonable assurance that the radiological consequences of such an accident will be within the dose reference values established in 10 CFR Part 100.

To verify GE's conclusion, the staff independently assessed the radiological consequences of a hypothetical LOCA. In its assessment, the staff used assumptions and methods described in the appendices to SRP Section 15.6.5 and in RG 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Boiling Water Reactors," (Rev. 2). The major parameters and assumptions used in the staff's evaluation are listed in Table 15.4 of this report.

Since no specific site is associated with the ABWR plant, GE previously defined these two boundaries only in terms of various hypothetical atmospheric relative concentrations at fixed EAB and LPZ distances. Subsequently, GE proposed and the staff accepted the bounding χ/Q values as stated in Section 15.4 of this report. Using these bounding χ/Q values, the recalculated offsite doses resulting from a hypothetical LOCA are listed in Table 15.1. The computed doses in Table 15.1 are expressed as thyroid and whole-body exposure for 2 hours at the EAB and for 30 days at the boundary of the LPZ to allow direct comparison with the dose reference values established in 10 CFR 100.11.

The staff evaluated offsite radiological consequences using the current TID-14844 source term assumptions that are consistent with the guidelines in the applicable SRP sections and regulatory guides, except for the following two deviations: The staff (1) provided a credit for radioactive iodine removal in the main steamlines and in the main condensers by holdup for decay and deposition and (2) accepted the ABWR design without an MSIV leakage control system (LCS) (see Section 15.4.4.2 for technical bases).

The staff postulated a hypothetical LOCA to determine the adequacy of the ESF systems designed to prevent release of fission products into the environment, using bounding meteorological conditions. The passive ESF systems for this purpose in the ABWR design are the primary containment and a secondary containment (reactor building). The staff considered these in conjunction with the standby gas treatment system (SGTS) and the pressure suppression pool scrubbing, both of which are active ESF systems.

The primary containment spray system, which is designed as a containment heat removal feature, has not been given credit in any drywell airborne fission product removal calculations. The spray system is designed as a safety-related system but is not automatically initiated as required in the SRP for iodine removal credit. GE has not requested any iodine removal credit for the spray system. In its assessment, the staff included containment leakage, main steam isolation valve leakage, and post-LOCA leakage from ESF systems outside containment as sources and radioactivity transport paths to the environment following a LOCA.

15.4.4.1 Containment Leakage Contribution

The ABWR primary containment design consists of a drywell, a wetwell, and supporting systems to limit fission product leakage during and following a LOCA with rapid isolation of all pipes or ducts that penetrate the containment boundary. It is designed to prevent the uncontrolled release of airborne radioactivity to the environment. GE proposed that the primary containment will be built and tested periodically to have a leak rate at design pressure of less than 0.5 percent by weight per day at the calculated peak containment pressure associated with a LOCA. The staff used this leak rate in its radiological assessment.

The secondary containment structure and supporting systems will collect and process radioactive material that may leak from the primary containment following a LOCA or that may result from a fuel-handling accident. The secondary containment is a seismic Category I reinforced-concrete structure with a volume of approximately 8.5E+4 m^3 (3E+6 ft^3).

In SSAR Section 15.6.5.5.1, GE assumed that the primary containment leak rate into the reactor building through penetrations and ESF system components will not be greater than an equivalent release of 0.5 percent by weight per day of the primary containment free air weight for the first 24 hours after a LOCA and half of that value (0.25 percent/day) after 24 hours. GE based its assumption of a reduced leak rate on the finding that the primary containment pressure is reduced by more than a factor of 1/2 during the 12 hours following initiation of a LOCA; therefore, the driving force for leakage through the pathway is correspondingly reduced.

RG 1.3 (Rev. 2) assumes a constant containment leak rate for the duration of a LOCA, although it permits a reduced leak rate with supporting justification. Two rationales support a constant containment leak rate: (1) the pressure profile for a BWR does remain high for a long period of time and (2) for most plants, the leakage is only measured at the maximum value in accordance with Appendix J to 10 CFR Part 50. Therefore, the primary containment leak rate was identified as Open Item 143 in the DSER (SECY-91-355). Subsequently, GE revised the primary containment leak rate of 0.5 percent by weight per day for the entire duration of a LOCA, accepting the staff's position in a draft revision to the SSAR. This was identified as DFSER Confirmatory Item 15.4.4.1-1. In Amendment 24 to the SSAR Section 15.6.5.5.1, GE revised the leak rate to 0.5 percent by weight per day for the duration of a LOCA and this resolved DFSER Confirmatory Item 15.4.4.1-1.

The pressure within the reactor building is maintained slightly negative during normal operation by exhausting the reactor building air through the normal reactor building ventilation system. On receipt of an ESF actuation signal, the normal ventilation system is automatically switched off and the SGTS actuated. Following a postulated LOCA, the pressure in the secondary containment could increase from its initially negative pressure to a slightly positive value as a result of inleakage, air expansion because of heat, and the time required for the startup of the SGTS.

The SGTS is designed to achieve a negative pressure of 0.635 cm (0.25 in.) water gauge in the secondary containment within 20 minutes (drawdown time). Following the drawdown time, the staff assumed an iodine removal efficiency of 97 percent by the SGTS charcoal absorber with a depth of 15.24 cm (6 in.). In accordance with SRP Section 6.5.3, the staff assumed reactor building air mixing efficiency of 50 percent for the primary containment leakage before the release to the environment.

The staff estimates that wind speed in excess of 10 m/s (33 ft/sec) may create potential exfiltration of the reactor building due to the low atmospheric pressure created outside the reactor building. The loss of the SGTS for the entire period of an accident (720 hours) will increase the offsite radiological consequences by a factor of 100, while the increased wind speed will improve atmospheric dispersion parameters by a factor of 33 based on the meteorological data submitted by GE on November 17, 1992. In the submittal, GE also stated that wind speed will exceed 10 m/s (33 ft/sec) no more than 5 percent of the time, based on meteorological data obtained from 28 BWR sites in the United States. Therefore, the loss of the SGTS for 36 hours (5 percent of 720 hours) could increase the offsite radiological consequences by 15 percent. This increase is reflected in revised Table 15.1 of this report.

Because the secondary containment (reactor building) is designed to completely enclose the primary containment, the staff assumed no bypass leakage to the environment from the primary containment except that directed through the main steamlines. In Section 6.2.3.1 of the DSER (SECY-91-355), the staff concluded that GE adequately addressed the criteria in Branch Technical Position CSB 6-3 of the SRP and that the design of barriers to preclude the secondary containment bypass flow (excluding that through main steamlines) is acceptable.

The fundamental characteristic of an BWR pressure-suppression containment is that steam released from the reactor coolant system will be condensed and scrubbed of radionuclides in a pool of water (the suppression pool) and the pressure rise in the containment will thereby be limited. This is accomplished by directing the steam from the reactor coolant system to the suppression pool through a vent system. However, leakage paths could exist in the pathway between the drywell and the wetwell airspace that could allow steam to bypass the suppression pool, potentially overpressurizing the containment. Potential sources of steam bypass include leakage through the vacuum relief valves, cracking of the drywell structure, and penetrations through the drywell structure.

In SSAR Section 15.6.5.5.1, GE assumed that any elemental and particulate iodine species purged to the suppression pool would be subject to a decontamination factor (DF) of 10. In the DSER (SECY-91-355), the staff found that credit may be given for the removal of fission products by the suppression pool if suppression pool DFs are evaluated in accordance with the methodology prescribed in revised SRP Section 6.5.5, "Pressure Suppression Pools or Fission Product Cleanup Systems" (issued in December 1988), which states that suppression pools are capable of scrubbing airborne fission products and that it is unduly conservative to ignore this capability. However, the staff also stated in the DSER (SECY-91-355) that GE should provide the drywell leakage capability to justify the suppression pool DF of 10. The suppression pool DF was identified as Open Item 141 in the DSER (SECY-91-355).

In the DFSER, the staff conservatively assumed, in its radiological consequence analysis, that a DF of 2 is provided by the ABWR pressure suppression pool (equivalent to suppression pool steam bypass of 50 percent) for airborne radioactive iodine in elemental and particulate forms. This assumption is further supported by safety-related drywell and wetwell containment sprays in the drywell or wetwell, or both, which also would reduce the effect of suppression pool bypass leakage on containment performance; therefore, the staff stated in the DFSER that this DSER Open Item 141 was resolved. In Amendment 31 to the SSAR, GE accepted the staff position amending Section 15.6.5.5.1.1 of SSAR to reflect a DF of 2 by the suppression pool for airborne radioactive iodine in elemental and particulate forms. This item remains resolved.

The staff recently accepted the drywell leakage value of (.05 m^2) (.06 yd^2) of effective leakage pathway) proposed by GE for the ABWR design. GE submitted a letter dated August 31, 1993, enclosing a revised set of design description and ITAAC on the suppression pool bypass issue. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report.

While containment sprays in the drywell and/or wetwell also would reduce the effect of suppression pool bypass leakage on containment performance lowering its temperature and pressure, the sprays also scrub the containment atmosphere of fission products (even though the staff has not given a credit for scrubbing) and mitigate the effects of bypass on fission product distribution.

The staff will review, for each COL applicant, plant-specific TS, which require periodic inspections to confirm suppression pool depth, and surveillance tests to confirm drywell leak tightness. In DFSER, the staff identified this as TS Item 15.4.4.1-1. GE included this inspection requirement in the ABWR STS for such surveillance tests. Therefore, this item is resolved. The calculated doses for the ABWR resulting from the containment leak of 0.5 percent by weight per day are shown in Table 15.1 of this report.

15.4.4.2 Main Steam Isolation Valve Leakage

The main steamlines in BWR plants contain dual quick-closing main steam isolation valves (MSIVs) which isolate the reactor system in the event of a break in a steamline outside the primary containment, a design-basis LOCA, or other events requiring containment isolation. Although the MSIVs are designed to provide a leaktight barrier, it is recognized that they allow some leakage. The current BWR TS limit for MSIV leakage is typically 0.325 m^3/hr (11.5 ft^3/hr) per valve. Operating experience has indicated that degradation has occasionally occurred in the leak-tightness of MSIVs and that the valves may exceed their specified leakage limit.

Because of recurring problems with excessive leakage of MSIVs, the staff recommended in RG 1.96, "Design of Main Steam Isolation Valve Leakage Control Systems for Boiling Water Reactor Nuclear Power Plants," (Rev. 1), installation of a supplemental leakage control system (LCS) to ensure that the isolation function of the MSIVs complies with the specified limits. Most of the currently operating BWRs have an LCS.

In response to the MSIV leakage concerns, the BWR Owners Group (BWROG) commissioned a program of studies to determine the causes of high leak rates and the means to eliminate them. The results of these studies were submitted to the Nuclear Regulatory Commission (NRC) in General Electric proprietary reports, NEDO-31643P (November 1988) and NEDO-31858P (February 1991), both entitled, "Increasing Main Steam Isolation Valve Leakage Rate Limits and Elimination of Leakage Control Systems." GE referenced these reports as the basis for not including an LCS and requesting a substantially higher (0.99 m^3/hr (35 ft^3/hr) per valve) MSIV leakage limit.

The MSIVs generally have not provided a leak-tight containment pressure boundary to the extent intended in the plant design. Although substantial progress has been made in recent years to identify the causes of the leakage and reduce the amount of leakage, the current typical TS limit of 0.325 m^3/hr (11.5 ft^3/hr) per valve is still difficult to achieve when the valve is rapidly closed against a full-flow condition at reactor operating pressure and temperature.

The staff conservatively assumes for operating plants that the TS leakage limit of 0.325 m^3/hr (11.5 ft^3/hr) per MSIV is released directly into the environment in calculating offsite radiological consequences of potential accidents (10 CFR Part 100). No credit is currently given for the integrity and leaktightness of the main steam piping and condenser to provide holdup and plate-out of fission products.

The BWROG proposed an approach adopted by GE which would allow higher leakage limits of 3.96 m^3/hr (140 ft^3/hr) total for four steamlines and would not include an LCS for the ABWR design. In this alternative approach, GE used the main steam piping (and its associated piping) and the condenser to mitigate the radiological consequences of an accident that could result in offsite exposures comparable to the dose reference values specified in 10 CFR Part 100.

In the DSER (SECY-91-355), the staff stated that it was evaluating whether a credit for the fission product attenuation in the main steamlines and for the condenser is appropriate and reasonable for BWRs even though the main steamlines downstream from the MSIV and its condensers are not designed to withstand the safe shutdown earthquake (SSE), as defined in Appendix A, Section III(c), to 10 CFR Part 100. This was identified as DSER (SECY-91-355) Open Item 142. The staff completed its evaluation, and has accepted the BWROG and GE proposals for the ABWR design based on the following radiological and seismic analyses.

15.4.4.2.1 Radiological Analysis

- MSIV Leakage Pathways

  Following a LOCA, three potential release pathways exist for main steam leakage through the MSIVs:

  (1) main steam drain lines (typically 5.08 cm (2 in.) in diameter) to the condenser with delayed release to the environment through the low-pressure turbine seals

  (2) turbine bypass lines (typically 15.2 cm (6 in.) in diameter) to the condenser with delayed release to the environment through the low-pressure turbine seals

  (3) main steamline (typically 61 cm (24 in.) in diameter) turbine stop and control valves through and high-pressure turbine seals to the environment bypassing the condenser

  The consequences of leakage from pathways 1 and 2 will be essentially the same because the condenser can be used to process MSIV leakage. The condenser iodine removal efficiency will vary depending on the inlet location of the bypass or drain line piping, but in either case, iodine removal will occur. However, for pathway 3, MSIV leakage through the closed turbine stop and control valves will not be processed via the condenser. For this case, iodine will be removed in the high-pressure turbine, which has a large internal surface area associated with the turbine blades for plate-out of the radioactive material.

The staff assumed that turbine bypass, stop, and control valves will be closed upon loss of the electro-hydraulic control system following a LOCA, and the drain valve can be manually opened to provide a leakage pathway from the control room to the condenser via a safety-related power source following a LOCA. GE submitted on August 31, 1993, a revised set of Tier 1 design descriptions of the nuclear boiler system. In Section 2.1.2 and corresponding Figure 2.1.2b of this document, GE stated that the main control room will include main steam drain valve controls and status displays. The staff finds that the remote control requirement placed in the ABWR Tier 1 document is acceptable. The overall adequacy and acceptability of the ABWR Tier 1 design descriptions are evaluated in Chapter 14.3 of this report.

In the DFSER, the staff stated that the TS must address operability of the drain valve. Subsequently, the staff agreed with GE that the operability of the drain valve need not to be included in the ABWR STSs but should be included in the plant operating procedures. This position makes GE's TS consistent with the BWR STS developed under the Commission's TS improvement program. See the general discussion in Chapter 16 of this report. Therefore, this resolved DFSER TS Item 15.5.1.2.1-1.

The BWROG has also identified the same three leakage pathways as discussed above. Of these, the BWROG and GE proposed to use the drain line downstream of the MSIVs as a leakage pathway to the condenser. The staff also chose to use the main steamline drains for the MSIV leakage pathway in assessing iodine holdup and deposition for the ABWR design.

The BWROG and GE indicated that the bypass piping of the main steamline is another appropriate leakage pathway that can be used for MSIV leakage control. Either pathway is an acceptable approach, provided that the chosen pathway will be available for MSIV steam leakage to reach the main condenser. Comparing the two pathways, more iodine is removed (by holdup and deposition) through the turbine bypass pathway than through the drain line pathway because of the larger pipe size, which will have more surface area for deposition.

The staff believes that as long as either the turbine bypass or the drain line leakage pathway is available, MSIV leakage through the closed turbine stop and control valves (pathway 3) will be negligible and essentially all of the releases will be through the main condenser. This is because there will be essentially no differential pressure in the main steamline downstream of the MSIVs following the closure of the valves.

Furthermore, any MSIV leakage through pathway 3, if any, will have been subjected to the same iodine removal processes in the main steamlines (up to turbine stop valves) as the other pathways, and it will be further subjected to iodine removal by deposition in the high-pressure turbine internal surfaces. The main condenser does not remove iodine in pathway 3.

In calculating the contribution to the LOCA dose, the staff assumed that one of the inboard isolation MSIVs failed to close, thus allowing contaminated steam to travel to the outboard valve. This outboard valve and the outboard valves from the other three steamlines were assumed to have a total leak rate of 3.96 m^3/hr (140 ft^3/hr).

- <u>Iodine Transport Model</u>

Basic chemical and physical principles indicate that gaseous iodine and airborne iodine particulate material will deposit on surfaces. Several laboratory and in-plant studies have demonstrated that gaseous iodine deposits will occur by chemical adsorption, and particulate iodine deposits will occur through a combination of sedimentation, molecular diffusion, turbulent diffusion, and impaction. Gaseous iodine exists in nuclear power plants in several forms: elemental (I_2), hypoiodous acid (HIO), organic (CH_3), and particulates. In accordance with RG 1.3, the staff assumed 91 percent of iodine will be in the elemental form (inhypoiodous acid), 5 percent in the particulate form, and 4 percent in the form of organic iodides.

Each of these forms deposits on surfaces at a different rate, described by a parameter known as the deposition velocity. The elemental iodine form, being the most reactive, has the largest deposition velocity, and organic iodide has the smallest. Further, studies of in-plant airborne iodine show that elemental and particulate iodine deposited on the surface undergoes both physical and chemical changes and can either be resuspended as an airborne gas or become permanently fixed to the surface. The data also shows that the iodine can change its form so that iodine deposited as one form (usually elemental) can be resuspended in the same or in another form (usually organic). Conversion can be described in terms of resuspension rates that are different for each iodine species. Chemical surface fixation can similarly be described in terms of a surface fixation rate constant.

The transport of gaseous iodine in elemental and particulate forms has been studied for many years and several groups have proposed different models to describe the observed phenomena. Examples of the studies are listed below.

(1) NUREG/CR-2713, "Vapor Deposition Velocity Measurement and Consolidation for Iodine and Cesium Iodine," S.L. Nicolosi and P. Baybutt, May 1982.

(2) NUREG/CR-4397, "In-Plant Source Term Measurements at Prairie Island Nuclear Generating Station," J.W. Mandler et al., September 1985.

(3) IN-1394, Idaho Nuclear/National Reactor Testing Station, "Deposition of Iodine 131 in CDE Experiments," Nebecker at al., 1969.

(4) BMI-1863, Fission Product Deposition and Its Enhancement Under Reactor Accident Condition: Deposition on Primary System Surfaces, J.M. Genco et al., May 1969.

(5) "Transmission of Iodine Through Sampling Lines," 18th DOE Nuclear Airborne Waste Management and Air Cleaning Conference, P.J. Unrein et al., October 1984.

The staff used the model developed by an NRC contractor (J.E. Cline and Associates, Inc., 1991) for iodine removal in BWR main steamlines and the main condenser following a LOCA.

The staff model treats the MSIV leakage pathway as a sequence of small segments for which instantaneous and homogeneous mixing is assumed; the mixing computed for each segment is passed along as input to the next segment. The number of segments depends on the parameter of the line and flow rate and can be as many as 100,000 for a long, large-diameter pipe and a low flow. Each line segment is divided into five compartments that represent the concentrations of the three airborne iodine species, the surface that contains iodine available for resuspension, and surface iodine that has reacted and is fixed on the surface. The staff's model considers three iodine species: elemental, particulate, and organic. A fourth species, hypoiodous acid, is considered for the purpose of the staff's model to be a form of elemental iodine. All iodine in the segment undergoes radioactive decay. The resulting concentration from each segment of the deposition compartment serves as the input to the next segment.

The staff's transport model also assumes iodine transport through the condenser as a dilution flow rather than the plug flow as in the steamlines. The staff assumes that the iodine entering the condenser mixes instantaneously with a volume of air in the condenser and that the diluted air exhausts at the same time and same rate as the input air (MSIV leakage) flows into the condenser.

The staff developed the equations for iodine deposition velocities, resuspension rates, and surface fixation rates as a function of temperature using published data from the above-mentioned literature. The equations and data are contained in the report by Cline and Associates. The equation for the deposition velocity of elemental iodine is based on the least-squares fit to the available data. Deposition velocity equations for HOI and organic iodine are based on the values at 30 °C (86 °F); because of the lack of data at elevated temperatures, their temperature dependence is assumed to be similar to that of elemental iodine. The staff based its resuspension and fixation equations as a function of temperatures available in the literature based on measurements taken at ambient temperature. The staff assumed that resuspension and fixation rates will increase with increasing temperature.

The parameter and assumptions used in the development of the iodine transport model are listed in Table 15.6 of this report. Calculated iodine releases from the condenser after holdup and plate-out in the main steamlines and condensers are shown in Table 15.7 of this report.

The technical references mentioned above and the staff's model indicate that particulate and elemental iodine would be expected to deposit on surfaces at rates varying with temperature, pressure, gas composition, surface material, and particulate size. The staff, therefore, concludes that an appropriate credit for the removal of iodine in the main steamlines and main condensers should be provided in the radiological consequence assessment following a design-basis accident for the ABWR. The staff considers DSER Open Item 142 resolved.

The amount of iodine removal credit for the ABWR design is shown in Table 15.7 of this report. In the DFSER, the staff stated that the COL applicant will need to recalculate removal credit on the basis of its design characteristics of main steamlines, drain or bypass line, and main condenser. This was identified as COL Action Item 15.4.4.2.1. In Sections 15.6.5.5.1.2 and 15.6.5.5.1.3 of the SSAR, GE stated that COL applicants will recalculate iodine removal credit on the basis of its design characteristics of main steamlines, drain lines, and main condenser. The staff finds this acceptable.

15.4.4.2.2 Seismic Analysis

Section III(c) and VI of Appendix A to 10 CFR Part 100 require that structures, systems, and components (SSCs) necessary to ensure the capability to mitigate the radiological consequences of accidents that could result in exposures comparable to the dose guideline exposures of Part 100 be designed to remain functional during and after an SSE. Thus, the main steamline, portions of its associated piping, and the main condenser are required to remain functional if the SSE occurs. Consequently, these components are required to be classified as safety-related and seismic Category I. In addition, Appendix A to 10 CFR Part 100 requires that the engineering method used to ensure that the safety functions are maintained during and after occurrence of an SSE involve the use of either a suitable dynamic analysis or a suitable qualification test.

For the purpose of giving credit to iodine holdup and plate-out in the main steamlines and condensers, the staff's model requires that the main steam piping (including its associated piping to the condenser) and the condenser to remain structurally intact following an SSE, so it can act as a holdup volume for fission products.

In the DFSER, the staff stated that the ABWR design did not fully comply with the requirements of 10 CFR Part 100 because portions of the main steamlines, bypass and drain piping, and the condenser were not classified as seismic Category I. This DFSER statement was incorrect. The staff's position as discussed in detail in Section 3.2.1 of the DFSER and of this report is that the ABWR design provides reasonable assurance that the main steam piping from the outmost isolation valve up to the turbine stop valve, the MSIV leakage pathway (i.e., the drain line or bypass line) up to the condenser, and the main condenser will remain structurally intact and leaktight, so that they can act as a holdup volume for fission product during and following an SSE. The staff has determined that the ABWR design does comply with the requirements of 10 CFR Part 100 because the SSCs described above are designed to remain functional during and following a SSE. This issue is also discussed in the staff's SECY-93-087 entitled, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs," dated July 21, 1993.

Open Item 15.4.4.2.2-1 in the DFSER indicated that the staff would have to describe new ABWR design requirements which would enable it to meet 10 CFR Part 100 in a different manner or require that GE request exemption from 10 CFR Part 100 and provide an evaluation of the request in the SSAR. Section 3.4.1 of this report describes the key design requirements for the ABWR that resolve DFSER Open Item 15.4.4.2.2-1.

15.4.4.3 Post-LOCA Leakage Contribution from Engineered Safety Features Systems Outside Containment

Any leakage of water from ESF components (valve stems and pump seals) located outside of the primary containment releases fission products to the secondary containment during the recirculation phase of long-term core cooling following a LOCA. GE estimated such leakage from the RHR system, HPCF system, and reactor core isolation cooling (RCIC) system to be less than 6.2E-3 L/min (1.6E-3 gpm). GE's leakage estimate indicated valve stem leakage of 4.5E-3 L/min (1.2E-3 gpm) from a total of 45 valves and 1.7E-3 L/min (4.4E-4 gpm) from a total of 5 pumps in these systems. In Section 5.2.5 of this report, the staff states that an identified leakage within the drywell (inside of the primary containment) could reach 5.8 m^3/hr (25 gpm) at which point it would activate an alarm in the control room.

The staff used a conservative leakage value of 3.8 L/min (1.0 gpm) (instead of 6.2E-3 L/min (1.6E-3 gpm) proposed by GE), which is the typical BWR leakage value used by the staff in its radiological consequence assessment. The COL applicant will provide this leakage value in its plant-specific TS. Section 17.6.3 (limiting condition for operation (LCO) 3.4.3 of the ABWR TS) requires unidentified reactor coolant system leakage to be less than 1 gpm. This resolved DFSER TS Item 15.4.4.3-1.

The staff further assumed that 10 percent of the waterborne iodine that leaks from the ECCSs (RHR, HPCF, and RCIC) systems will become airborne and be released to the environment through the SGTS after 20 minutes of the secondary building drawdown time (no filtration credit is given prior to the drawdown time). The offsite radiological consequences of an ESF component leak outside the primary containment are relatively small and are given in Table 15.1 of this report.

15.4.4.4 Conclusion

The staff reviewed GE's analysis and performed an independent analysis of the radiological consequences resulting from each of the transport paths described in Section 15.4.4 of this report. The calculated thyroid and whole-body doses are listed in Table 15.1. Based on the above evaluation and the recalculated radiological consequences shown in Table 15.1, the staff concludes that the MSIV leak rate limit of 3.96 m^3/hr (140 ft^3/hr) total and the proposed ABWR design without an MSIV LCS are acceptable. It further concludes that GE's proposed bounding atmospheric relative concentrations (χ/Q) for the EAB and to the LPZ of the ABWR plant, in conjunction with the ESF systems provided in the ABWR design are sufficient to provide reasonable assurance that the radiological consequences of a postulated LOCA will be within the dose reference values in 10 CFR Part 100.

15.4.5 Fuel Handling and Fuel Cask Drop Accidents

In SSAR Sections 15.7.4 and 15.7.5, GE presented radiological consequence analyses of fuel handling and spent fuel cask drop accidents. The staff analyzed the radiological consequences for the following three different types of fuel handling accidents resulting from an unspecified failure of a lift mechanism due to (1) a single fuel assembly with 124 spent fuel rods dropped onto the irradiated fuel stored in the spent fuel pool, (2) a raised single fuel cask containing 18 fuel assemblies in it dropped from the level of the refueling floor to ground level through the refueling floor maintenance hatch, and (3) a raised heavy object (i.e., steam dryer, moisture separator) dropped onto the fuel in the reactor vessel during refueling operation.

For the single fuel assembly accident, the kinetic energy of a single falling fuel assembly was assumed to break open the maximum possible number of fuel rods (124 spent fuel rods) using perfect mechanical efficiency. Instantaneous release of noble gases and radioiodine vapor from the gaps of the broken rods (10 percent of noble gases and iodine inventories in the reactor core except 30 percent assumed for krypton-85) was assumed to occur, with the released gases bubbling up through the fuel pool water (with an overall effective DF of 100 for iodines and of 1 for noble gases) Radiation monitors located within the normal ventilation system are designed to isolate that system automatically and direct all fuel handling building exhaust to the SGTS.

The SGTS is designed to achieve a negative pressure of 0.635 cm (0.25 in.) water gauge in the secondary containment within 20 minutes (drawdown time). During the drawdown time, the staff assumed that radioactive iodines would be released directly to the environment without credit for filtration. Following the drawdown time, the staff assumed the SGTS charcoal absorber with a depth of 15.24 cm (6 in.) would remove iodine from the released gas at a 99 percent efficiency.

The fuel cask drop accident was assumed to result from an unspecified failure of the cask-lifting mechanism, thereby allowing the cask to fall approximately 19.5 m (64 ft) from the level of the refueling floor to ground level through the refueling floor maintenance hatch. GE stated that each cask will have the maximum capacity of 1116 spent fuel rods (18 fuel assemblies) based on the largest capacity cask projected to be available. GE proposed, and the staff accepted, the minimum fuel storage (decay) time of 120 days before cask-loading operation commences after reactor fueling. The minimum storage time of 120 days is based on the administrative controls currently used by operating BWRs.

In NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," the staff discusses the potential for an accidental load drop on nuclear fuel or safety-related equipment causing excessive offsite radioactivity releases, inadvertent criticality, loss of water inventory in the reactor or spent fuel pool, or loss of safe-shutdown equipment. The NUREG also recommends guidelines to prevent or mitigate these potential consequences and states that the guidelines in NUREG-0554, "Single-Failure-Proof Cranes for Nuclear Power Plants," can be followed in lieu of upgrading existing crane and lifting devices. NUREG-0554 provides guidance for the design, fabrication, installation, and testing of new cranes that are of a high reliability design and it defines that a single-failure-proof crane system as a system designed so that a single failure will not result in the loss of the capacity of the system to safely retain the load.

In Section 9.1.5.2.1 of the SSAR, GE states that the reactor building crane will be designed to meet the single-failure-proof requirements of NUREG-0554, and in Section 9.1.6.6 of the SSAR, GE states that the COL applicant should provide, among other things, heavy load handling system operating and equipment maintenance procedures for NRC review. This review will confirm that a heavy load drop is not a design-basis event by determining that the crane system meets (1) single-failure-proof criteria, and (2) prevention of load unbalancing (e.g., improper placement of slings) which could potentially defeat the single-failure-proof criteria.

For the heavy load drop accident to occur, the following steps must occur in sequence: (1) a heavy load is slung wrong (procedural), (2) the heavy load is transported to an incorrect position over the core (procedural), (3) the sling fails (equipment failure), and (4) an alternate slung hook fails (equipment failure).

Nevertheless, the staff assumed in the DFSER that the heavy load drop accident occurs because the lifting device fails (unspecified cause), drops, or tips a heavy object onto the fuel in the reactor vessel and breaks open all fuel rods in 10 percent of fuel assemblies (approximately

80 assemblies) in the reactor vessel. The staff assumed the minimum radioactivity decay time of 7 days from the time of reactor shutdown until the lifting of the heavy object based on a conservative BWR refueling schedule.

The staff evaluated a postulated fuel handling accidents for the ABWR in accordance with the guidance of SRP Section 15.7.4, using assumptions consistent with Positions C.1.a through C.1.k of RG 1.25 (Rev. 0), "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling Storage Facility for Boiling and Pressurized Water Reactors," (Rev. 0).

The assumptions used for the spent fuel handling accidents are listed in Table 15.8. The offsite doses recalculated for these accidents are given in Table 15.1. The calculated offsite doses for all three cases considered are within the dose guidelines specified in SRP Section 15.7.4, specifically, the estimated doses resulting from this type of accident are less than or equal to 25 percent of the 10 CFR Part 100 dose limits (listed in Section 15.4 above). Therefore, the staff concludes that the standard ABWR design is adequate to control the release of fission products following postulated spent fuel handling accidents.

15.4.6 Postulated Radioactive Releases Resulting from Liquid Tank Failures

The staff reviewed this accident in accordance with SRP Section 15.7.3. Tanks and associated components containing radioactive liquids outside containment are acceptable if the failure of the equipment does not lead to radioactive concentrations that exceed the limits of 10 CFR Part 20, Appendix B, Table II, Column 2, at the nearest potable water supply in an unrestricted area or if special design features to mitigate the effects of the accident are incorporated in the design of systems that do not meet the requirements of 10 CFR Part 20.

All the liquid radwaste tanks, including the evaporator concentrate tanks, that could adversely affect the potable water supply if they fail are located in the radwaste building. The base mat and outside walls of the building are seismic Category I to a height necessary to retain spill liquids within the building (SSAR Section 11.2.1.2.2). Additionally, in accordance with SSAR Section 15.7.3.1, all compartments containing liquid radwastes are steel lined up to a height capable of containing the release of all the liquid radwastes into the compartments. For the above reasons, GE considers it as remote that any major accident that involves the release of liquid radwastes into these volumes would result in the release of liquid radwastes to the environment via the liquid pathway. In SSAR Section 15.7.3.1, GE states that initially the releases would be contained in other holding tanks or emergency tanks. GE further states that plant operating techniques and administrative procedures will contain detailed system and equipment operating instructions and, therefore, will significantly minimize the potential for operator error that can cause liquid radwaste release. In addition, the liquid radwaste system design will include a positive action interlock system to prevent inadvertent opening of a drain valve. GE concludes that if a release of liquid radwaste occurs, the steel lining would contain the release until the operator could use the floor door sump pumps to pump the release into holding tanks or emergency tanks.

GDC 60 requires the control of releases of radioactive materials to the environment including both gaseous and liquid effluents under both normal and anticipated operational occurrences. Further, sufficient holdup capacity is required. As discussed above, GE's liquid radwaste design provides the necessary control and liquid effluent storage capacity to reduce the potential effect of a failure of a radioactive liquid containing tank and its associated components. Therefore, GE's design complies with GDC 60 and any potential release associated with a liquid tank failure will not result in radionuclide concentrations in water exceeding the limits of 10 CFR Part 20, Appendix B, Table II, Column 2, in any unrestricted area. The staff finds this aspect of the design acceptable.

15.4.7 Reactor Water Cleanup System Pipe Break Accident

In Section 15.6.6 of the SSAR, GE postulated a reactor water cleanup system pipe break outside the primary containment as a DBA in response to an inquiry by the Advisory Committee on Reactor Safeguards (ACRS) Subcommittee on ABWRs during its meeting on June 17, 1993, at San Jose, California. The break was assumed to be instantaneous, circumferential, and to occur on the downstream side of the outmost containment isolation valve but on the upstream side of the reactor water demineralizer. GE further assumed 75 seconds of break flow time (45-second builtin delay time for flow differential pressure instrumentation to activate an isolation signal and 30 seconds for the motor-operated isolation valve to close).

GE analyzed this hypothetical accident and concluded that no more than 28,800 kg (61,670 lb) of reactor coolant would be lost through the break before automatic isolation occurred and less than 9,900 kg (21,800 lb) of that would be lost as steam. The break line is equipped with a 140-cm² (23 in.²) flow limiter. The staff accepted and used in its assessment the reactor coolant break flow of 28,800 kg estimated by GE.

The current SRP neither lists nor classifies the reactor water cleanup system pipe break accident as a DBA. However, the staff assessed the radiological consequences of this accident assuming two reactor coolant conditions for the evaluation. In Case 1, the lost coolant was contaminated with radioactive iodine at the limits in the STS (i.e., 7.4E-3 mBq) of dose-equivalent I-131 per gram (90.8 mCi of dose-equivalent I-131 per pound) at continued full-power operation. In Case 2, a concentration of 1.5E-1 mBq of dose-equivalent I-131 per gram (1816.0 mCiq dose-equivalent I-131 per pound) was assumed (the limits in the STSs above which the reactor is required to be shut down).

The staff chose the dose reference values of 10 CFR 100.11 for Case 2 and less than 10 percent of these values for Case 1 as two acceptance criteria. These are the same as those given for the main steamline break outside containment in SRP Section 15.6.4. Dose-equivalent I-131 is defined as any mixture of iodine isotopes yielding the same inhalation thyroid dose as the stated amount of pure I-131. The staff also considered the amounts of 13 noble gas isotopes that also would be released.

The calculated doses are listed in Table 15.1 and are within the staff's acceptance criteria listed above. Therefore, the staff concludes that the ABWR design is adequate to control the release of fission products following a postulated reactor cleanup system pipe break accident.

15.5 Anticipated Transient Without Scram

15.5.1 Design Features

The ABWR design incorporates electric-hydraulic fine motion control rod drives (FMCRDs) to perform motor-driven scram and hydraulic scram. In response to a scram signal, the control rods will be inserted hydraulically by means of the stored energy in the scram accumulator, similar to the current operating BWR CRDs. A scram signal also will be given simultaneously to insert the FMCRD electrically via the FMCRD motor drive. This diversity, hydraulic and electric methods of scramming, provides a high degree of assurance of rod insertion on demand.

The ABWR is designed with an alternate rod insertion (ARI) system that will be independent from the existing reactor protection system (RPS) from the sensor output to the final actuation device. The ARI system will have redundant scram air header exhaust valves. The ARI system is designed to perform its function in a reliable manner. Detailed evaluations of the ARI and RPS systems are given in Chapter 7 of this report.

The ABWR also is designed with a standby liquid control system (SLCS) that will automatically inject 379 L/min (100 gpm) of sodium pentaborate solution into the reactor pressure vessel (RPV) with the simultaneous operation of both pumps. The 326 L/min (86 gpm) equivalency specified in the ATWS rule (10 CFR 50.62) for the 638-cm (251-in.) RPV is satisfied by the 379 L/min (100 gpm) provided for the 706-cm (278-in.) ABWR vessel. The ABWR SLCS initiation is automatic as required by the ATWS rule and is designed to perform its function in a reliable manner. The detailed evaluation of SLCS is given in Section 9.3.5 of this report.

ABWR has equipment to trip the RIPs automatically under conditions indicative of an ATWS. The RIPs are automatically tripped on reactor high pressure (7,860 kPa (1125 psig)) and RPV Level 2. The RIP trip equipment is designed to perform its function in a reliable manner as required by the ATWS rule.

The ABWR design also provides recirculation runback for all scram signals and feedwater runback on reactor high pressure and startup range neutron monitoring system (SRNM) not downscale for 2 minutes. Automatic depressurization system automatic inhibit is also provided with reactor water level 1.5 and average power range monitor ATWS permissive. This feature is discussed in Section 7.4.1.1 of this report.

The ABWR complies with the prescriptive design requirements of the ATWS rule and is designed to mitigate the effects of an ATWS event.

15.5.2 Analysis

GE submitted the ATWS analysis in SSAR Appendix 15E. GE analyzed the limiting transients identified in NEDO-24222 (proprietary) (a supporting document used for the ATWS rule). GE used the NRC-approved ODYNA and REDYA computer codes for the analysis. GE analyzed cases with the ARI system operational, without the ARI system but with FMCRD system operational, and without the ARI and FMCRD systems but with SLCS operational. GE compared the results with the performance guidelines for fuel integrity (coolable core geometry), containment integrity (45 psi (414 kPa)), suppression pool temperature (207 °F (97 °C)), primary system pressure (1,500 psig (10446 kPa)) and long-term shutdown cooling and found them acceptable.

The staff performed audit calculations to verify that the ABWR design is satisfactory to mitigate the effects of an

ATWS. The study focused on the consequences of manual SLCS actuation and no recirculation pump runback on scram signals other than reactor high pressure and reactor low level. One problem that was identified initially was the potential for fuel damage if only the FMCRD system provides slow scram (motor-driven scram) and there were no other actions to lower power. In this case, the power shifted to the top of the core when the control rods entered from the bottom and, in combination with the delay in shutting down the reactor because of the slowness of the FMCRD system, could lead to higher linear power densities and the potential for fuel damage. If the use of the FMCRD system was always in combination with a reactor internal pump trip or recirculation runback, then overall power would be reduced sufficiently to avoid excessive power densities when the power shifted to the top of the core.

The new design feature of recirculation runback on any scram signals or any ARI/FMCRD run-in signals ensures that there is no potential for any unacceptable power shift to the top of the core.

Another feature included of the initial study was the time available for SLCS actuation. Assuming the failure of ARI and FMCRD, the original design called for manual operation of SLCS; however, if the system was not started within a few minutes, containment integrity would be threatened. GE resolved this issue by changing the ABWR design so that the SLCS is automatically started at 3 minutes into an ATWS event.

The results of calculations to evaluate the thermal hydraulic stability of ABWR under the recirculation runback and feedwater runback conditions associated with ATWS events have been provided for staff review. ABWR design response to ATWS enables avoidance of large oscillations and the staff audit of stability calculations is complete. This is discussed in detail in Section 4.4 of this report (DFSER Open Item 4.4-1).

DSER (SECY-91-355) Open Items 140 and 144 are dependent on and are superseded by the DFSER Open Item 4.4-1. Since Open Item 4.4-1 has been resolved the DSER open items are also resolved.

16 TECHNICAL SPECIFICATIONS

16.1 Introduction

As part of its design certification application for the advanced boiling water reactor (ABWR) and as required by paragraph (1)(i) of 10 CFR 52.47(a), GE must provide Technical Specifications (TS) that are technically relevant and not site-specific for the ABWR design. The TS provided must comply with the requirements of paragraph (b)(6)(vi) 10 CFR 50.34, which states that proposed TS are to be prepared in accordance with the requirements of 10 CFR 50.36. 10 CFR 50.36 details the specific items (such as safety limits, limiting safety system settings, limiting control settings, limiting conditions for operation, etc.) that must be included in the TS.

In reviewing proposed TS for compliance with 10 CFR 50.36, the staff evaluates the conformance of the proposed TS to standard technical specifications (STS) which most closely resemble the plant design.

The staff review of the ABWR TS was closely coupled to the development of the improved STS under the TS Improvement Program in accordance with the interim Commission Policy Statement on TS Improvements for Nuclear Power Plants (52 FR 3788 dated February 6, 1987). Since the ABWR design evolved primarily from the BWR/6 design, most of the ABWR TS were modeled after NUREG-1434, "Standard Technical Specifications - General Electric Plants, BWR/6." However, since the ABWR containment most closely resembles plant containments covered by NUREG-1433, "Standard Technical Specifications - General Electric Plants, BWR/4." the ABWR containment TS were modeled after NUREG-1433. These improved STS reflect the accumulated operating experience of currently operating light water reactors.

The proposed ABWR TSs were developed from the BWR/6 and the BWR/4 STS. The staff reviewed the proposed ABWR TS to confirm similarities between them and the STS, as appropriate. The staff then concentrated its review on the parts of the ABWR TS that are unique because of ABWR-specific design features. Dispositions of comments resulting from the staff review were incorporated into the proof-and-review ABWR TS. The proof-and-review ABWR TS were then issued to GE and made available to the staff for comment. Dispositions of comments from GE and the staff on the proof-and-review TS have been incorporated into the final ABWR TS. The final ABWR TS will be produced in the industry format and certified as accurate by GE.

In the DFSER, the staff stated that GE should prepare the relevant portions of the TS for review by the staff in the WordPerfect 5.1 format. This was DFSER Open Item 16-1. The NRC TS staff assisted GE in developing the ABWR proof-and-review TS in the WordPerfect 5.1 format. GE committed to maintain the ABWR TS in this format. This resolved Open Item 16-1.

16.2 Evaluation

The staff evaluated the ABWR TS to confirm that they will preserve the validity of the design plant SSAR by ensuring that ABWR plants will be operated within the required conditions bounded by the SSAR and with operable equipment that is essential to prevent accidents and to mitigate the consequences of accidents postulated in the SSAR.

In the DFSER, the staff noted that for the ABWR design, GE had attempted the use of a three-subsystem concept for the engineered safety feature (ESF) systems in order to facilitate maintenance on one subsystem by extending the completion times (CTs) for one inoperable subsystem from 7 days to 30 days. GE was asked to provide justification that the emergency core cooling systems design consists of three independent redundant subsystems. This was DFSER Open Item 16-3. GE stated that it could not provide this justification. Instead, it stated that the ABWR design includes additional redundancy, beyond that for currently operating BWR plants, for the ESF systems. The staff reviewed GE's response and determined that, although this added redundancy does not allow extending the CT for an inoperable system from 7 to 30 days, it does allow extending the associated CTs beyond those specified for ESF systems in currently operating BWR plants. The staff's finding on the relaxed CTs, ranging from 8 hours to 14 days, is based on probabilistic risk evaluation, engineering evaluation, and operating experience and judgment for various components and combinations of components. The staff noted that these evaluated configurations match the configurations of inoperable components delineated in the TS limiting conditions for operation. The rationale for the relaxed CTs is given in Chapter 19 of the SSAR and the TS bases. On the basis of the above, DFSER Open Item 16-3 is resolved.

The ABWR instrumentation and controls (I&C) systems design concept incorporates microprocessor-based digital technology and multiplexed fiber optic signal transmission. TS limits for the reactor protection system are based on a four-sensor channel design. This requires that the applicable portions of the BWR/6 STS be restructured to address the unique ABWR data collection and transmission design features. In the DFSER, the staff asked GE to submit the ABWR TS so that it could complete its review of the ABWR I&C TS. This was DSER Open Item 16-2.

As part of the resolution of DFSER Open Item 16-1 noted previously, the I&C TS were issued by the staff to GE for comment. After meetings during which GE's comments were resolved, the staff issued the ABWR TS to GE for certification. This resolved DFSER Open Item 16-2.

In the DFSER, the staff stated that the COL applicant should include plant- and site-specific information in the ABWR TS. This was DFSER COL Action Item 16-1. SSAR Chapter 16, "Technical Specifications," contains guidelines enabling the COL applicant to complete the plant- and site-specific portions of the TS on the basis of as-procured hardware and software. This is acceptable.

As part of the TS Improvement Program, the staff concluded that portions of STS Section 5.0, "Administrative Controls," could be relocated to licensee-controlled documents. This improvement was incorporated into the ABWR TS.

COL applicants will have to ensure that the portions of Section 5.0 relocated to licensee-controlled documents are controlled in accordance with an administrative control system acceptable to the staff. To complete its review of the ABWR TS, the staff performed an independent audit to verify the accuracy and completeness of the TS. The staff provided comments to GE in a series of letters dated February 2, 10, 14, and 16, 1994, which were addressed by GE in the certified final ABWR TS provided in SSAR Amendment 34.

16.3 Conclusion

The staff concludes that the ABWR TS are consistent with the regulatory guidance in the BWR/6 and BWR/4 STS and contain design-specific parameters and additional TS requirements considered appropriate by the staff. Therefore, the ABWR TS satisfy 10 CFR 50.34 and 50.36 and are acceptable.

17 QUALITY ASSURANCE

17.1 Quality Assurance During the Design Phase

17.1.1 General

Standard safety analysis report (SSAR) Chapter 17 describes the quality assurance (QA) program for the design phase of the advanced boiling water reactor (ABWR) and references GE Nuclear Energy (GE) QA topical report, NEDO-11209-04A, which the staff reviewed and found acceptable. SSAR Chapter 17 also provides additional QA information specifically applicable to the ABWR. The staff assessed GE's description of the QA program for the design phase of the ABWR to determine if it complies with the requirements of 10 CFR Part 50, Appendix B, and with applicable QA-related regulatory guides listed in Table 17.1 of this report.

The basis of the staff's review was Standard Review Plan (SRP) Section 17.1, which addresses both design and construction QA. The development and implementation of the construction QA program were identified in the draft final safety evaluation report (DFSER) as DFSER combined license (COL) Action Item 17.1.1-1. GE addressed this item in Amendment 31 of the SSAR (SSAR Section 17.0.1.1), which is acceptable to the staff. GE also included this action item in the SSAR.

17.1.2 Organization

The structure of the organization responsible for the design of the ABWR and for the establishment and execution of the design-phase QA program is shown in Figure 17.1 of this report. The line organizations have been assigned specific QA responsibilities, including both internal audits and audits of suppliers, to ensure compliance with the QA program. Audits conducted by GE's Nuclear Quality Assurance (NQA) organization are superimposed on these audits.

The General Manager of Nuclear Operations is responsible for ensuring that (1) the intent of GE's nuclear quality policy is reflected in its nuclear products and services, (2) a system is in place to independently assess the performance of organizations that affect the quality of these products and services, and (3) a system is in place to resolve issues that could affect GE's ability to satisfy its nuclear quality policy and other quality-related commitments.

NQA is a staff organization responsible for establishing the nuclear quality policy and procedures that are issued by the Vice President and General Manager of GE. NQA is also responsible for (1) auditing the various line organizations involved in the nuclear business and ensuring conformance of these organizations' procedures and practices with applicable corporate and nuclear quality-related policy and procedures, (2) ensuring integration of the organizations' quality planning into an effective QA program, (3) participating in management review boards that operate independently of the design verification by the line organizations, and (4) specifying how the line organizations are to comply with the nuclear quality policy and procedures. For the ABWR design, NQA is responsible for coordinating and integrating the QA program as it relates to engineering and management of the project.

A quality council aids NQA in fulfilling its responsibilities. The council's responsibility is to ensure total quality system coverage, uniformity, consistency, and continuity and to eliminate system deficiencies. The Manager of NQA chairs the quality council. Members of this council, as shown in Figure 17.1, are the managers responsible for QA in each of the major nuclear organizations. The council provides these managers direct access to top-level management and acts as a forum for the review of quality problems and corrective actions.

The line organizations are responsible for planning and implementing the QA functions performed within their areas of responsibility so that each organization's QA program complies with the nuclear QA policy and procedures established by NQA. The individual QA managers report to their department-level management and have the organizational independence and authority to identify quality-related problems; initiate, recommend, or provide solutions pertaining to conditions adverse to quality; and verify implementation of such solutions.

GE and its major technical associates, Hitachi and Toshiba, are designing the ABWR. The lead responsibility tc produce each specification (through the major purchasing specifications) and drawing is assigned to one design organization within GE, Hitachi, or Toshiba. The content of each of these common engineering documents is reviewed and approved by GE engineering personnel, and GE is responsible for the design and the supporting calculations and records for the ABWR.

GE engineering organizations are responsible for the U.S. ABWR design and design control by

- ensuring incorporation of applicable regulatory requirements, codes, standards, criteria, and design bases into the design
- ensuring incorporation of project design requirements into the design

Table 17.1 Quality assurance regulatory guide commitments

No.	Title	Revision	Date
1.8	Personnel Selection and Training	1	September 1975
1.26	Quality Group Classification, and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants	3	February 1976
1.28	Quality Assurance Program Requirements (Design and Construction), using NQA-1 and NQA-2*	3	August 1985
1.29	Seismic Design Classification	3	September 1978
1.30	Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment	0	August 1972
1.37	Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants*	0	March 1973
1.38	Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling of Items for Water-Cooled Nuclear Power Plants*	2	May 1977
1.39	Housekeeping Requirements for Water-Cooled Nuclear Power Plants	2	September 1977
1.58	Qualification of Nuclear Power Plant Inspection, Examination, and Testing Personnel*	**	
1.64	Quality Assurance Requirements for the Design of Nuclear Power Plants*	**	
1.74	Quality Assurance Terms and Definitions	**	
1.88	Collection, Storage, and Maintenance of Nuclear Power Plant Quality Assurance Records*	**	
1.94	Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants	1	April 1976
1.116	Quality Assurance Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems*	0-R	June 1976
1.123	Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants*	**	
1.144	Auditing of Quality Assurance Programs for Nuclear Power Plants	**	
1.146	Qualification of Quality Assurance Program Audit of Personnel for Nuclear Power Plants*	**	

* NRC accepted the GE positions given in Topical Report NEDO-11209-04A, Revision 8, March 31, 1989.

** Superseded by Revision 3 of Regulatory Guide 1.28.

Source:
GE Topical Report NEDO-11209-04A
GE Transmittal Letter dated January 12, 1993
"GE NUCLEAR QUALITY ASSURANCE ORGANIZATION CHART

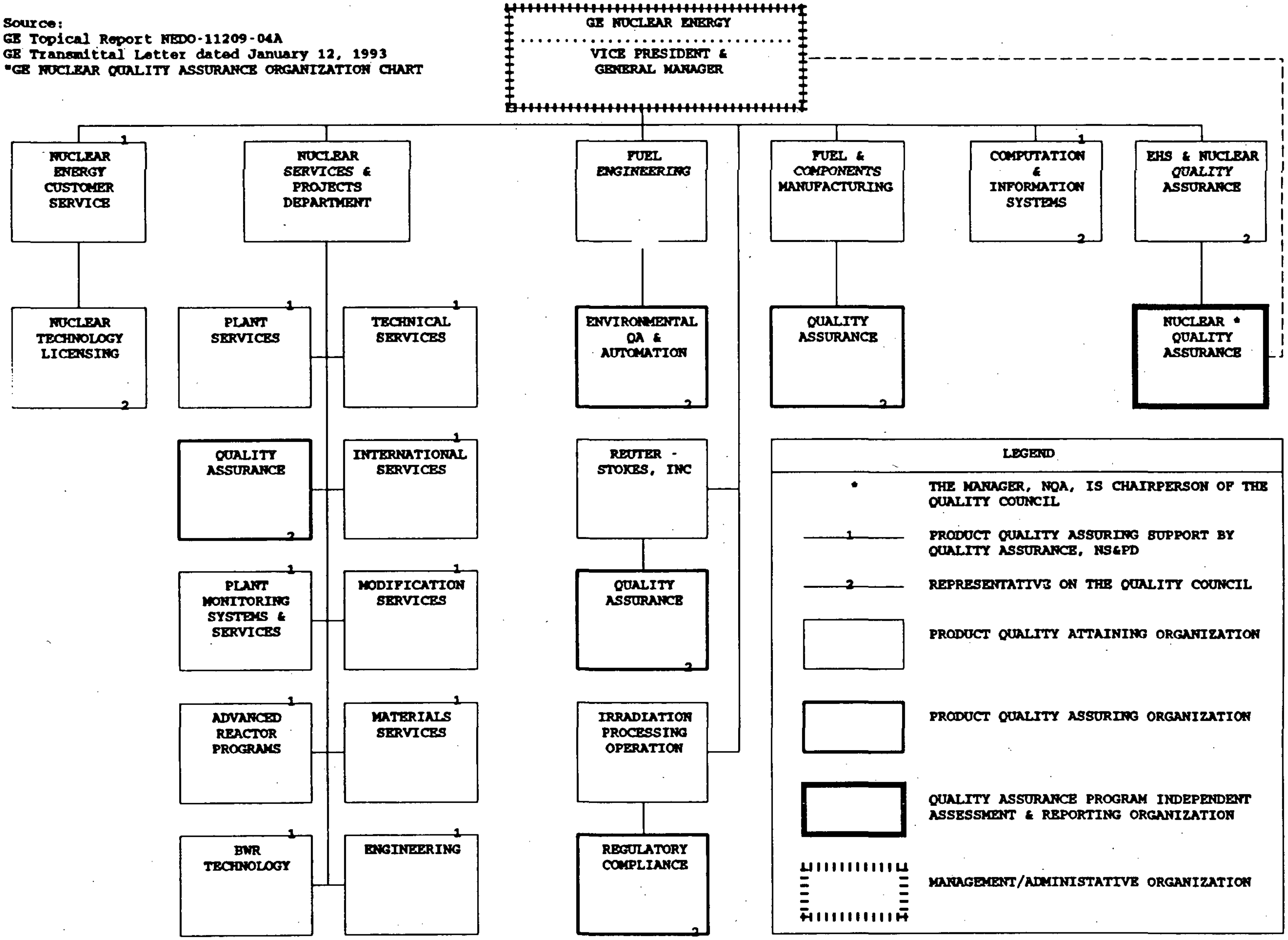


Figure 17-1 ABWR design organization structure

- translating the design information into the appropriate design documents
- verifying the design adequacy either through independent design review, the use of alternative or simplified calculational methods, or the performance of a suitable testing program
- coordinating design activities among interfacing design engineers and design organizations
- reviewing, approving, issuing, and distributing design documents under a controlled document system
- controlling design changes and changes to design documents in accordance with documented procedures
- providing for the retention, storage, control, and retrievability of design record documents
- taking corrective action as necessary to correct design errors and to improve the design control function

17.1.3 Quality Assurance Program

GE structured its nuclear QA program to satisfy Appendix B to 10 CFR Part 50 and the provisions of the applicable Nuclear Regulatory Commission (NRC) regulatory guides identified in Table 17.1 of this report. GE uses this QA program to control its design of the ABWR. GE has written policies, procedures, and instructions to implement the program. These documents control quality-related activities in accordance with the requirements of Appendix B to 10 CFR Part 50 and with applicable regulations, codes, and standards. The GE QA organizations ensure that procedures and instructions are provided for meeting the QA requirements. In addition, QA personnel conduct reviews and audits to verify the effective implementation of the program.

GE's nuclear QA program requires that implementing documents encompass detailed controls for (1) translating codes, standards, regulatory requirements, technical specifications, engineering requirements, and process requirements into drawings, specifications, procedures, and instructions; (2) developing, reviewing, and approving procurement documents and changes thereto; (3) prescribing all quality-related activities by documented instructions, procedures, drawings, and specifications; (4) issuing and distributing approved documents; (5) purchasing items and services; (6) identifying materials, parts, and components; (7) performing special processes; (8) inspecting and/or testing materials, equipment, processes, and services; (9) calibrating and maintaining measuring and test equipment; (10) handling, storing, and shipping items; (11) identifying the inspection, test, and operating status of items; (12) identifying and dispositioning nonconforming items; (13) correcting conditions adverse to quality; (14) preparing and maintaining QA records; and (15) auditing activities that affect quality.

Training and experience requirements are defined for each position in the GE organization. In addition, GE indoctrinates and trains personnel performing activities affecting quality to ensure that appropriate proficiency is achieved and maintained and that personnel responsible for quality-related activities are instructed as to the purpose, scope, and implementation of the quality-related manuals, instructions, and procedures. The indoctrination and training are carried out through documented procedures, on-the-job training, personal contacts, and meetings.

The ABWR design and changes to it are formally verified. Design verification is a process for an independent review of the design against design requirements to confirm that the designer's methods and conclusions are consistent with requirements and that the resulting design is adequate for its specified purpose. Design verification is performed and documented by persons other than those responsible for the design, using the method specified by the design organization. Designs are verified by one or more of the following methods: design review, qualification testing, alternative or simplified calculations, or checking. Team design reviews are continuing reviews of a design, selected by engineering management, to evaluate design adequacy that includes concepts, the design process, methods, analytical models, criteria, materials, applications, or development programs. When appropriate, team design reviews are used to verify that product designs meet functional, contractual, safety, regulatory, industrial codes and standards, and GE requirements. The selection of the design review team depends on the product design and the type of review. The technical competence of the members of each team encompasses three broad categories: (1) those with broad experience on similar products; (2) those with specialized technical expertise such as in heat transfer, materials, and structural analysis; and (3) those with a functional expertise such as QA, manufacturing, engineering, and product service.

For the international ABWR design, the lead design organization prepares the common engineering design document and circulates it internally for engineering review, approval, and design verification. Evidence of verification is entered into the design records of the responsible design organization. Each document is distributed to the design organizations of the other parties for their review and approval of technical content and design interfaces. All comments resulting from this process are resolved. After the comments are resolved, the design verification is reviewed and, when necessary, updated to ensure that changes did not invalidate the original verification. After final agreement is reached, the document is finalized by the lead design organization, circulated to the other parties for their approval signatures, and issued. Changes to ABWR documents are handled similarly. Differences between international and U.S. ABWR designs are identified in a controlled list (called the design action list) for future design action and application.

GE's QA organizations are responsible for establishing and implementing the audit program. Audits are performed in accordance with preestablished written checklists by qualified personnel not having direct responsibilities in the areas being audited. Periodic audits are performed to evaluate all aspects of the QA program, including the effectiveness of implementation. The QA program requires the review of audit results by the person having responsibility in the area audited to determine and take corrective action where necessary.

Followup audits are performed to determine if nonconformances and deficiencies have been effectively corrected and the corrective action precludes repetitive occurrences. Audit reviews, which indicate performance trends and the effectiveness of the QA program, are reported to responsible management for review and assessment.

The staff concludes that GE's QA program for the design phase of the ABWR describes requirements, procedures, and controls that, when properly implemented, will comply with the QA requirements of Appendix B to 10 CFR Part 50, applicable QA-related regulatory guides, and the acceptance criteria in SRP Section 17.1, related to QA during design and construction. In addition, SSAR Chapter 17 references GE QA topical report, NEDO-11209-04A, which the staff has reviewed and found acceptable.

During its review of the QA program described in the ABWR SSAR, the staff audited the implementation of the program at GE's offices in San Jose, California, during the week of February 6, 1989. The report of this audit is in the Commission's Public Document Room, the Gelman Building, 2120 L Street NW, Washington, DC. On the basis of the sample of design activities audited, which included Hitachi and Toshiba documents requested by the staff and translated into English, the auditors concluded that the design QA programs implemented by GE, Hitachi, and Toshiba met the applicable requirements of Appendix B to 10 CFR Part 50 and were acceptable for designing the ABWR.

An inspection of the ABWR design process was performed from September 7 through 10, 1993. The inspection results are documented in NRC Inspection Report 99900403/93-02. The inspection scope included an examination of GE QA controls applied to the ABWR project. This included a review of design record files (DRFs), selected computer codes used for accident analysis and transient modeling, test activities, design calculations, and audits. The inspection questioned the technical adequacy of supporting calculations generated by the international technical associates (TAs). Some test data for the Full Integral Simulation Test could not be retrieved by GE, and some calculation notebooks were poorly maintained. The staff evaluation of GE's response to the findings of that inspection was Open Item F17.1.3-1.

GE provided a response to the staff's inspection report on November 24, 1993, which addressed the items of concern and proposed corrective and preventive actions such as: verifying the accuracy of an input parameter for a LOCA analysis and performing related sensitivity studies, disseminating training reminders to technical staff about the QA requirements for design analysis and DRFs, increasing the GE audit emphasis on the content of DRFs, verifying that installed test instrumentation was within specified tolerances, supplementing transient analysis code DRFs, confirming that engineering services were provided under the auspices of an Appendix B quality program, correcting SSAR inaccuracies, and performing design verification on a design calculation. The staff found these proposed actions to be acceptable with a few exceptions. A request for further information and clarification was sent to GE on December 22, 1993, for the issues involving the technical oversight by GE of supporting calculations generated by the TAs and the conduct of computer code design verification. GE's response dated January 17, 1994, was found to be acceptable with one exception discussed as follows.

During the course of the inspection in September 1993, the staff identified that the common engineering documents (design specifications, process flow diagrams (PFDs), instrument block diagrams (IBDs), and piping and instrument diagrams (P&IDs)) have received a considerable level of GE design review. However, the level of GE review performed on the supporting calculations generated by the international TAs was not found to be rigorous. For example, the NRC inspection found that the depth of technical review afforded by the GE program reviews (QA audits) was minimal as the audit teams had not been supplemented by technical reviewers. In addition, little documented evidence was found in the DRFs to substantiate GE's review of the supporting calculations.

GE informed the staff that a sufficient level of confidence was obtained in the supporting calculations through the performance of GE program reviews of each TA, the GE engineering reviews of the common engineering documents, and participation by GE staff in numerous design review meetings. In addition, GE provided amplifying information during meetings with the staff on March 14 and 15, 1994, with respect to the extensive GE involvement during the ABWR design evolution. GE stated that, during the period from 1978 through 1985, extensive technical interaction transpired between GE and the TAs.

On March 22 through 24, 1994, a second NRC inspection was performed to substantiate the extent of the GE technical oversight of the TA's supporting design and analysis efforts. The inspection spanned a representative sampling of ABWR systems for which a TA had lead design responsibility. The staff examined the associated GE DRFs, interviewed cognizant GE design engineers, reviewed engineering correspondence from the TAs, and searched for examples of GE verification of TA calculations.

The three-day inspection resulted in the identification of evidence of GE's technical oversight of the supporting design as documented by the Phase 3 "Advanced BWR Plant Evaluation Report," GE comparisons of the ABWR design parameters with respect to the BWR 5 and 6 plant designs, thorough GE review of the common engineering documents that included proposed design revisions and independent GE calculations, the existence of selected TA supporting calculations in the GE DRFs, and GE review of system analysis, system performance, and capacity calculations generated by the TAs.

The inspection determined that reasonable assurance was provided by the depth, extent, and duration of the GE technical oversight of the joint design process to resolve the remaining issue from the September 1993 inspection. During the March 1994 inspection the staff additionally reviewed selected GE corrective and preventive actions that had been implemented in response to other concerns raised during the September 1993 inspection and found them satisfactory. Therefore, Open Item F17.1.3-1 is resolved based on the March 1994 inspection findings and the corrective and preventive measures instituted by GE in response to the QA and design control concerns identified in NRC Inspection Report 99900403/93-02.

An applicant for a COL, when completing its detailed design and equipment selection during the COL design phase, will submit its QA program for the design phase for staff review. This will be in addition to the staff review of the COL applicant's QA program for both the construction and the operation of the facility. When the COL applicant's QA programs are submitted, whether they are the GE QA programs augmented with information by the COL applicant, or a completely new QA program, the staff will perform the necessary reviews in ensure compliance to 10 CFR Part 50, Appendix B. This was DFSER COL Action Item 17.1.1-1.

17.2 Quality Assurance During the Operations Phase

The operations QA program is beyond the scope of GE's application for design certification (DC) and was identified by the staff in the DFSER as DFSER COL Action Item 17.2-1. This item was addressed by GE Amendment 31 of the SSAR (SSAR Section 17.0.1.1), which is acceptable to the staff. GE has also included this action item in the SSAR. For a discussion on the relationship of the COL applicant's Operational Reliability Assurance Process to the QA program, see Section 17.3 of this report.

17.3 Reliability Assurance Program

Introduction

SSAR Section 17.3 describes the reliability assurance program (RAP) for the design phase of the ABWR. GE implements the design reliability assurance program (D-RAP) for its scope of design during detailed design and specific equipment selection phases to ensure that the important ABWR reliability assumptions of the probabilistic risk assessment (PRA) will be considered throughout plant life. The COL applicant will augment and implement the D-RAP for its scope of design and equipment selection (See SSAR 17.3.13). Additionally, the COL applicant should develop and implement a process whose objectives are to monitor equipment performance and evaluate equipment reliability to provide reasonable assurance that the plant is operated and maintained commensurate with PRA assumptions so that the overall safety is not unknowingly degraded and remains within acceptable limits (See SSAR 17.3.13). This process could be described as an operational reliability assurance process (O-RAP) that should be included under existing programs for quality assurance and maintenance. When structures, systems, and components (SSCs) monitoring and evaluation identifies performance or condition problems, appropriate corrective action will be taken to ensure SSCs remain capable of performing their intended functions. However, the RAP does not attempt to statistically verify the numeric values used in the PRA through performance monitoring.

The staff has evaluated SSAR Chapter 17.3, which included the GE response (dated March 5, 1992) to the staff's request for additional information (RAI) contained in its request for resolution of issues related to SSAR Chapter 19, Appendix K (letter dated September 19, 1991). The acceptability of the particular method to identify and prioritize risk-significant structures, systems, and components (SSCs) associated with PRA was reviewed as part of SSAR Chapter 19 and is addressed in Chapter 19 of this report.

Background

The need for a safety-oriented reliability effort for the nuclear industry was identified by the NRC in Three Mile Island (TMI) Action (NUREG-0660) Item II.C.4. Subsequently, initial NRC research in the area of reliability assurance began in the early 1980s. The results of this research showed that an operational reliability program based on a feedback process of monitoring performance, identifying problems, taking corrective action, and verifying the effectiveness of these actions was needed and that other NRC initiatives (e.g., maintenance inspections, performance indicators, aging programs, and technical specification improvements) would address this need. The NRC concluded from this research that an operational reliability program could be implemented most effectively in a performance-based, nonprescriptive regulation, where NRC mandates the level of safety performance to be achieved. For example, licensees could be required to set availability and reliability targets for selected systems and to measure performance compared to the targets.

The TMI item was closed out for operating reactors in October 1988 without further action because several NRC initiatives had effectively subsumed the operational reliability program effort. These initiatives included efforts to (1) improve maintenance and better manage the effects of aging, (2) improve technical specifications, (3) develop and use plant performance indicators, and (4) develop an operational reliability program as an acceptable means of meeting the station blackout rule (10 CFR 50.63).

NUREG-1070, "NRC Policy on Future Reactor Designs" included the concept of a systems reliability program to ensure that the reliability of components and systems important to safety would remain at a sufficient level. To ensure that reliability objectives are met and to prevent degradation of reliability during operation, the NRC envisioned that the PRA, performed at the design stage, would be used as a tool in making detailed design decisions affecting procurement, testing, and the formulation of operations and maintenance procedures.

In a few specific instances, the NRC is studying or has established reliability targets for systems and components. For example, SRP Section 10.4.9 requires that an acceptable auxiliary feedwater system design have an unreliability in the range of 10E-4 to 10E-5 per demand. The resolution of Generic Issue B-56 involves efforts to determine, monitor, and maintain emergency diesel generator reliability levels. Additional regulatory bases for key elements of a RAP can be found in Appendices A and B to 10 CFR Part 50 and 10 CFR 50.65.

In SECY-89-013, "Design Requirements Related to the Evolutionary Advanced Light Water Reactors," dated January 19, 1989, the staff identified several issues for next-generation light water reactors that go beyond present acceptance criteria defined in the SRP. RAP, as one of these issues, was defined as a program to ensure that the design reliability of safety-significant SSCs is maintained over the life of a plant. In SECY-89-013, the staff informed the Commission that RAP would be required for final design approval or design certification (DC). In November 1989, potential applicants for DC were informed by letter that "the NRC staff was considering matters that went beyond the current SRP . . . that [the NRC] expects these advanced reactor designs to embody." Reliability assurance was identified as one of these matters.

In SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs," dated April 2, 1993, the staff recommended that the Commission approve its interim position that a high-level commitment to a RAP application to design certification be required as a non-system generic Tier 1 requirement with no associated inspections, tests, analyses, and acceptance criteria (ITAAC). The details of the D-RAP, including the conceptual framework, program structure, and essential elements, should be provided in the SSAR. The SSAR for the D-RAP should also (1) identify and prioritize a list of risk-significant SSCs based on the DC PRA and other sources; (2) ensure that the vendor's design organization determines that significant design assumptions, such as equipment that satisfies the design reliability and unavailability, are realistic and achievable; (3) provide input to the procurement process for obtaining equipment that satisfies the design reliability assumptions; and (4) provide these design assumptions as input to the COL applicant for consideration in the O-RAP. A COL applicant would augment the D-RAP with site-specific design information and would implement the balance of the D-RAP, including input to the procurement process (See SSAR 17.3.13).

The RAP consists of two distinct parts: (1) D-RAP and (2) O-RAP. D-RAP involves a top-level program at the design stage that is used to define the scope, conceptual framework, and essential elements of an effective RAP. D-RAP is also used to implement those aspects of the program that are applicable to the design process. In addition, D-RAP is used to identify the relevant aspects of plant operation, maintenance, and performance monitoring for the risk-significant SSCs for the operator's consideration in developing an O-RAP. The O-RAP objectives should be incorporated into existing programs (i.e., quality assurance and maintenance) that will be used to monitor equipment performance and evaluate equipment reliability to provide reasonable assurance that the plant is operated and maintained commensurate with PRA assumptions so that the overall safety is not unknowingly degraded and remains within acceptable limits. When SSC monitoring and evaluation identifies performance or condition problems, appropriate corrective action will be taken to assure SSCs remain capable of performing their intended functions. However, the RAP does not attempt to statistically verify the numerical values used in the PRA through performance monitoring.

The staff's final position on RAP was presented in the Commission paper on the Regulatory Treatment of Non-Safety Systems (RTNSS) SECY-94-084 dated March 28, 1994. The Commission approved the following applicable regulation for D-RAP:

> An application for design certification or for a combined license must contain:
>
> (a) the description of the reliability assurance program used during the design that includes, scope, purpose, and objectives;
>
> (b) the process used to evaluate and prioritize the structures, systems, and components in the design, based on their degree of risk-significance;
>
> (c) a list of structures, systems, and components designated as risk-significant; and
>
> (d) for those structures, systems, and components designated as risk-significant:
>
> (i) A process to determine dominant failure modes that considered industry experience, analytical models, and applicable requirements; and
>
> (ii) Key assumptions and risk insights from probabilistic, deterministic and other methods that considered operation, maintenance and monitoring activities.
>
> Each COL applicant that references the ABWR design must implement the D-RAP approved by the NRC.

The staff evaluated the SSAR on the basis of the applicable regulation stated above.

A COL applicant would augment the designer's RAP to reflect plant-specific information and implement those elements applicable during the construction and operation phases. The staff's COL application review will be similar to the design certification review and include an evaluation of the updated (site-specific) PRA, probabilistic, deterministic and other insights (e.g., operating experience) to assess any changes to risk-significant SSCs and site-specific vulnerabilities. The staff will review the COL applicant's proposed design reliability assurance program plan to determine if it satisfies the above requirements at the time of the COL application.

Evaluation

In its RAI dated September 19, 1991, the staff stated that the GE ABWR RAP submittal should (1) describe the basic framework of a RAP including the scope, purpose, objective, basic definitions, and elements (RAI Question (Q)1); (2) when describing the RAP concepts and elements, include a discussion on performance goal or targets, problem prediction and recognition, problem prioritization and correction, and problem closeout (RAI Q2); (3) describe how RAP will address plant aging concerns (RAI Q3); (4) describe the organizational and administrative aspects for implementing an effective RAP (RAI Q4); (5) describe the approach for providing feedback to the designer when actual plant performance data consistently differ from the designers PRA and RAP assumptions (RAI Q5); (6) describe the major programmatic interface between the RAP and areas such as design, construction, startup testing, operations, maintenance, engineering, safety, licensing, quality assurance, and procurement (RAI Q6); and (7) provide an example of how the GE RAP would be implemented using a specific SSC identified as risk significant in the PRA (RAI Q7). These questions included the use of the term "RAP;" however, the intent was for the questions to apply to that portion of the RAP that GE is responsible for preparing and implementing (e.g., the ABWR D-RAP). The RAI questions provided an outline of the staff's expectations on RAP for evolutionary designs and explicitly stated the details of these expectations.

In its letter dated March 5, 1992, GE stated that a new SSAR Section 17.3 would address the staff's RAI questions. After reviewing the initial GE response and the modifications to SSAR Section 17.3 in Amendment 26 (dated March 24, 1993) the staff determined that (1) RAI Q1 was answered by Sections 17.3.2, 17.3.3, 17.3.4, 17.3.6, 17.3.7, and 17.3.8; (2) RAI Q2 was answered by Sections 17.3.6, 17.3.9, and 17.3.10; (3) RAI Q3 was answered by Section 17.3.10; (4) RAI Q4 was answered by Section 17.3.5; (5) RAI Q5 was answered by Section 17.3.10; (6) RAI Q6 was answered by Section 17.3.10; and (7) RAI Q7 was answered by Section 17.3.11. Therefore, the answers to the RAI questions contained in SSAR Section 17.3 address the staff's expectations on RAP for evolutionary designs and explicitly state GE's response to these expectations. The details of the staff's evaluation are presented in Sections 17.3.1 through 17.3.11 below.

17.3.1 General

RAP for the design phase of the ABWR described in SSAR Section 17.3 ensures that the important ABWR reliability assumptions of the PRA will be considered throughout plant life. The PRA is used to evaluate the anticipated plant response to initiating events. The PRA evaluates the plant response to initiating events to substantiate, in part, that plant damage has a very low probability and risk to the public is very low. Input to the PRA includes details of the plant design and assumptions about the reliability of the plant risk-significant SSCs. Changes to equipment and system reliabilities will be reevaluated and as necessary will be reflected in a revised PRA throughout plant life. GE started the D-RAP during design of the ABWR. The COL applicant will utilize the ABWR D-RAP during the detailed design and specific equipment selection phases to complete the D-RAP. The COL applicant will complete the D-RAP and will also incorporate the objectives of the O-RAP into existing programs (i.e., maintenance and quality assurance) that will be used to monitor equipment performance and evaluate equipment reliability to provide reasonable assurance that the plant is operated and maintained commensurate with PRA assumptions so that the overall safety is not unknowingly degraded and remains within acceptable limits. When SSC monitoring and evaluation identifies performance or condition problems, appropriate corrective action will be taken to assure SSCs remain capable of performing their intended functions.

GE states that the D-RAP will include the design evaluation of the ABWR. It will be used to identify relevant aspects of plant operation, maintenance, and performance monitoring of important plant SSCs for the COL applicant's consideration in ensuring safety of the equipment and limited risk to the public. The COL applicant will specify the policy and implementation procedures for using D-RAP information provided by GE (See SSAR 17.3.1-13). SSAR Section 17.3 also includes a descriptive example of how the D-RAP will apply to the standby liquid control system (SLCS). This example shows how the principles of D-RAP will be applied to other systems identified by the PRA as significant with regard to risk.

The staff concludes that Section 17.3.1 of the SSAR meets the requirement of the applicable regulation for D-RAP to provide a description of the RAP used during the initial design as discussed above in Section 17.3 of this report and is acceptable.

17.3.2 Scope

In response to the part of RAI Q1 associated with the RAP scope, GE provided the additional information in Section 17.3.2 of the SSAR. In that section, GE states that the scope of the ABWR D-RAP includes identifying relevant aspects of plant operation, maintenance and monitoring of plant risk-significant SSCs. The PRA and other industry sources are used to identify and prioritize those SSCs that are important to prevent or mitigate plant transients or other events that could present a risk to the public.

The staff reviewed SSAR Section 17.3.2 with respect to the scope of the ABWR D-RAP and concludes that it is responsive to the staff's RAI question, meets the requirement of the applicable regulation for D-RAP to include the scope of the RAP as described in Section 17.3 of this report and is acceptable.

17.3.3 Purpose

In response to the part of RAI Q1 associated with the RAP purpose, GE provided the additional information in Section 17.3.3 of the SSAR. In that section, GE states that the purpose of the ABWR D-RAP is to ensure that plant safety, as estimated by the PRA, is maintained as the detailed design evolves through the implementation and procurement phases. Additionally, GE states that pertinent information is to be provided in the design documentation to the COL applicant so that equipment reliability, as it affects plant safety, can be maintained through operation and maintenance during the entire plant life.

The staff reviewed SSAR Section 17.3.3 with respect to the purpose of the ABWR D-RAP and concludes that it is responsive to the staff's RAI question, meets the requirement of the applicable regulation for D-RAP to include the purpose of the RAP as described in Section 17.3 of this report and is acceptable.

17.3.4 Objective

In response to the part of RAI Q1 associated with the RAP objective, GE provided the additional information in Section 17.3.4 of the SSAR. In that section, GE states that the objective of the ABWR D-RAP is to identify those plant SSCs that are significant contributors to risk, as shown by the PRA or other sources, and to ensure that, during the implementation phase, the plant design continues to utilize risk-significant SSCs whose reliability is commensurate with the PRA assumptions. The D-RAP also will be used to identify key assumptions regarding any operation, maintenance, and monitoring activities that the COL applicant should consider in developing its approach to implementing an O-RAP using existing programs such as quality assurance and maintenance to provide reasonable assurance that such SSCs can be expected to operate with reliability commensurate with that assumed in the PRA. A major factor in the ABWR D-RAP is risk-focused maintenance that considers all plant modes and equipment directly relied on in ABWR emergency operating procedures (EOPs).

The staff reviewed SSAR Section 17.3.4 with respect to the objective of the ABWR D-RAP and concludes that it is responsive to the staff's RAI question, meets the requirement of the applicable regulation for D-RAP to include the objective of the RAP as described in Section 17.3 of this report and is acceptable.

17.3.5 GE Organization for D-RAP

In response to RAI Q4 associated with the RAP organizational aspects, GE provided the additional information in Section 17.3.5 of the SSAR dated March 5, 1992. However, in its DFSER, the staff identified an inconsistency between the SSAR Section 17.3.5 narrative and SSAR Figure 17.3-1. Additionally, GE described the D-RAP organizational structure in the future tense in SSAR Section 17.3.5. This item remained open subject to a revision of SSAR Section 17.3.5 (DFSER Open Item 17.3.5-1). On January 18, 1993, GE submitted a markup to SSAR Section 17.3.5 that satisfactorily addressed DFSER Open Item 17.3.5.1 and deleted the organization chart (SSAR Figure 17.3-1). This markup was incorporated into SSAR Section 17.3.5 as Amendment 26 dated March 24, 1993. GE has also included this information in the SSAR. On the basis of this evaluation, this item is resolved.

In SSAR Section 17.3.5, GE states that the reliability analyses and the PRA, including SSAR Appendix 19K, were performed by GE. GE also developed the D-RAP definition. Responsibility for the design of key equipment, components, and subsystems was shared by GE and external organizations, including the organization performing architect-engineering functions. The GE manager assigned the responsibility of managing and integrating the D-RAP program had direct access to the ABWR project manager and kept him abreast of D-RAP critical items, program needs, and status. He had organizational freedom to (1) identify D-RAP problems; (2) initiate, recommend, or provide solutions to problems through designated organizations; (3) verify implementation of the solution; and (4) function as an integral part of the final design process.

The staff reviewed Amendment 26 of SSAR Section 17.3.5, which incorporated the markup submitted on January 18, 1993, with respect to the GE organizational description and accountability for implementing the ABWR D-RAP for DC, and concludes that it is responsive to the staff's DFSER open item concern, is responsive to the staff's original RAI Q4, satisfies the staff position that the D-RAP ensures that the vendor's design organization determine that significant design assumptions are realistic and achievable as discussed in SECY-93-087 and Section 17.3 of this report, and is acceptable. The staff also concludes that the GE organizational description and accountability for implementing the ABWR D-RAP were acceptable and resolved DFSER Open Item 17.3.5-1.

A COL applicant completing its detailed design and equipment selection during the COL design phase will submit its specific D-RAP organization for staff review (See SSAR 17.3.13).

17.3.6 SSC Identification and Prioritization

In response to the part of RAI Q1 associated with the RAP definitions and elements and the part of RAI Q2 associated with the RAP scope, GE provided the additional information in Section 17.3.6 of the SSAR.

In SSAR Section 17.3.6, GE states that the PRA prepared for the ABWR will be the primary source for identifying risk-significant SSCs that should be given special consideration during detailed design and procurement phases and considered for inclusion in the COL applicant's O-RAP. It also is possible that risk-significant SSCs will be identified from sources other than the PRA, such as nuclear plant operating experience, other industrial experience, and relevant component failure databases. SSAR Chapter 19 describes the method of identifying risk-significant SSCs using the PRA.

The staff reviewed SSAR Section 17.3.6 with respect to identifying and prioritizing risk-significant SSCs for the ABWR D-RAP and concludes that it is responsive to the staff's RAI questions, meets the requirement of the applicable regulation for D-RAP to describe the methodology used to evaluate and prioritize risk significant SSCs as described in Section 17.3 of this report, and is acceptable. The acceptability of the PRA methods or techniques to prioritize SSCs is addressed in Chapter 19 of this report.

17.3.7 Design Considerations

In response to the part of RAI Q1 associated with the RAP definitions and elements, GE provided the additional information in Section 17.3.7 of the SSAR. In SSAR Section 17.3.7, GE states that the reliability of SSCs identified by the PRA as risk significant will be evaluated at the detailed design stage by appropriate design reviews and reliability analysis. Current databases will be used to identify appropriate values for failure rates of equipment as designed, and these failure rates will be compared with those used in the PRA. Normally, the failure rates will be similar, but in some cases they may differ because of recent design or database changes. Whenever failure rates of designed equipment are significantly greater than those assumed in the PRA, an evaluation will be performed to determine if the equipment is acceptable or if it must be redesigned to achieve a lower failure rate.

For those SSCs identified by the PRA or other sources as risk significant redesign will be considered as a way to reduce the core damage frequency (CDF) contribution. If there are practical ways to redesign a risk-significant SSC, it will be redesigned and the change in system fault tree results will be calculated. Following the redesign phase, dominant SSC failure modes will be identified so that protection against such failure modes can be accomplished by appropriate activities during plant life. (See Chapter 19 of this report.)

For the COL applicant, GE will identify in the PRA, or other design documents, the risk-significant SSCs and the associated reliability assumptions, including any pertinent bases and uncertainties considered in the PRA (See Chapter 19 of this report). GE will also provide this information for the COL applicant to consider in developing an O-RAP to help assure that PRA results will be achieved over the life of the plant. The COL applicant can use this information for establishing appropriate reliability targets and the associated maintenance practices for achieving them.

The staff reviewed SSAR Sections 17.3.6, 17.3.7, and 17.3.8 and concludes that GE has provided a process for evaluating risk-significant SSCs for redesign and a process for providing information to a COL applicant for establishing appropriate reliability targets and the associated maintenance practices for an O-RAP. The staff also concludes that SSAR Section 17.3.7 is responsive to its RAI question, meets the requirement of the applicable regulation for D-RAP to describe the methodology used to evaluate and prioritize risk significant SSCs as described in Section 17.3 of this report, and is acceptable.

17.3.8 Defining Failure Modes

In response to the part of RAI Q1 associated with the RAP definitions and elements, GE provided the additional information in Section 17.3.8 of the SSAR. In SSAR Section 17.3.8, GE uses the methodology of NUREG/CR-5695, Section 5, to determine dominant failure modes of risk-significant SSCs in the D-RAP. The method includes using historical information, analytical models, and existing requirements.

The staff reviewed SSAR Sections 17.3.6, 17.3.7, and 17.3.8 and concludes that GE has provided a method for determining dominant failure modes for risk-significant SSCs in the ABWR D-RAP. The staff also concludes that SSAR Section 17.3.8 is responsive to its RAI question, meets the requirement of the applicable regulation for D-RAP to define failure modes as described in Section 17.3 of this report, and is acceptable.

17.3.9 Operational Reliability Assurance Activities

In response to the part of RAI Q2 associated with the RAP performance goals and targets, problem prediction, and problem recognition, GE provided the additional information in Section 17.3.9 of the SSAR. In SSAR Section 17.3.9, GE states that once the dominant failure modes are determined for risk-significant SSCs, an assessment is required to determine suggested O-RAP activities that will ensure acceptable performance during plant life. Such activities may consist of periodic surveillance inspections or tests, monitoring of SSC performance, or periodic preventive maintenance (PM).

Periodic testing of SSCs may include startup of standby systems, surveillance testing of instrument circuits to ensure that they will respond to appropriate signals, and inspection of passive SSCs to show that they are available to perform as designed. Performance monitoring, including condition monitoring, can consist of measurements of output, measurement of magnitude of an important variable, and testing for abnormal conditions. Periodic PM will be performed at regular intervals to preclude problems that could occur before the next PM interval.

Planned maintenance activities will be integrated with regular operating plans. Maintenance that will be performed more frequently than during refueling outages must be planned to avoid disrupting safe operation or causing a reactor scram, engineered safety feature actuation, or abnormal transient. Maintenance performed during refueling outages must not adversely affect plant safety.

The staff reviewed SSAR Sections 17.3.9 and 17.3.10 and concludes that GE has provided a process to determine operational reliability assurance activities using the dominant failure modes identified in the ABWR D-RAP. The staff also concludes that SSAR Section 17.3.9 is responsive to its RAI question, to include a description of O-RAP activities, and is acceptable. The COL applicant should incorporate O-RAP activities into existing programs such as maintenance and quality assurance and provide the staff with a description of how these activities are met at the time of the COL application. (See SSAR 17.3.13).

17.3.10 COL Applicant's Reliability Assurance Process

In response to the part of RAI Q2 associated with the RAP performance goals and targets, problem prediction, and recognition, problem prioritization and correction, and problem closeout, GE provided the additional information in Section 17.3.10 of the SSAR. Additionally, the O-RAP description of plant aging, feedback to designer, and programmatic interfaces responds to RAI Q3, Q5, and Q6, respectively. In SSAR Section 17.3.10, GE states that the O-RAP will be prepared and implemented by the COL applicant, using the information provided by GE. The information will help the COL applicant determine activities that should be included in the O-RAP. Examples of activities that might be included in an O-RAP are:

- reliability performance monitoring
- reliability methodology
- problem prioritization
- root cause analysis
- corrective action determination
- corrective action implementation
- corrective action verification
- plant aging
- feedback to designer
- programmatic interfaces

The COL applicant will address in its O-RAP the interfaces with construction, startup testing, operations, maintenance, engineering, safety, licensing, QA, and procurement of replacement equipment.

The staff concludes that the outline of an O-RAP provided by GE to be used by the COL applicant in SSAR Section 17.3.10 is responsive to its RAI questions, to include a description of O-RAP activities, and is acceptable. The COL applicant should also incorporate O-RAP activities into existing programs such as maintenance and quality assurance and provide the staff with a description of how these activities are met. The COL applicant will provide the D-RAP for completion of the detailed design and specific equipment selection phases (e.g., procurement of risk-significant SSCs), and a complete O-RAP to be reviewed by the staff as described in Sections 17.3.1 and 17.3.9 of this report.

17.3.11 D-RAP Implementation

In response to RAI Q7, GE provided in Section 17.3.11 of the SSAR, an example of how the GE RAP would be implemented using a specific SSC identified as risk significant in the PRA. For example purposes only, the SLCS was assumed to be a significant contributor to CDF or to offsite risk. The system description (including operation and differences from current BWRs) and a system fault tree were provided in the example. Seven SLCS risk-significant components identified in the example as having high importance in the SLCS fault tree were considered for redesign. Also, failure modes and maintenance requirements for the seven components were identified.

The staff concludes that the SSAR Section 17.3.11 example using the SLCS satisfactorily demonstrated GE's cognitive understanding of the RAP concept and their ability to incorporate it into the design, is responsive to its RAI question, and is acceptable. The process description for the implementation of the ABWR D-RAP by a COL applicant will be reviewed by the staff at the time a COL application is submitted.

17.3.12 Conclusion

The staff has reviewed Section 17.3 of the ABWR SSAR. The staff finds that the ABWR SSAR satisfies the requirements of the applicable regulation for D-RAP for the ABWR design phase of the reliability assurance program as described in Section 17.3 of this report, and is therefore acceptable.

18 HUMAN FACTORS ENGINEERING

To perform its evaluation of Standard Safety Analysis Report (SSAR) Chapter 18, the staff reviewed the information described in Section 18.1.2 of this report. The review was based on the current regulatory requirements in 10 CFR 52.47, 10 CFR 50.34(g), and 10 CFR 50.34(f) and the guidance in Standard Review Plan (SRP) Sections 13 and 18; NUREG-0700, "Guidelines for Control Room Design Review," September 1981; and NUREG-0933, "Prioritization of Generic Safety Issues Main Report," April 1983. The staff developed additional review criteria to provide a basis for its review of aspects of the advanced boiling water reactor (ABWR) human factors engineering (HFE) program that were not fully addressed by the previously mentioned documents. These criteria are contained in the staff's "Human Factors Engineering Program Review Model (PRM) and Acceptance Criteria for Evolutionary Reactors" which was forwarded to the Commission in SECY-92-299 dated August 27, 1992, and is attached as Appendix J to this report.

Section 18.1 of this report describes the methodology used to conduct the review, including the development of general review criteria that supplement the regulatory requirements and established guidelines. The results described in Sections 18.2 through 18.9 address the following eight major topics:

- design goals
- main control room (CR) standard design features
- inventory of controls, displays, and alarms
- remote shutdown system (RSS)
- local valve position indication (VPI)
- unresolved and generic safety issues
- emergency procedure guidelines (EPGs)
- design and implementation process

Section 18.10 gives a summary of the evaluation findings and overall conclusions.

As a result of the staff's initial review of the SSAR, many outstanding issues were identified and documented in the draft safety evaluation report (DSER) (SECY-91-320), and subsequently in the DFSER (SECY-92-349). One of the major issues to emerge from the initial review was that detailed human-system interface (HSI) information concerning the final design was not available for staff review as part of the design certification evaluation. GE's HSI analysis and design efforts resulted in a list of key CR design features characterized at a general level (not a detailed specification) and a minimum inventory of fixed safety-significant information and control requirements derived from an analysis of the ABWR EPGs and probabilistic risk assessment (PRA). Evaluation of the key features and the inventory is part of the certification review. However, they reflect a design in its preliminary stages (not a detailed design or specification) and by themselves do not provide a basis on which a safety determination can be made.

In SECY-92-053, "Use of Design Acceptance Criteria During 10 CFR Part 52 Design Certification Reviews," dated February 19, 1992, the staff proposed using design acceptance criteria (DAC) as an approach to the ABWR design review because detailed design information was unavailable for selected areas of rapidly changing technology, including human factors aspects of the CR and remote shutdown station design. Therefore, it was inadvisable to require detailed design specifications at this point. The inclusion of DAC in specific system inspections, tests, analyses, and acceptance criteria (ITAAC) is discussed in Section 14.3 of this report.

The staff will verify conformance with the ITAAC at several points during the CR design process. The documents that demonstrate satisfactory implementation of the ITAAC will be available for inspection as they are completed. In the DFSER the staff introduced the concept of conformance review points as part of the DAC process. These points would be key points during the DAC process at which the staff would complete an adequacy review. As its review of the development of the ITAAC and DAC continued, the staff determined that five specific conformance points were unnecessary and were impractical. The deletion of the discussion of specific conformance review points was the result of the staff's continued review of the design certification applications from several vendors. At the time the DFSER was written, the staff envisioned that a design would progress in such a fashion that at a given time (e.g., a conformance review point) all information related to a particular aspect of the design process associated with that particular conformance point would be available; for example, all implementation plans would be available at one time. Because of the nature of the design process, it is not practical to assume that all documentation on a specific conformance review would be available at one point in time.

DAC are prescribed limits, parameters, procedures, and attributes on which the staff relies to make a final safety determination to support design certification. The DAC are measurable or testable and must be verified in order for the staff to accept the final design. DAC delineate the process and requirements that a combined license (COL) applicant must implement during the development of detailed design information for the CR and the remote shutdown station. The adequacy of the detailed design will be periodically assessed as it develops. The COL applicant must demonstrate that the DAC are met. Failure to successfully complete ITAAC and any supporting DAC may require repeating earlier ITAAC and/or changing the system design.

Because the criteria for review of the design and implementation process for a CR or other control system were not clearly defined in current regulations and guidance documents, the staff developed criteria as part of this review. These criteria provided the basis with which to (1) assess if the appropriate HFE elements are included in the design and implementation process, (2) identify what materials need to be reviewed for each element, and (3) evaluate the adequacy of DAC and ITAAC to be used by the staff to verify each of the review elements, as developed by GE.

The staff design certification evaluation is based partially on design information and partially on an implementation process plan that describes the HFE program elements required to develop the key features and inventory into an acceptable detailed design specification. Along with the design and implementation process plan, GE has submitted the necessary DAC and ITAAC to ensure that the design and implementation process is properly executed by the COL applicant. GE has submitted a design and implementation process plan for the major design activities for the ABWR HFE effort. The first part of the plan presents GE's plant and system design elements; the second part describes the elements that must be implemented by a COL applicant to complete the design activity. The staff required that the design and implementation process plan contain descriptions of all required human factors activities (elements) that are necessary and sufficient for the development and implementation of the ABWR HSI.

18.1 Methodology

The staff review was performed in two phases. A preliminary review was performed on early versions of the SSAR and was documented in the DSER (SECY-91-320). This review is summarized in Section 18.1.1 of this report. DSER (SECY-91-320) issue resolution and further development of the SSAR were then reviewed and documented in the DFSER (SECY-92-349). The scope of this subsequent review is described in Section 18.1.2. As part of the final review, the staff developed the HFE PRM, or the review model, identified above for the evaluation of a design process. Development of the HFE PRM, found in Appendix J of this report, is described in Section 18.1.3.

18.1.1 Preliminary Review and Draft Safety Evaluation Report Issues

The primary source of information reviewed by the staff for the DSER (SECY-91-320) was SSAR Chapter 18 (updated through Amendment 15) and Chapter 13 (updated through Amendment 7) and GE's responses to staff requests for additional information (RAI), Questions 620.1 through 620.37, as documented in Chapter 20 of the SSAR. The review focused on four important aspects of ABWR human factors considerations:

- the organizational structure of the human factors function
- design goals and assumptions
- design processes
- the specification of HSI design requirements

In addition, the review included GE's resolution of those safety issues (unresolved safety issues, generic safety issues and the construction permit/manufacturing license (CP/ML) rule of 10 CFR 50.34(f)) related to human factors considerations addressed in SSAR Chapters 13 and 18.

From its initial review, the staff concluded that the human factors program for the HSI was generally inadequate as presented in GE's initial documentation and that SSAR Chapter 18 and Sections 13.2 and 13.5 did not provide sufficient information to support a determination that the ABWR design as proposed by GE for certification would adequately incorporate accepted human factors considerations in a manner that would achieve required safety and reliability. The principal reasons for this finding were: (1) design bases were specified in the SSAR without supporting rationale, (2) a design process was presented in insufficient detail and without results, (3) HSI design requirements were presented without evidence that they were derived from the design process and without supporting tests and evaluations, and (4) the documentation did not provide sufficient detail to support the review of the ABWR human factors efforts to a level necessary for design certification. Twenty-four issues were identified as requiring resolution. The issues are listed in Table 18.1 of this report. The table shows the section in this chapter where each DSER (SECY-91-320) open issue is addressed. In the discussions with GE that followed issuance of the DSER (SECY-91-320), two issues identified below were added in order to address the overall lack of design detail. These two issues are discussed in detail in Sections 18.4 and 18.9 of this report.

Table 18.1 DSER (SECY-91-320) Chapter 18 human factors engineering issues

Issue Number	Issue	DSER Section	SER Section Were Addressed
18.01	Qualifications of GE ABWR human factors design team	18.3.1	18.1.1
18.02	Human-systems interface (HSI) design and evaluation process	18.3.1	18.3
18.03	Number of staff in control room	18.3.2	18.2
18.04	Operator and system reliability	18.3.2	18.9
18.05	Operator workload analysis	18.3.2	18.9
18.06	Tests and analysis to support design implementation: a. Analyses conducted to date b. Further testing	18.3.2	18.3
18.07	ABWR human factors program plan (HFPP)	18.3.3.1	18.9
18.08	Control room prototype: a. Standardized features b. Prototype evaluation	18.3.3.1	18.3 18.9
18.09	Operator workload	18.3.3.2	18.3
18.10	Detailed task analyses	18.3.2	18.9
18.11	Tests, evaluations, studies to support design approaches	18.3.2	18.3
18.12	Adequacy of HSI design requirements	18.3.4.1	18.3
18.13	HSI design requirements for cathode ray tube (CRT), flat panel, and large-screen displays	18.3.4.1	18.9
18.14	Analysis to justify sole operator attentiveness and rationale for number of operators at main console	18.3.4.2	18.2
18.15	CRT display information	18.3.4.2	18.9
18.16	Power generation control system reliability	18.3.4.2	18.9
18.17	Alarm suppression criteria, alarm points	18.3.4.2	18.9
18.18	Safety parameter display system design scope	18.3.4.2	18.3
18.19	Remote shutdown system design rationale	18.3.4.2	18.5
18.20	Local valve position indication	18.3.4.2	18.6
18.21	Procedure development	18.3.4.4	18.1.1
18.22	Training materials	18.3.4.4	18.1.1
18.23	Unresolved safety issues and generic safety issues	18.4.1	18.7
18.24	Construction permit/manufacturing license-issues	18.4.2	18.7
18.25	Design process	-	18.9
18.26	Inventory	-	18.4

- Issue 18.25 - Design Process
- Issue 18.26 - Inventory

DSER (SECY-91-320) Issues 18.01, 18.21, and 18.22 were resolved before the SSAR review for reasons discussed below.

DSER (SECY-91-320) Issue 18.01: Qualifications of the GE ABWR Human Factors Design Team

In Section 18.3.1 of the DSER (SECY-91-320), the staff stated that "additional detailed information on the human factors qualifications of the GE ABWR human factors design team is required." This information was deemed important in the DSER review because of the assumption that the SSAR was to include an essentially complete design for the HSI. GE responded by providing design team information in its letter of October 1, 1991.

Evaluation: Since HFE is to be primarily performed by the COL applicant, the issue is no longer a concern. Instead, the focus on the design team was shifted toward establishing the qualification requirements of the design team that will actually implement the HFE program discussed in Section 18.9 of this report. Because of the change in focus for certification from an HFE design to a design process, GE's initial response to this issue in its letter of October 1, 1991, provided sufficient information for the staff's evaluation of this issue. Since GE has addressed the qualification requirements for its HFE design team, this item is resolved.

DSER (SECY-91-320) Issue 18.21: Procedure Development

In the DSER (SECY-91-320), the staff stated that system-level operating procedures had been developed concurrently with the development of the ABWR systems design. These procedures and the associated task analyses on which the HSI requirements are based were not included in the references (SSAR Section 18.6); thus, they could not be evaluated. For the ABWR design certification, the staff expects the COL applicant to provide detailed program descriptions for the development of standardized plant procedures and standardized plant personnel training materials. Further, the staff expects the COL applicant to develop integrated operating procedures that reflect the full level of detail consistent with and included as part of the final plant design. In addition, the COL applicant should develop procedure development guidelines (e.g., procedure writers guide, verification and validation (V&V) guidelines, and generic technical guidelines) with sufficient detail to ensure that the COL applicant's implementation of the processes and criteria delineated in these guidelines, when revising procedures will preserve the human factors insights and the overall ABWR design.

Evaluation: The staff determined that the development of detailed procedures and training materials was beyond the scope of the ABWR design certification and was the responsibility of the COL applicant. This was DFSER Confirmatory Item 18.9.2.2.7-1. Chapter 13.5.3.3 of the SSAR states that procedures will be used during the V&V process as described in Article VII of Table 18E-1. The staff accepts GE's determination that procedure development is beyond the scope of the ABWR design certification, and GE has included a COL action item that is acceptable. Therefore, this item is resolved.

DSER (SECY-91-320) Issue 18.22: Training Materials

In the DSER (SECY-91-320), the staff stated that it "expects GE to develop and submit for certification a detailed program description for developing the training material as part of the design certification for the ABWR." After the DSER was issued, the staff determined that training materials were already a part of the established licensing review process under 10 CFR Part 50 for the COL applicant and did not need to be addressed in the certification review under 10 CFR Part 52. The staff accepts that the development of training materials is beyond the scope of the ABWR design certification and that the materials will be developed and submitted by the COL applicant as part of the licensing process; therefore, this item is resolved. A specific COL action item was deemed unnecessary for this item, because 10 CFR Part 50 requirements are sufficiently explicit and the COL action item list is not intended to be exhaustive.

18.1.2 Final Standard Safety Analysis Report Review

The sources of information used for the final review described in this chapter were SSAR Chapter 18 through Amendment 34, GE's responses to the DSER (SECY-91-320) issues as documented in the SSAR, and GE's responses to the RAIs as documented in Chapter 20 of the SSAR. In support of design certification, GE personnel presented additional information on the standard design features to the NRC staff at a meeting during a visit to Japan to observe the Japanese ABWR CR prototypes ("Advanced Reactor Programs - ABWR Control Room Design" - presented to the United States Nuclear Regulatory Commission, M.A. Ross (GE), April 3 and 4, 1992, Tokyo, Japan - GE Proprietary). This information was latter summarized and included in the design certification application as Appendix 18G to the SSAR. These standard design features are evaluated in Section 18.3 of this report.

18.1.3 Development of Review Criteria

18.1.3.1 Objectives

As stated in the beginning of this chapter, complete detailed HSI design information would not be available for review before design certification. Certification is based on the staff's approval of GE's design and implementation process plan. For a design and implementation process plan to result in an acceptable design, it must contain (1) descriptions of all required HFE program elements for the design and development and implementation of the ABWR HSI and (2) DAC for the conformance reviews under the ITAAC.

To review the GE-proposed HFE process, the staff had to (1) assess if all the appropriate HFE elements were included, (2) identify what materials needed to be reviewed for each element, and (3) evaluate the proposed DAC and ITAAC to verify each of the elements. To conduct the review, the staff identified (1) which aspects of the HSI design process were required to ensure that HFE safety design in support of safe plant operation is achieved and (2) the review criteria by which each element is assessed. Review criteria independent of that provided by GE were required to ensure that GE's plan reflects currently accepted HFE practices and is a thorough, complete, and workable plan. To support such a review, the staff developed a technical basis for review of the HSI design process. The specific objectives of this effort were:

(1) To develop an HFE PRM to serve as a technical basis for the review of the GE-proposed process for certification. The HFE PRM is (a) based on currently accepted HFE practices, (b) well defined, and (c) validated through experience with the development of complex, high-reliability systems.

(2) To identify the HFE elements in a system development, design, and evaluation process that are necessary and sufficient for successful integration of the human component in complex systems.

(3) To identify which aspects of each HFE element are key to a safety review and are required to monitor implementation of the process.

(4) To specify the acceptance criteria by which HFE elements can be evaluated as design development progresses.

18.1.3.2 HFE PRM Development

The staff reviewed current HFE guidance and practices described in a wide range of nuclear industry and non-nuclear industry documents to identify important human factors program plan (HFPP) elements relevant to a design process review. A generic system development, design, and evaluation process was defined with eight key HFE elements that included criteria by which they could be assessed. This is referred to as the HFE PRM, or the review model.

The HFE PRM was based largely on applied general systems theory and the Department of Defense (DOD) systems development process (which is rooted in systems theory). Applied general systems theory provides a broad approach to system design and development that is based on a series of clearly defined developmental steps, each with clearly defined goals and specific management processes to attain them. Systems engineering has been defined as "the management function which controls the total system development effort for the purpose of achieving an optimum balance of all system elements. It is a process which transforms an operational need into a description of system parameters and integrates those parameters to optimize the overall system effectiveness." (F. Kockler, et al., Systems Engineering Management Guide (AD/A223 168), Defense Systems Management College, Fort Belvoir, Virginia, 1990.)

Use of the DOD system development process and procedure in the development of the HFE PRM was based on several factors. DOD policy identifies personnel as a specific component of the total system. A systems approach implies that all system components (hardware, software, personnel, support, procedures, and training) are given adequate consideration in the developmental process. A basic assumption is that the personnel component receives serious consideration from the very beginning of the design process. In addition, DOD has the most experience in applying HFE to the development of complex, technical systems (as compared with nonmilitary system developers); thus, its process is mature, formalized, and represents the most highly developed and well defined model of the HFE process available.

Within the DOD system, the development of a complex system begins with the mission or purpose of the system and the capability requirements needed to satisfy mission objectives. Systems engineering methods must be used as early as possible to develop the system concept and to define the system requirements. During the detailed design of the system, systems engineering ensures

- balanced influence of all required design specialties
- resolution of interface problems
- effective conduct of tradeoff analyses
- effective conduct of design reviews
- V&V of system performance

The effective integration of HFE considerations into the design is accomplished by providing (1) a structured topdown approach to system development that is iterative, integrative, and interdisciplinary and (2) a management structure that details the HFE considerations in each step of the overall process. A structured topdown approach to nuclear power plant (NPP) HFE is consistent with the approach to new CR design as described in Appendix B to NUREG-0700 and the more recent internationally accepted industry standard, International Electrotechnical Commission 964, for advanced CR design. The approach also is consistent with the recognition that human factors issues and problems emerge throughout the NPP design and evaluation process; therefore, human factors issues are best addressed with a comprehensive topdown program.

The scope of the HFE PRM excluded a training program development element because training is adequately addressed by existing NRC requirements. In addition, human reliability analysis was excluded and is addressed in Section 19.1 of this report.

The HFE PRM incorporates the requirements (as discussed below in Section 18.1.3.3) of 10 CFR 50.34(f)(2)(iii) as required by 10 CFR 52.47(a)(1)(ii). The HFE PRM is briefly described below and is included in Appendix J of this report.

18.1.3.3 HFE PRM Model Description

The overall purpose of the HFE PRM review is to ensure that

- the applicant has integrated HFE into plant development and design
- the applicant has provided HSIs that make possible safe, efficient, and reliable operator performance of operation, maintenance, test, inspection, and surveillance tasks
- the HSIs reflect "state-of-the-art human factors principles" as required by 10 CFR 50.34(f)(2)(iii).

State-of-the-art human factors principles are defined as those principles currently accepted by human factors practitioners. "Current" refers to the time when a program management or implementation plan is prepared. "Accepted" refers to a practice, method, or guide that is (1) documented in the human factors literature within a standard or guidance document that has undergone a peer-review process or (2) can be justified through scientific or industry research and practices.

All aspects of HSI will be developed, designed, and evaluated on the basis of a structured topdown system analysis using accepted HFE principles based on current HFE practices. HSI is used here in the very broad sense and shall include all operations, maintenance, test, and inspection interfaces and procedures materials.

The model developed to achieve this commitment contains eight elements:

(1) human factors engineering program management
(2) operating experience review (OER)
(3) system functional requirements analysis
(4) allocation of function
(5) task analysis
(6) human-system interface design
(7) plant and emergency operating procedure development
(8) human factors verification and validation

The elements and their interrelationships are illustrated in Figure J.1 of Appendix J of this report. Also illustrated are the minimal set of items to be submitted by the COL applicant for NRC staff review of the applicant's HFE efforts. A description of the purpose of each element follows.

<u>Element 1 - Human Factors Engineering Program Management</u>

To ensure the integration of HFE into system development and the achievement of the goals of the HFE effort, an HFE design team and an HFE program plan must be established to ensure the proper development, execution, oversight, and documentation of the HFE program. An HFE issue tracking system (to document and track HFE-related problems, concerns, and issues, and their solutions throughout the HFE program) will be established as part of the program plan. The HFE issue tracking system will be used as a mechanism to log ABWR-specific design issues as part of the COL applicant's overall design process.

<u>Element 2 - Operating Experience Review</u>

The accident at Three Mile Island (TMI) in 1979 and other reactor incidents have illustrated that significant problems in the actual design and design philosophy of NPP HSIs exist. There have been many studies as a result of these accidents and incidents. Utilities have implemented both NRC-mandated changes and additional improvements on their own initiative. However, the changes were formed on the basis of the constraints associated with backfits to existing CRs using early 1980s technology, which limited the scope of corrective actions that might have been considered (i.e., more effective changes can be made in the case of a new CR with the modern technology typical of advanced CRs). Problems and issues encountered in similar systems of previous designs must be identified and analyzed so that they are avoided in the development of the current system, or in the case of positive features, to ensure their retention.

Element 3 - System Functional Requirements Analysis

System requirements shall be analyzed to identify those functions that must be performed to satisfy the objectives of each functional area. System function analysis shall (1) determine the objective, performance requirements, and constraints of the design and (2) establish the functions that must be accomplished to meet the objectives and required performance.

Element 4 - Allocation of Function

Functions shall be allocated to take advantage of human strengths and to avoid functions that would be affected by human limitations. To ensure that functions are allocated according to accepted HFE principles, a structured and well-documented methodology of allocating functions to personnel, system elements, and personnel-system combinations shall be developed.

Element 5 - Task Analysis

Task analysis shall include the systematic study of the behavioral requirements of the tasks personnel are required to perform in order to achieve the functions allocated to them. The task analysis shall

- provide one of the bases for making design decisions (e.g., determining before hardware fabrication, to the extent practicable, whether system performance requirements can be met by combinations of anticipated equipment, software, and personnel)
- ensure that human performance requirements do not exceed human capabilities
- be used as basic information for developing procedures
- be used as basic information for developing staffing, skill, training, and communication requirements of the system
- form the basis for specifying the requirements for the displays, data processing, and controls needed to carry out tasks

Element 6 - Human-System Interface Design

Human engineering principles and criteria shall be applied along with all other design requirements to identify, select, and design the particular equipment to be operated, maintained, and controlled by plant personnel.

Element 7 - Plant and Emergency Operating Procedure Development

Plant and emergency operating procedures (EOPs) shall be developed to support and guide human interaction with plant systems and to control plant-related events and activities. Human engineering principles and criteria shall be applied along with all other design requirements to develop procedures that are technically accurate, comprehensive, explicit, easy to use, and validated. The types of procedures covered in the element are

- normal plant and system operations (including startup, power, and shutdown operations)
- abnormal and emergency operations
- alarm response

Element 8 - Human Factors Verification and Validation

Using HFE procedures, guidelines, standards, and principles, the acceptability of the final HSI design shall be evaluated as an integrated system. The integrated system includes all

- human-hardware interfaces
- human-software interfaces
- communications (human-human interfaces)
- procedures
- workstation and console configurations
- control room design
- remote shutdown system (RSS)
- design of the overall work environment

High fidelity with regard to the final design is expected (i.e., only minor differences between the actual final design and the evaluated design are acceptable). Validation should be accomplished through dynamic task performance of trained operating personnel using evaluation tools that are appropriate to the accomplishment of this objective as stated in Table 3.1 of the certified design material (CDM) and SSAR Appendix 18E.

18.2 Design Goals

18.2.1 General Discussion in the Standard Safety Analysis Report

The HSI design goals are described in SSAR Section 18.2. GE's "primary goal for the operator interface design is to facilitate safe, efficient, and reliable operator performance during all phases of normal plant operation, abnormal events, and accident conditions." It further states that, to achieve these goals, the HSIs will be designed and provided in a manner consistent with "good human factors engineering practices."

Within the context of this review, GE's eight specific design bases are considered to be design assumptions since they are "givens" and, as presented, have not been derived from analysis. They thus become design drivers. These design assumptions are evaluated in Section 18.2.2.2 of this report.

18.2.2 Analysis

18.2.2.1 DSER (SECY-91-320) Issue Resolution

Two DSER (SECY-91-320) issues related to the design goals are summarized below, including the resolution that was achieved.

<u>DSER (SECY-91-320) Issue 18.03: Number of Staff in the Control Room</u>

In the DSER (SECY-91-320), the staff stated that the number of main control room (CR) operating staff needed to be clearly established for the ABWR. Design Basis 1 in the original SSAR stated that for "normal operations, the ABWR shall be operable by one senior reactor operator who will be directly involved in manipulation of the reactor controls, one assistant CR SS (shift supervisor), one CR SS (shift supervisor), and two auxiliary equipment operators." The operating crew could be increased during accident conditions. In the DSER (SECY-91-320), the staff stated that the specification of a single operator at the control boards during normal operations was considered to be a significant design driver.

Evaluation: This issue is related to DSER (SECY-91-320) Issue 18.03 and was similarly resolved. GE satisfactorily clarified this issue in amended SSAR Section 18.2 by indicating that the ABWR operating crew will be consistent with the requirements of 10 CFR 50.54(m) and that two operators will be available during all phases of ABWR operation. Therefore, this item is resolved.

<u>DSER (SECY-91-320) Issue 18.14: Analysis To Justify Sole Operator Attentiveness and Rationale for Number of Operators at Main Console</u>

In the DSER (SECY-91-320), the staff stated that an appropriate analysis should be provided to justify how one operator at the main console will remain attentive to his/her duties, and that the maximum number of operators who are expected to monitor and operate the plant during an emergency at the main control console needs to be specified with the rationale to support this number. The staff stated that this rationale should be based on the function and task analysis performed to support the CR design. The staff required the rationale for the number of operators anticipated to monitor and control functions on the main control console while the operator is performing other functions.

Evaluation: GE satisfactorily clarified this issue in amended SSAR Section 18.2 by indicating that the ABWR operating crew will be consistent with the staffing requirements of 10 CFR 50.54(m) and that two operators will be available during all phases of ABWR operation. Therefore, this item is resolved.

18.2.2.2 Evaluation of the Current SSAR Design Bases

GE defined the following eight discrete design bases for use in the design development of the ABWR CR:

Design Basis 1 - The ABWR will be operated by two reactor operators, and four licensed operators will be on shift at all times, consistent with the requirements of 10 CFR 50.54(m).

Design Basis 2 - Efficient and reliable operation will be promoted through increased automation.

Design Basis 3 - Only proven technology will be used for the HSI design.

Design Basis 4 - Safety-related systems monitoring displays and control capability will meet the requirements for independence and electrical separation.

Design Basis 5 - The operator interface design will be highly reliable and provide functional redundancy.

Design Basis 6 - The principal functions of the ABWR safety parameter display system (SPDS) will be integrated into the HSI design.

Design Basis 7 - GDC 19 will be met (GDC 19 states that a control room shall be provided from which actions can be taken to operate the nuclear power unit and that equipment outside of the control room shall be provided to shutdown the reactor.)

Design Basis 8 - Design bases for the RSS will be specified in SSAR Section 7.4.

The staff reviewed the GE design bases which will be met through the design and implementation process described in SSAR Chapter 18. For two of the bases issues were identified which are described in detail below.

Design Basis 1 states that the ABWR will be operated by two reactor operators and that four licensed operators will be on shift at all times consistent with the requirements of 10 CFR 50.54(m). These will include a licensed senior reactor operator SS and assistant SS. While this may be a reasonable design goal on the basis of the preliminary analyses and evaluations conducted thus far, the capability of the main control console to accommodate two operators will have to be validated as part of HFE V&V activities (Element 8). Further, it will have to be verified that no more than two operators need to access the controls and displays at the main control console under all normal, abnormal, and accident conditions. The roles and responsibilities of the SS and assistant SS also will need to be defined, including their information requirements and access to displays. As reported in the preliminary validation tests, using the Japanese CR prototypes, in SSAR Appendix 18G, the main console would be crowded if more than two operators were required there. SSAR Table 18E-1 states that the COL applicant will validate that the CR design will support acceptable performance of all tasks assigned to the operating crew under a variety of plant conditions. In the DFSER the evaluation of the number of operators needing access to controls at the main console and the specification of the roles and responsibilities of the SS and assistant SS was identified as DFSER COL Action Items 18.2.2.2-1 and 18.2.2.2-2. GE has adequately incorporated these issues in the SSAR as Item 18.8.2 in SSAR Section 18.8. The staff agrees with GE's assertion that this issue (results of the evaluation shall be placed in the HFE Issue Tracking System - Item II.2 of SSAR Table 18E-1) should be resolved by the COL applicant as noted in SSAR Section 18.8.2 as part of the design and implementation process.

Design Basis 2 states that efficient and reliable operation will be promoted through increased automation. This design basis is acceptable only if it can be demonstrated that the increases in automation promote operational reliability and that automation is not introduced in such an arbitrary manner that it may impair human and/or system performance. This basis must be evaluated as part of the requirements of Element 4, "Allocation of Function." Decisions regarding which functions should be automated are more effectively made after the function analyses have been conducted and functions have been allocated as documented in NUREG/CR-3331, "A Methodology for Allocating Nuclear Power Plant Control Functions to Human or Automatic Control." In the DFSER, the evaluation of automation strategies and their effects on operator reliability was identified as COL Action Item 18.2.2.2-3. GE has acceptably incorporated this issue (results of the evaluation shall be placed in the HFE Issue Tracking System - Item II.2 of SSAR Table 18E-1) as noted in SSAR Section 18.8.3 as part of the design and implementation process.

No issues were identified for SSAR Design Bases 3 through 8 listed above. Therefore, these items are resolved.

18.2.3 Finding

As discussed in Section 18.2.2.1 of this report, DSER (SECY-91-320) Issues 18.03 and 18.14 are resolved. In addition, the staff identified three issues to be addressed, as part of the design and implementation process, as described in SSAR Table 18E-1.

- evaluation of the number of operators needing access to controls and displays at the main console

- specification of the roles and responsibilities of the SS and assistant SS

- evaluation of the impact of automation on operator reliability

SSAR Section 18.8 states that these issues will be resolved by the COL applicant after system functional requirements analyses, the allocation of functions, and the task analysis are performed.

18.3 Main Control Room Standard Design Features

18.3.1 General Discussion in the Standard Safety Analysis Report

The CR is characterized by 18 standard features, each of which is reviewed in Section 18.3.2.2 below. The features were derived from a system analysis and verified through V&V testing using two Japanese CR prototypes. The standard design features are described in SSAR Section 18.4, and their development is described in SSAR Sections 18.3.1 and 18.3.2.

A possible ABWR main CR is partially characterized in SSAR Appendix 18C. Since this serves as an example of how the standard features might be implemented, it has no specific application in the safety review of the ABWR for design certification. SSAR Appendix 18C should not be misinterpreted as providing any information specific to the ABWR CR design safety finding.

18.3.2 Analysis

18.3.2.1 DSER (SECY-91-320) Issue Resolution

DSER (SECY-91-320) Issues 18.02, 18.06, 18.08, 18.09, 18.11, 18.12, and 18.18 were related to the standard features that are described in SSAR Section 18.4. Each of these issues is summarized below, the path to resolution that was proposed in the discussions with GE after the DSER (SECY-91-320) was issued is given, and issue resolution is evaluated. These issues generally address the need for information regarding the design process and analyses leading to the standard features and their test and evaluation.

DSER (SECY-91-320) Issues 18.13 (HSI design requirements for the cathode ray tube (CRT), flat panel, and large-screen displays), 18.15 (CRT display information), and 18.17 (alarm suppression criteria, alarm points) address a level of design detail beyond the description of the standard features in the SSAR. Thus, these DSER issues are addressed in Section 18.9 because they will be resolved by the COL's design and implementation process.

<u>DSER (SECY-91-320) Issue 18.02: HSI Design and Evaluation Process and Issue 18.06: Tests and Analysis To Support Design Implementation</u>

In the DSER (SECY-91-320), the staff stated that additional detailed information regarding the HSI design and evaluation process was necessary (Issue 18.02). Additional detailed information also was necessary about the methods, criteria, and results of analyses performed to support the level and type of staffing, automation, and function allocation to achieve the goals of safe and reliable performance of the operating crew and overall system (Issue 18.06).

Evaluation: These issues are addressed in SSAR Section 18.4 and Appendix 18G. The standard design features were the result of a 5-year development program that included

- the preparation of implementation plans for major design and evaluation activities
- the derivation of general HSI requirements from the design of individual systems
- task analyses for safety-related functions based on manual operations
- a systematic allocation-of-function strategy based on workload analysis and an analysis of such task characteristics as degree of repetitiveness and complexity
- an analysis of current trends and technology assessments of the major CR features including approaches to automation, console design, video display units (VDUs), display techniques, large display panels, use of fixed-position displays, alarms, and CR layout

After the features were identified, GE assessed them in a validation testing program using two Japanese CR prototypes that had the standard features. Three teams of operators participated in the validation tests by performing a range of operational tasks, including normal operation, equipment failures, scrams, and accidents. Information collected for the validation tests included that collected by videotape and observations and operator opinion.

The test results generally supported the use of the standard features. However, the staff noted several limitations in the tests:

- The standard features were not individually tested; instead, the entire design as a package (which included standard features and other design detail) was evaluated. Also, the features as implemented in the validation tests were designed at a considerably greater level of detail when compared with their SSAR definition as standard features. Thus, it is possible that the same set of standard features (as defined in the SSAR) could be improperly designed and/or poorly integrated to result in an unacceptable design.
- The data collected were limited mainly to observations and subjective evaluations. A more complete performance measurement evaluation such as that described by HFE PRM, Element 8, was not used.
- The selection of accident and transient test scenarios was limited. More extensive tests will be needed but will be performed as part of the Element 8 validation test program (as defined in the HFE PRM).

- The results were expressed in general terms without the compilation of specific findings related to each of the general features (e.g., the relative merits and problems encountered for each specific feature).

- The test program showed that the main control console would be crowded if more than two operators were required there. The result is specific to the design configuration and test scenarios used in the test program. The SSAR level of detail does not include console dimensions; therefore, the test results from the Japanese prototypes do not directly pertain to the key feature as represented in the SSAR. Analysis of the console's suitability under maximum staffing demands should be evaluated as part of the detailed task analysis and validation test program.

Although the limitations noted above of the testing of the standard features were identified, the staff determined that the level of validation provided by the testing was sufficient to support the use of these features for the ABWR design because the suitability of these features would be revisited in more detail during the V&V of actual ABWR CR designs by the COL applicant.

DSER (SECY-91-320) Issues 18.02 and 18.06 became DFSER Confirmatory Item 18.3.2.1-1. GE has provided the description of the design development and validation testing in SSAR Appendix 18G. The staff agrees that GE's description of the design development and validation testing as discussed above is acceptable and therefore considers this item resolved.

In addition, the standard features are defined at a very general level and provide a general approach to CR design. They are not a final design specification and not at the level of detail needed for a final safety determination without consideration in the context of the design development process. Therefore, the detailed design implementation of the standard features and their integration into the CR and RSS designs will be included as part of the staff's review of the COL's DAC submittals related to the activities of HFE PRM Elements 6 and 8. In addition, the validation of the final design of the standard features is specifically identified as COL Action Item 18.8.5 in SSAR Section 18.8.

DSER (SECY-91-320) Issue 18.08: Standardized Features and Prototype Evaluation

In the DSER (SECY-91-320), the staff stated that additional detailed information was necessary to precisely indicate the aspects of the CR design that are part of the standardized design and that are unique to a COL applicant's implementation consistent with accepted human factors principles and practices and the requirements of 10 CFR Part 52. The staff believes development of a fully functional CR prototype of the standard design is appropriate in order to demonstrate acceptable human performance. Thus, there are three parts to this issue: (1) the aspects of the CR that are part of the standardized design, (2) the level of detail with which the standard features are described, and (3) the use of a prototype.

Evaluation: For Parts 1 and 2 of this issue, the staff conducted a feature-by-feature evaluation to determine the level of design detail that was supported by GE's design efforts. GE agreed to modify the description of the standard features to a level of detail supported by the design and evaluation efforts discussed with respect to Issue 18.06 above. The results were provided in GE's letter of February 18, 1992. This was DFSER Confirmatory Item 18.3.2.1-2.

SSAR Section 18.4 gives a revised description of the standard CR design features. The description has been modified to a level of detail commensurate with the test and evaluation program. The staff has determined that the design development description provided by GE is acceptable on the basis of the approved DAC.

In regard to Issue Part 3, the use of a prototype in design and evaluation is addressed as part of the design process discussed in Section 18.9 of this report.

GE has submitted the description of the design development and validation testing and revised the description of the standard features in SSAR Section 18.4. On the basis of its review, the staff finds GE's submittals acceptable; therefore, this item is resolved.

DSER (SECY-91-320) Issue 18.09: Operator Workload

In the DSER (SECY-91-320), the staff stated that GE had not indicated how the workload was defined and measured (in the context of allocation of functions) or what constitutes an appropriate operator workload level. The staff further stated that it was unclear how validating allocation-of-function decisions by a COL applicant at this late point in the design process could result in a standardized design. Thus, there are three parts to this issue: (1) workload definition for allocation-of-function studies, (2) determination of satisfactory workload, and (3) implications for postcertification evaluations that require modification of certified aspects of the design, in this case, allocation of function.

Evaluation: GE addressed this issue in its letter of February 18, 1992. Parts 1 and 2 of this issue are addressed in the evaluation of Issue 18.06 above. The specification of workload evaluations during postcertification is addressed in the discussion of the design process in Section 18.9 of this report and is required by the COL applicant in SSAR Table 18.E-1. This was DFSER Confirmatory Item 18.3.2.1-3. GE has submitted the information requested by the staff on operator workload in SSAR Appendix 18G. The staff has reviewed GE's submittal and found it to be acceptable; therefore, this item is resolved.

<u>DSER (SECY-91-320) Issue 18.11: Tests, Evaluations, Studies to Support Design Approaches</u>

In the DSER (SECY-91-320), the staff stated that information was needed on tests, evaluations, and trade studies performed to support the selection of design approaches (e.g., the use of touch-screen interfaces).

Evaluation: GE addressed this issue in its letter of February 18, 1992, and it is discussed under Issue 18.06. After discussions with the staff, GE eliminated several design details from the specification as standard features, including the use of touch-screen interfaces. Those that remain are those supported by the test program. DSER (SECY-91-320) Issue 18.11 was to be resolved subject to receipt of the amended SSAR. This was DFSER Confirmatory Item 18.3.2.1-4. GE has provided, in SSAR Appendix 18G, the information to support the staff's determination that GE's design details, as well as the appropriate DAC, are acceptable. Therefore, this item is resolved.

<u>DSER (SECY-91-320) Issue 18.12: Adequacy of HSI Design Requirements</u>

In the DSER (SECY-91-320), the staff stated that, in the absence of a systems analysis and test and evaluation results, there was no basis to evaluate the reasonableness and adequacy of the HSI design requirements from a top-down (or bottom-up) perspective.

Evaluation: GE addressed this issue in its letter of February 18, 1992, as is discussed under Issue 18.06 above. DSER (SECY-91-320) Issue 18.12 was to be resolved subject to receipt of the amended SSAR. This was DFSER Confirmatory Item 18.3.2.1-5. GE has provided, in SSAR Appendix 18G, the information to support the staff's determination that the HSI design requirements and appropriate DAC are acceptable. Therefore, this item is resolved.

<u>DSER (SECY-91-320) Issue 18.18: Safety Parameter Display System Design Scope</u>

In the DSER (SECY-91-320), the staff stated that, at the present stage of design, it could not determine if the ABWR safety parameter display system (SPDS) will meet all the NRC SPDS design criteria in NUREG-0737, Supplement 1, "Clarification of TMI Action Plan Requirements," 1982. The requirements regarding the SPDS in the ABWR CR are discussed in detail under "Standard Feature N."

The SPDS function and the list of critical parameters, as described in SSAR Section 18.4.6, did not include parameters that would provide operators with information about radioactivity control should there be a release of radioactive materials. The SSAR further stated that the COL applicant may provide a radioactivity release control information display. GE's initial approach to meeting NRC requirements for the SPDS function was not sufficient.

Evaluation: SPDS design is part of the ABWR standard features. Therefore, it is reviewed in the next section under "Standard Feature N." The commitment to the requirements of NUREG-0737, Supplement 1, has been incorporated into the description in SSAR Section 18.2, which states that the SPDS functions will comply with the NUREG report requirements. Therefore, this item is resolved.

18.3.2.2 Evaluation of the Current SSAR

As a result of the DSER (SECY-91-320) issue resolutions discussed above, GE agreed to

- clarify how the standard features were defined from the design process
- provide support for the validation of the standard features
- redefine the SSAR descriptions of the standard features to bring them in line with the supporting design and validation efforts

The list of standard feature descriptions below was the result of that process. It is important to reemphasize that a final safety determination for the detailed design of the standard features will be made by the staff using the DAC as part of ITAAC.

Standard Feature A

Feature Description - The use of a single, integrated control console staffed by two operators; the console has a low profile so that the operators can see over the console from a seated position.

Feature Evaluation - This feature, as presented, is generally supported by GE's design analyses and evaluations as discussed in Section 18.3.2.1 above.

Standard Feature B

Feature Description - The use of an on-screen control video display unit (VDU) for safety system monitoring and non-safety system control and monitoring that is driven by the plant process computer system.

Feature Evaluation - This feature, as presented, is generally supported by GE's design analyses and evaluations as discussed in Section 18.3.2.1 above.

Standard Feature C

Feature Description - The use of a separate set of on-screen control VDUs for safety system control and monitoring and separate on-screen control VDUs for non-safety system control and monitoring; the operation of these two sets of VDUs is entirely independent of the process computer system. Further, the first set of VDUs and all equipment associated with their functions of safety system control and monitoring are divisionally separate and qualified to Class 1E standards.

Feature Evaluation - This feature, as presented, is generally supported by GE's design analyses and evaluations as discussed in Section 18.3.2.1 above.

Standard Feature D

Feature Description - The use of dedicated function switches on the control console.

Feature Evaluation - This feature, as presented, is generally supported by GE's design analyses and evaluations as discussed in Section 18.3.2.1 above. GE's rationale for specifying the use of dedicated switches for the identified functions is consistent with human factors engineering practices and is acceptable for this application. This feature is described in SSAR Section 18.4.2.5, indicating that several different types of switches are used, incorporating a technology that has been retained from the previous BWR designs. The type of switch will be determined by the design implementation process addressed in Section 18.9 of this report.

Standard Features E, F, and G

Feature Descriptions -

E - Operator selectable automation of predefined plant operational sequences.

F - The incorporation of an operator selectable semiautomated mode of plant operations. This mode will provide procedural guidance to the operators using the plant operating procedures as a basis for that guidance. This "feature" is stated at the general level and will be further specified during the design implementation process.

G - The capability to conduct plant operations in an operator manual mode.

Feature Evaluation - These features, as presented, are generally supported by GE's design analyses and evaluations as discussed in Section 18.3.2.1 above. The features as stated are at the level of general requirements for the levels of automation available to the operator. The operator maintains the capability to assume manual control at any time. This basic approach is consistent with current human factors engineering practice and is acceptable for this application.

Standard Feature H

Feature Description - The incorporation of a large display panel that presents information for use by the entire CR operating staff.

Feature Evaluation - This feature, as presented, is generally supported by GE's design analyses and evaluations as discussed in Section 18.3.2.1 above. It should be noted that the safety significance of this display approach will be dependent on the final design of display formats and distribution of information between the main control console and the large display. This is addressed as part of the COL's HSI design and V&V ITAAC.

Standard Feature I

Feature Description - The inclusion on the large display panel of fixed-position displays of key plant parameters and major equipment status

Feature Evaluation - This feature, as presented, is generally supported by GE's design analyses and evaluations as discussed in Section 18.3.2.1 above. Information on the large panel will be designed so that it can be observed from the supervisor's console (which is farther from the panel than the main control console). As the COL applicant develops the design after certification, evaluations are required by the HFE PRM to confirm the allocation of information between the large panel and the main control console. This is addressed as part of the COL's HSI design and V&V ITAAC.

Standard Features J and K

Feature Descriptions -

J- The inclusion in the fixed-position displays of both Class 1E qualified (those that contain safety-related information) and non-Class 1E display elements.

K- The independence of the fixed-position displays from the plant process computer.

Feature Evaluation - The SSAR acceptably requires that the fixed-position displays of safety-related information conform to Class 1E standards and be independent of the plant process computer.

Standard Feature L

Feature Description - The inclusion in the large display panel of a large VDU that is driven by the plant process computer system.

Feature Evaluation - This feature, as presented, is generally supported by GE's design analyses and evaluations as discussed in Section 18.3.2.1 above.

Standard Feature M

Feature Description - The incorporation of a "monitoring only" supervisor's console that includes VDUs on which display formats available to the operators on the main control console also are available to the supervisors.

Feature Evaluation - The role of the supervisor and the supervisor's information requirements need to be more completely specified by the COL applicant before a determination of the design requirements of the console can be made. The present design of the main control console is for two seated operators. Making no provisions for control capability from the supervisor's console leads to the assumption that under no circumstances would control capability beyond that of the two operators be required. This will be considered by the COL applicant in the design process under Element 8 of the HFE PRM and will be resolved by the COL applicant on completion of the functional analysis as required by SSAR Table 18.E-2(II).

Standard Feature N

Feature Description - The incorporation of the SPDS function as part of the plant status summary information that will continuously be displayed on the fixed-position displays on the large display panel.

Feature Evaluation - This feature, as presented, is generally supported by GE's design analyses and evaluations as discussed in Section 18.3.2.1 above. Details pertinent to the SPDS are discussed below.

The staff reviewed GE's proposed approach to SPDS development in accordance with the requirements in 10 CFR 50.34(f)(2)(iv) and NUREG-0737, Supplement 1, and the guidance in NUREG-1342 "A Status Report Regarding Industry Implementation of Safety Parameter Display Systems" (1989), which describes SPDS implementation methods acceptable to the NRC staff as well as problem areas identified in operating plant SPDS reviews. GE describes its SPDS design in SSAR Section 18.4.2.11, as well as in DSER (SECY-91-320) Responses 3.b.1(n), 3.b.5, Table 3.b-1, and 5.e (GE letter of February 18, 1992).

The SPDS review for the ABWR is part of the staff's review to be conducted as part of HFE PRM Elements 6 (Design) and 8 (V&V). The ABWR CR and SPDS design, while not complete, is described in the SSAR. The discussion below addresses several SPDS requirements.

Paragraph 3.8a of NUREG-0737, Supplement 1, discusses the integration of the SPDS with related emergency response capabilities and includes the following comment on the SPDS:

> (1) Review the functions of the NPP operating staff that are necessary to recognize and cope with rare events that (a) pose significant contributions to risk, (b) could cause operators to make cognitive errors in diagnosing them, and (c) are not included in routine operator training programs.

This guidance was not specifically addressed in GE's response. Therefore, the COL applicant will need to consider incorporation of insights from the PRA into the SPDS selection process. For example, loss of power/station blackout (SBO) was very important to risk in the ABWR PRA (90 percent of core damage frequency (CDF) in the first revision of the PRA). Hence, there should be some monitoring of electric power sources as part of the SPDS. Also, with the addition of a gas turbine generator to the design to reduce the dominance of the SBO sequence, parameters related to the gas turbine should be included on the SPDS. Other safety-system failures that appear prominently in the accident sequences are those of the high-pressure core flooder, reactor core isolation cooling (RCIC) system, low-pressure core flooder, residual heat removal system, and automatic depressurization system (ADS).

Additionally, cognitive errors made by operators and items not covered in operator training programs would need to be addressed similarly to the risk-significant items. Addressing the criteria of Paragraph 3.8a of NUREG-0737, Supplement 1, will be part of the COL applicant's CR design responsibility. This was DFSER COL Action Item 18.3.2.2-1. The commitment to the criteria in NUREG-0737, Supplement 1, is specified in SSAR Section 18.2 and is incorporated in the DAC. This is acceptable to the staff.

Each of the paragraphs of Section 4.1 of NUREG-0737, Supplement 1, contains specific details pertaining to the SPDS. Some areas that require further consideration by the COL applicant in the design process are noted below.

- Paragraph 4.1a calls for a "concise" SPDS display. The concept of a "concise" display is further amplified in NUREG-1342, Section III.A.1. It is not clear from the description of the ABWR SPDS how the "concise" criteria will be met. This issue will be part of the COL applicant's detailed design development process.

- Paragraph 4.1c requires procedures and training on the SPDS. This will be addressed by the COL applicant in the development of procedures as addressed in SSAR Chapter 13.

- Paragraph 4.1d was addressed in GE's DSER (SECY-91-320) response, in which GE stated that the selection of information for inclusion in the SPDS display was based on the current BWR Owners Group (BWROG) EPGs rather than the ABWR EPGs. GE stated that this would be corrected to specifically address the ABWR EPGs. This was DFSER Confirmatory Item 18.3.2.2-1. The SSAR has been modified to state that "selection of the parameters for inclusion in the SPDS display is based upon the Advanced Boiling Water Reactor Emergency Procedure Guidelines . . . ." Therefore, this item is resolved.

- Paragraph 4.1f addresses the functional information required in the SPDS. While the specific parameters are to be determined by the COL applicant, GE has provided specific parameters in SSAR Chapter 18.4.2.11 that follow the NUREG-1342 recommendations, except as noted below.

  - The COL applicant should consider the need for some standby liquid control (SLC) indication along with the reactivity control covered in the ABWR parameters. Existing BWR designs do not have this indication; however, because the ABWR is a new reactor design, the COL applicant should consider the items listed in NUREG-1342 as "desirable enhancements." Also, the comment on PRA insights from Paragraph 3.8a above and the fact that SLC is needed to mitigate the anticipated transient without scram (ATWS) sequence (which is 31 percent of CDF in the probabilistic risk analysis) in the ABWR illustrates the importance of having SLC indication.

  - GE states that indication of radioactivity control will be selected by the first ABWR license applicant. Thus, parameter selection and display implementation will be the responsibility of the COL applicant as part of the design development.

DSER (SECY-91-320) Issue 18.18 concerning the SPDS was resolved, and the remaining issues listed above will be addressed as part of the COL applicant's design process through incorporation into the HFE issue tracking system. This was DFSER Confirmatory Issue 18.3.2.2-2. GE has included the SPDS design in SSAR Section 18.8, "COL License Information," as Item 18.8.4 and has committed to meeting NUREG-0737, Supplement 1, and the SPDS guidance in SSAR Sections 18.2 and 18.4.2.11. This is acceptable; therefore, this item is resolved.

<u>EXEMPTION FROM 10 CFR 50.34(F)(2)(iv) FOR AN SPDS CONSOLE</u>

The regulation 10 CFR 50.34(f)(2)(iv) requires that an application:

> Provide a plant safety parameter display console that will display to operators a minimum set of parameters defining the safety status of the plant, capable of displaying a full range of important plant parameters and data trends on demand, and capable of indicating when process limits are being approached or exceeded (I.D.2).

GE, as part of the ABWR SSAR, commits to meet the intent of this requirement. However, as discussed below, the functions of the SPDS will be integrated into the control room design rather than on a separate "console." The purpose of the requirement for an SPDS, as stated in NUREG-0737, Supplement 1, is to ". . . provide a concise display of critical plant variables to the control room operators to aid them in rapidly and reliably determining the safety status of the plant. ... and in assessing whether abnormal conditions warrant corrective action by operators to avoid a degraded core."

The ABWR design does not provide a separate SPDS, but rather, the functions of the SPDS are integrated into the overall control room display capabilities. In lieu of the requirements in 10 CFR 50.34(f)(2)(iv) for a "console," GE has proposed the following commitments in the ABWR SSAR:

(1) Section 18.2(6) states that the functions of the SPDS will be integrated into the design,

(2) Section 18.4.2.1(14) states that the SPDS function will be part of the plant summary information which is continuously displayed on the fixed-position displays on the large display panel,

(3) Section 18.4.2.8 states that the information presented in the fixed-position displays includes the critical plant parameter information, and

(4) Section 18.4.2.11 describes the SPDS for the ABWR and states that the displays of critical plant variables sufficient to provide information to plant operators about the following critical safety functions are continuously displayed on the large display panel as an integral part of the fixed-position displays:

(a) Reactivity control,
(b) Reactor core cooling and heat removal from the primary system,

(c) Reactor coolant system integrity,
(d) Radioactivity control, and
(e) Containment conditions.

The Commission may, upon its own initiative or at the request of an applicant, grant exemptions from the requirements of the regulations of Part 50. The exemption must comply with 10 CFR 50.12(a) criteria regarding special circumstances. An exemption from the "console" of the SPDS may be granted since not having an SPDS "console" (1) does not present an undue risk to the public health and safety, and is consistent with the common defense and security (10 CFR 50.12(a)(1)); and (2) special circumstances exist that application of the regulation to the ABWR design of the SPDS rule is not necessary to achieve the underlying purpose of the SPDS rule (10 CFR 50.12(a)(2)(ii)). As presented here, the staff uses the special circumstances in 10 CFR 50.12(a)(2)(ii) to justify the deviation from the regulation (exemption) for an SPDS "console" for the ABWR design.

In conclusion, the staff finds an exemption from the requirement for an SPDS "console" to be appropriate based upon (1) the description in the GE SSAR of the intent of the ABWR design to incorporate the SPDS function as part of the plant status summary information which is continuously displayed on the fixed-position displays on the large display panel; and (2) a separate "console" is not necessary to achieve the underlying purpose of the SPDS rule which is to display to operators a minimum set of parameters defining the safety status of the plant. The staff, therefore, finds that GE has adequately supported an exemption from 10 CFR 50.34(f)(2)(iv) because SSAR Sections 18.2(6), 18.4.2.1(14), 18.4.2.8 and 18.4.2.11 achieve the underlying purpose of the rule by ensuring that the SPDS functional requirements are satisfactorily incorporated in the control room design without a separate "console."

<u>Standard Feature O</u>

Feature Description - The use of fixed-position alarm tiles on the large display panel.

Feature Evaluation - This feature, as presented, is supported by GE's design analyses and evaluations as discussed in Section 18.3.2.1 above. In addition, the use of fixed-position alarms for key parameters is supported by research and industry experience with advanced alarm systems.

<u>Standard Feature P</u>

Feature Description - The application of alarm processing logic to prioritize alarm indications and to filter unnecessary alarms.

Feature Evaluation - This feature, as presented, is supported by GE's design analyses and evaluations as discussed in Section 18.3.2.1 above. GE's alarm prioritization scheme is based on three basic principles: mode suppression, redundancy suppression, and consequence suppression. GE does not propose advanced alarm suppression techniques, such as expert system-based analyses. Thus, the suppression is based on well-understood techniques that have been tested in other nuclear industry studies and have been found to be beneficial. In addition, the design specifies operator control over suppression. Operators may turn suppression on or off. As with other CR features, the alarm processing techniques must be verified and validated as required by the V&V ITAAC in Table 3.1 of the CDM and SSAR Appendix 18E analyses.

It should be noted that this standard feature only includes the application of alarm processing logic to prioritize alarm indications and to filter unnecessary alarms. The general issue of alarm processing is the subject of much industry and NRC research, and much additional research data pertaining to alarm implementation will be available for the COL applicant during the CR design process.

Standard Feature Q

Feature Description - A spatial arrangement between the large display panel, the main control console, and the shift supervisor's console allows the entire CR operating crew to conveniently view the information presented on the large display panel.

Feature Evaluation - This feature, as presented, is generally supported by GE's design analyses and evaluations as discussed in Section 18.3.2.1 above.

Standard Feature R

Feature Description - The use of VDUs to provide alarm information in addition to the alarm information provided by the fixed-position alarm tiles on the large display panel.

Feature Evaluation - This feature, as presented, is generally supported by GE's design analyses and evaluations as discussed in Section 18.3.2.1 above. Component-level alarms will be provided on main console VDUs while important alarms and system-level alarms will be provided on the fixed-position tiles on the large display panel. The alarms presented on the large panel also will be available on the VDUs.

18.3.3 Finding

The staff did not identify any safety issues in the description of the standard features for the level of design detail presented. It is important to note that the description of the standard features is at a general level and essentially presents a general approach to CR design. The features are not a final design specification and are not described at a level of detail sufficient for a final safety determination without consideration of the design development process. When the design becomes more detailed as part of the implementation and integration of the design process, the COL applicant will evaluate the standard features in accordance with Item 18.8.5 in SSAR Section 18.8, the V&V ITAAC in Table 3.1 of the CDM, and the SSAR Appendix 18E analyses.

DSER (SECY-91-320) Issues 18.02, 18.06, 18.08, 18.09, 18.11, and 18.12 became DFSER Confirmatory Item 18.3.3-1. As discussed above, GE amended the SSAR and included Appendix 18G, which satisfactorily addressed each of these issues. Therefore, these items are resolved.

DSER (SECY-91-320) Issue 18.18, "SPDS Integration With Related Emergency Response Capabilities," will be addressed by the COL applicant and has been included as Item 18.8.4 in SSAR Section 18.8. This is acceptable; therefore, this item is resolved.

18.4 Inventory of Controls, Displays, and Alarms

18.4.1 General Discussion in the Standard Safety Analysis Report

GE's initial SSAR contained insufficient information about controls, displays, and alarms to be utilized for the ABWR CR, resulting in a DSER (SECY-91-320) open issue. As part of the general resolution of the lack of CR detail, GE provided the detailed CR design implementation process through which the specific controls, displays, and alarms will be specified and designed. However, to provide an initial set of controls, displays, and alarms for transient mitigation before design certification, GE developed the inventory presented in SSAR Appendix 18F. This inventory was developed by analyzing the ABWR EPGs and the important operator actions specified as a result of the ABWR PRA analysis (refer to SSAR Section 19.D.7). Subsequently, GE described an additional fixed-position subset of these controls, displays, and alarms (i.e., a minimum inventory) for inclusion in the design description (DD) and ITAAC (CDM Section 2.7.1 and Table 2.7.1.A) for the main CR panels.

As part of the review, GE submitted to the staff detailed task analysis information (formerly SSAR Tables 18F-1 through 18F-12, Amendment 21) to support the identification of important displays, controls, and alarms for EPG implementation. The staff's review and its comments were given in the DSER (SECY-91-320) and DFSER, respectively. In response to the staff comments, GE submitted revised inventory analyses in SSAR Amendments 25 and 30 and in a letter providing proprietary task analysis information dated June 9, 1993. In Amendment 34, GE submitted SSAR Appendix 18H which provided the supporting analysis for the inventory of controls, displays, and alarms in SSAR Appendix 18F. The staff reviewed these submittals and confirmed that the issues described in the DSER and DFSER were satisfactorily addressed. Each confirmatory item is discussed below. The staff's review of the issue is complete, the DFSER confirmatory items are considered resolved, and the minimum inventory of displays, controls, and alarms is adequate.

18.4.2 Analysis

18.4.2.1 DSER (SECY-91-320) Issue Resolution

One DSER (SECY-91-320) issue related to the inventory is summarized below along with the path to resolution that was proposed in the discussions with GE. The evaluation of the issue is given in the following section.

DSER (SECY-91-320) Issue 18.26: Inventory

The staff raised general questions in the DSER (SECY-91-320) about the absence of level of detail for controls, displays, and alarms to be incorporated into the main CR. GE and the staff agreed that since a detailed design regarding controls and displays would not be available for the design certification review, GE would use the EPGs to develop an inventory of the key minimum information and control requirements for the operator to perform necessary safety-related functions. This approach to developing an inventory is discussed below.

18.4.2.2 Evaluation of the Current SSAR

18.4.2.2.1 Review Methodology

The staff reviewed SSAR Sections 18.3.1, 18.3.3, and Appendices 18A, 18F, and 18H to determine if the inventory in Appendix 18F provided a reasonable minimum set of fixed controls, displays, and alarms to adequately implement the EPGs for the ABWR.

The analysis methods used for this evaluation included

- EPG Review: Selected steps of the EPGs were compared with the corresponding portions of SSAR Tables 18H-1 through 18H-14 to determine accuracy and technical validity of conclusions.

- PRA/Human Reliability Analysis (HRA) Review: The PRA/HRA was compared with SSAR Tables 18H-1 through 18H-14 to determine whether significant human actions were selected and if the analysis was correct.

- Summary Table Review: Summary SSAR Tables 18H-11 through 18H-13 were selectively compared with SSAR Tables 18H-1 through 18H-10 for accuracy.

GE's analysis process for the EPGs provided a large amount of specified equipment. Each step, caution, and note in the large body of EPGs was separately reviewed, analyzed, and documented in a table containing 14 columns. A number of important controls, displays, and alarms were identified. On the basis of discussions with GE, the staff determined that the results of the analysis would be provided to the COL applicant for use in the CR design implementation process. This will help ensure that the important indications, controls and alarms derived from the analysis are appropriately implemented in the HSI design. The staff has determined that GE's analysis process is acceptable.

18.4.2.2.2 General Results

(1) Level of HSI Detail

Discussions in earlier SSAR Section 18F.1 indicated that GE had made significant HSI design implementation decisions regarding displays and controls in the main CR. This contradicted other statements in SSAR Chapter 18. For example, GE stated: "In Tables 18F-1 through 18F-12, the particular method of design implementation for each control, display, and alarm function is indicated in brackets as part of each relevant table entry." Particular implementation methods were described for hundreds of items. In general, the bracketed information in the inventory was more detailed than what the rest of the SSAR supported. GE also stated that "all remote control equipment of a particular system can be accessed and controlled by touch operations when the VDU operate mode is selected." However, touch operations are not part of the standard features identified in the SSAR. GE committed to revise the discussions in Table 18F-1 and the output from the inventory to make them consistent with the remainder of SSAR Chapter 18.

In Amendment 25 of the SSAR, GE provided a revised version of the inventory that addressed the staff concerns about the level of HSI detail. The staff reviewed the revisions and found GE had adequately removed the specification of design implementation for each control, display, and alarm and had adequately revised the introduction to Appendix 18E to reflect this. The staff found these revisions acceptable; therefore, the issue is resolved.

(2) Fixed Versus VDU Selection Rationale

For each step of the EPGs and each important PRA operator action, GE defined the information

(Column 4) and control functions (Column 5) necessary to perform that step. The parameter displays (Column 6), controls (Column 7), and alarms (Column 8) that are needed also were defined. GE further specified whether they were fixed or on VDUs. GE, however, did not provide information as to how the use of fixed displays or VDU displays was determined. GE committed to provide the appropriate criteria and rationale for this determination in SSAR Appendix 18F.

As a result of further discussion with the staff, GE agreed to remove the VDU designation from the inventory list since specifying an implementation strategy before implementing the human-machine interface design implementation process would be premature. In SSAR Amendment 30, GE provided a revised version of the inventory that addressed the staff concerns about VDU selection rationale by removing the reference to design implementation detail (e.g., VDU). This is acceptable; therefore, the issue is resolved.

(3) HFE Input

Although the inventory contains a list of key minimum displays, controls, and alarms necessary to carry out operator actions associated with the EPGs, the COL applicant will need to identify and further define additional detailed characteristics of these displays and controls (e.g., ranges, scales, physical dimensions, and actual information presentation) during the detailed task analysis and HSI design efforts. On the basis of its discussions with GE, the staff concluded that at the time in the design implementation process the inventory is used for the actual CR design, the COL applicant will review the minimum inventory against the results of the detailed task analyses to ensure that the appropriate information is displayed for operations activities. Any discrepancies identified from this comparison will be documented and resolved by the COL applicant through the design process. As a result, the minimum inventory has been added to the required information to be used in the detailed HSI design element of the human-machine interface design implementation process. Therefore, the staff concerns regarding HFE input have been addressed as noted before and this issue is resolved.

(4) Use of Important Operator Actions

GE's earlier SSAR Section 18F-2 listed five actions considered to be important based on the results of GE's PRA sensitivity study. However, the SSAR did not contain a discussion of the rationale for the selection of these as the important actions. Additionally, Item 7 of the PRA/HRA review of March 1991 contained a discussion of the number of human errors in the HRA. Six errors were noted to be the only ones described in any detail in the PRA. Of these six errors, which were treated as important in the PRA, only one was on the list of five actions considered in the inventory development (failure to inject water from the facility fire protection suppression system into the reactor pressure vessel (RPV)). The others related to inhibiting ADS during an ATWS, initiating SLC during an ATWS, controlling flow during an ATWS, failing to depressurize the reactor, and failing to isolate a failed heat exchanger. These errors did not appear to be analyzed for inclusion in the inventory. GE stated that some of these were covered in EPG steps and, hence, were addressed in the inventory. Others were not included because of automation decisions that removed the need for certain operator actions (e.g., ATWS control). GE agreed to provide a discussion of the rationale for the selection of the important operator actions included in the inventory in the PRA discussion in Chapter 19 of the SSAR. GE also agreed to identify each operator action already covered in the body of the EPGs that was also identified through the PRA study. Finally, GE committed to update the inventory based on any additional important human actions from PRA after the PRA/HRA had been completed.

As a result of further discussion with the staff, GE revised SSAR Section 19.D.7 to discuss the operator actions considered important in the ABWR PRA. Additionally, the GE revised detailed design files (e.g., SSAR Tables 18F-1 through 18F-11) submitted on June 9, 1993, to reflect the operator actions identified through the PRA. Currently, SSAR Appendix 18H contains those detailed design files. Therefore, the staff concerns about the use of important operator actions as identified in Chapter 19 have been addressed in the inventory described in Appendix 18F of the SSAR. The staff finds this approach to be acceptable; therefore, this issue is resolved.

(5) Scope of the Inventory

GE has developed a minimum set of fixed displays, controls, and alarms required to mitigate transients and accidents associated with the EPGs and the PRA sensitivity study. It should be noted, however, that GE committed to providing additional fixed displays (e.g., in SSAR Section 18.4.2.11 on the SPDS and in Section 18.4.2.11 on dedicated hardware alarm windows of entry conditions for EOPs) beyond those identified in SSAR Tables 18H-1 through 18H-14. GE also identified additional fixed displays as part of the inventory analysis that were not considered as part of the minimum inventory.

GE committed to revising the SSAR to describe the scope of the inventory as limited to the EPGs and the PRA sensitivity study and to clarify that additional required fixed displays would not be superseded by this inventory. GE also agreed to provide a discussion in the introduction to Appendix 18F of the SSAR on the scope of the inventory and to describe the integration of the inventory with the detailed design process by the COL applicant.

SSAR Amendment 30 included a discussion on the scope of the inventory that was consistent with the scope previously established by the staff for the inventory specification. In addition, the introduction to Appendix 18F notes that other requirements on the CR panel inventory stemming from other design commitments were not necessarily incorporated into the minimum inventory. The staff reviewed the revised introduction to this appendix and found that it addressed its concerns about the scope of the minimum inventory. The issue is resolved.

18.4.3 Findings

In the DFSER, the staff concluded that GE had developed an acceptable minimum set of displays, controls, and alarms that will mitigate transients and accidents associated with the EPGs and the PRA sensitivity study subject to the incorporation of the comments in the DSER (SECY-91-320). As a result of further discussions with GE, the staff determined that the revision to SSAR Amendment 30, Appendix 18F, addressed its concerns and was, therefore, acceptable. The minimum inventory of displays, controls, and alarms described by GE is adequate; therefore, DSER (SECY-91-320) Issue 18.26 is resolved.

18.5 Remote Shutdown System

18.5.1 General Discussion in the Standard Safety Analysis Report

SSAR Sections 7.4 and 18.5 describe the RSS. The RSS will use conventional hardwired controls and indicators to maintain diversity from the main CR. Discussion of other HSIs outside the CR are tied to GE's design and implementation process contained in SSAR Appendix 18E.

18.5.2 Analysis

18.5.2.1 DSER (SECY-91-320) Issue Resolution

DSER (SECY-91-320) Issue 18.19: Remote Shutdown System Design Rationale

In the DSER (SECY-91-320), the staff stated that additional information (i.e., tests, evaluations, and results) addressing how human performance is affected when operators are required to use mixed control and display technologies (i.e., digital and analog) during emergency plant operations was needed to support GE's position and rationale on the RSS design for the ABWR. (This issue is discussed further in Section 18.5.2.2 below.)

Evaluation: GE stated that the RSS will not use digital technology in order to maintain diversity from the CR. An assessment of the mix of analog and digital technologies in the plant as a whole was to be included in the postcertification test activities conducted by the COL applicant as part of the V&V element. The staff noted that independence (i.e., isolation and separation) and diversity were needed for the RSS. This was DFSER Confirmatory Item 18.5.2.1-1. GE has included information regarding the RSS as a COL action in Item 18.8.6 of SSAR Section 18.8 and in the HFE design acceptance criteria (ITAAC) of Table 3.1 of the CDM. This approach is acceptable; therefore, this item is resolved.

18.5.2.2 Evaluation of the Current SSAR

SSAR Sections 18.5 and 7.4 describe the RSS. GE intends the RSS to use conventional, hardwired controls and indicators to maintain diversity from the main CR. One generally acknowledged HFE design principle is to maintain consistency across HSIs when similar tasks are being performed in order to (1) minimize the time operators must spend "switching gears" to adopt operations to different HSIs for similar tasks and (2) minimize the potential for human errors that arise from incorrect transfer of learned activities from one HSI to another. In response to RAI Question 620.32 concerning the possible human factors implications of using analog hardware in the RSS design rather than a digital design consistent with the main CR, GE submitted a rationale for the diversity that included protecting "against the improbable event of common mode hardware or software failure in the plant instrumentation and control systems." The importance of diversity is acknowledged; however, caution should be exercised with regard to when the diversity is applied.

The staff was, therefore, concerned with GE's rationale for using mixed HSI technology. In its response to RAI Question 620.32, GE stated that the human factors testing of the RSS would be conducted during part of the plant power ascension test program and that, because RSS operations are relatively simple, training the operators to adjust to the analog RSS should not be an undue burden on them. GE did not give the basis or rationale to support these conclusions. As discussed above, the issue is being incorporated as Item 18.8.6 in SSAR Section 18.8 as a COL action item, thus addressing the staff's concern.

An NRC human factors generic issue includes the RSS. An evaluation of the risk significance of this issue for an RSS and a value-impact assessment of proposed human factors fixes is documented in NUREG/CR-5572, "An Evaluation of the Effects of Local Control Station Design Configurations on Human Performance and Nuclear Power Plant Risk" (1992). This study showed that functionally centralizing (integrating many functions into one panel) the RSS into one integrated panel was risk significant. Since the issue of RSS HSI design versus the CR HSI can only be addressed when design detail is developed, the staff expects that RSS functional centralization will be addressed by the COL applicant during the detailed design process described in SSAR Appendix 18E.

18.5.3 Finding

The design of the RSS is covered under the COL applicant's design and implementation process addressed by the HFE ITAAC in Table 3.1 of the CDM and detailed in SSAR Appendix 18E. DSER (SECY-91-320) Issue 18.19 is resolved on inclusion of Item 18.8.6, a COL action item, in SSAR Section 18.8.

18.6 Local Valve Position Indication

DSER (SECY-91-320) Issue 18.20: Local Valve Position Indication

In the DSER (SECY-91-320), the staff stated that the ABWR design should include complete local valve position indicator (VPI) based on accepted human factors principles and practices. "Local" in this instance means at the location of the valve in the plant. However, GE's commitment to VPI was unclear. This was identified as DSER (SECY-91-320) Issue 18.20.

Evaluation: In its response of October 1, 1991 (Response 5g), GE stated that only valves in the CR task analysis are required to have positive position indication. In the DSER (SECY-91-320) issues response dated February 18, 1992, GE stated that "the ABWR design does not include requirements for local position indication on all valves." However, the response did not indicate which local valves will have position indication.

In the process of plant and system design development, including CR task analyses, a nuclear power plant vendor will determine the valves that require remote position indication in the CR. In the large majority of cases, the valves with remote VPI in the CR are motor-operated valves. However, in some cases of hydraulically operated valves, pneumatic valves, and even manual or check valves (e.g., valves for the low-pressure coolant injection line for BWRs) have CR VPI. The ABWR should have a full complement of CR VPI, with specific details (e.g., which valves and what type of displays) determined by the COL applicant through the approved CR design implementation process plan.

While the NRC has focused on CR VPI, developments over the past few years have shown the importance of VPI at the valve itself. The recognition of the need to know valve position has come about through review of operating events and an increased attention to good human-systems interface design, including VPI for manual valves. These developments indicate the need for local VPI at various types of valves (e.g., manual and motor operated).

Brookhaven National Laboratory (BNL) reviewed historic records (BNL Report A-3972-4-91) to determine the extent and type of human engineering deficiencies that exist at local control stations (LCSs) in actual plants and to determine the type of plant-level problems caused by these deficiencies. Table 4.b of the BNL report catalogues a number of such problems as a result of inadequate VPI identified in licensee event reports (LERs), NUREG/CR reports, and EOP inspections. Table 5 in the BNL report also identifies problems related to inadequate VPI and discusses an Institute of Nuclear Power Operations recommendation for local means of visually verifying actual and normal position of valves.

During an NRC research project, related to potential upgrades to LCSs, both the costs and benefits associated with improved VPI for manual valves were investigated. This project showed that human factors improvements (which included VPI as a key component) to only selected important manual valves could result in risk changes of about 10^{-5} core damage events per reactor-year. Additionally, cost analyses showed these upgrades or backfits to the important manual valves were cost beneficial. Further, cost analysis showed that most of the costs were related to the backfit situation. That is, when local VPI was included as part of the original valve design specification, the added costs were minimal, especially when compared to the increased assurance that valves were positioned properly at all times. Thus, it was concluded that most local valves for new plants should have VPI. For certain small local manual valves (e.g., root valves), VPI was determined not to be necessary.

The staff's position is that remote VPI should be provided in the CR as discussed above and local VPI should be provided for

- all power-operated valves (e.g., motor, hydraulic, and pneumatic)
- all large manual valves (5 cm (2 in.) or greater)
- those small manual valves (< 5 cm (2 in.)) determined to be important

The staff noted that DSER (SECY-91-320) Issue 18.20 will be addressed by the COL applicant as part of the design development process. This was DFSER COL Action Item 18.6-1. GE has included a COL action item regarding VPI in Item 18.8.7 of SSAR Section 18.8. This is acceptable.

18.7 Unresolved and Generic Safety Issues

18.7.1 General Discussion in the Standard Safety Analysis Report

As required by 10 CFR 52.47, the applicant for design certification must demonstrate compliance with any technically relevant portions of the TMI requirements in 10 CFR 50.34(f), which is sometimes referred to as the CP/ML rule. 10 CFR 52.47 also requires proposed technical resolutions of USI/GSI. The safety issues that relate to human factors are addressed in Section 18.7.2, which follows.

GE discusses its approach and proposed resolution of these issues in Appendix 19B to the SSAR. The staff's initial review of the issues related to human factors is documented in DSER (SECY-91-320) Chapter 18. Several of the items were left outstanding at that time. This section will address the human factors aspects of these issues, as well as those items left as outstanding in the DSER (SECY-91-320).

18.7.2 Analysis

18.7.2.1 DSER (SECY-91-320) Issue Resolution

Two DSER (SECY-91-320) issues related to the standard features are summarized below along with the path to resolution that was proposed in the discussions with GE. The evaluation of these issues is given in the following section.

<u>Issue 18.23: Unresolved Safety Issues and Generic Safety Issues (USI/GSIs)</u>

In the DSER (SECY-91-320), the staff identified several USIs/GSIs (see discussion below in Section 18.7.2.2).

Evaluation: This issue was to be resolved as part of the design implementation process review (see discussion below in Section 18.7.2.2).

<u>Issue 18.24: Construction Permit/Manufacturing License (CP/ML) Rule Issues</u>

In the DSER (SECY-91-320), the staff identified several CP/ML rule issues (see discussion below in Section 18.7.2.2).

Evaluation: This issue was to be resolved as part of the design implementation process review (see discussion below in Section 18.7.2.2).

18.7.2.2 Evaluation of the Current SSAR

In its response dated February 21, 1992, GE addressed the specific outstanding items associated with 10 CFR 50.34(f) that were identified in the DSER (SECY-91-320). These are summarized below.

<u>USI/GSI Item HF-1.1 and related items</u>

The staff considers Items HF-1.1, "Shift Staffing," and I.A.1.4, "Long Term Upgrade of Operating Personnel and Staffing," to be beyond the scope of the design certification. The COL applicant will be responsible for addressing these issues as part of the licensing process. This was DFSER COL Action Item 18.7.2.2-8. As Item 18.8.1 in SSAR Section 18.8, GE has included a general COL action item to conduct the detailed HFE design according to the design and implementation process defined by the DD and Table 3.1 of the CDM and SSAR Appendix 18E. The staff interprets this process to include the analysis of these USI/GSI items; therefore, this approach is acceptable.

<u>USI/GSI Items HF-5.1 and related items</u>

The staff considers Items HF-5.1, "Local Control Stations," HF-01.3.4.a, "Local Control Stations," and II.K.1(5), "Safety Related Valve Position Indication," to be beyond the scope of the design certification; the COL applicant will need to address these issues. This was

DFSER COL Action Item 18.7.2.2-7. GE has included information regarding these two issues as Item 18.8.11, "Local Control Stations," and 18.8.7, "Local Valve Position Indication," in SSAR Section 18.8. This approach is acceptable.

USI/GSI Items HF-5.2, B-17, and related items

The staff has reviewed the GE detailed CR design process and finds that the COL applicant will address several USI/GSI items as part of the detailed design implementation process. Among these items are (1) HF-5.2, "Review Criteria for Human Factors Aspects of Advanced Control Room Instrumentation"; (2) B-17, "Criteria for Safety Related Operator Actions"; (3) HF-01.3.4b, "Interface Annunciators"; (4) HF-01.3.4c, "Operational Aids"; (5) HF-01.3.4d, "Automation and Artificial Intelligence"; and (6) HF-01.3.4e, "Computers and Computer Displays." This was DFSER COL Action Item 18.7.2.2-6. As Item 18.8.1 in SSAR Section 18.8 GE, has included a general COL action item to conduct the detailed HFE design according to the design and implementation process defined by the DD and Table 3.1 of the CDM and SSAR Appendix 18E. The staff interprets this process to include the analysis of these USI/GSI items. In addition, for resolving Item B-17, GE has included an evaluation of the adequacy of the HSI to provide necessary controls, displays, and alarms for the timely performance of critical tasks as Item 18.8.15 in SSAR Section 18.8. This approach is acceptable.

TMI Action Item I.A.4.2

10 CFR 50.34(f)(2)(i) corresponds to TMI Action Item I.A.4.2 on simulator capabilities. GE states that "simulator facilities for use in performing operator training are outside the scope of the standard plant design certification." This is consistent with the treatment of training in SSAR Chapter 13 and is acceptable because training is to be addressed by the COL applicant. This was DFSER COL Action Item 18.7.2.2-1. GE, with the agreement of the staff, has included the requirement that the operator training program meet 10 CFR Part 50 as Item 18.8.8 in SSAR Section 18.8. This approach is acceptable.

TMI Action Item I.D.1

10 CFR 50.34(f)(2)(iii) corresponds to TMI Action Item I.D.1 on CR design. Item I.D.4 and Item I.D.5(1) also relate to CR design issues. GE states in the SSAR that these issues will be addressed by the COL applicant in the detailed design implementation process. This is acceptable and is further addressed by the HFE design process discussed in Section 18.8 below. This was DFSER COL Action Item 18.7.2.2-3. GE has included detailed CR development in Table 3.1 of the CDM, Item 5, and in SSAR Appendix 18E. Information regarding this issue is given as a COL action item in Item 18.8.1 of SSAR Section 18.8. This approach is acceptable.

TMI Action Item I.D.2

Section 50.34(f)(2)(iv) corresponds to TMI Action Item I.D.2 on the SPDS. GE addressed this item in SSAR Section 18.4.2.11 (see the previous discussion of the SPDS in Section 18.3 of this report). Additionally, Item 125.I.3 in NUREG-0933 on SPDS availability will be addressed as part of the detailed CR design process. This was DFSER COL Action Item 18.7.2.2-4. GE has included the COL action of SPDS design in Item 18.8.4 of SSAR Section 18.8 and, in SSAR Sections 18.2 and 18.4.2.11, has committed to meeting NUREG-0737, Supplement 1, SPDS requirements. This approach is acceptable.

TMI Action Item I.D.3

10 CFR 50.34(f)(2)(v) corresponds to TMI Action Item I.D.3 on the status of bypassed and inoperable systems. This issue is covered in SSAR Chapter 7; however, the human factors aspects are not addressed. GE states that these will be addressed by the COL applicant in the detailed CR design implementation process. In addition, the COL applicant will be required to meet RG 1.47, which requires automatic indication at the system level of the bypassed or deliberately induced inoperable protection system and systems activated or controlled by the protection system. This was DFSER COL Action Item 18.7.2.2-5. GE has included a general COL action item to conduct the detailed HFE design according to the design and implementation process (defined by Table 3.1 of the CDM and SSAR Appendix 18E) in Item 18.8.1 of SSAR Section 18.8. The staff interprets this process to include the analysis of these USI/GSI items; therefore, this approach is acceptable.

TMI Action Items II.F.1 and II.F.2

These items address detailed CR design issues related to instrumentation (II.F.1, "Additional accident monitoring instrumentation" and II.F.2, "Instrumentation for Detection of Inadequate Core-Cooling"). GE states that these issues will be addressed by the COL applicant in the detailed design implementation process. This is acceptable and is further addressed by the HFE design process discussed in Section 18.8 of this report. This was DFSER COL Action Item 18.7.2.2-3. GE has included detailed CR development in Table 3.1 of the CDM, Item 5, and in SSAR Appendix 18E. The COL applicant's action with the process is included in Item 18.8.1 of SSAR Section 18.8. Further, GE has specifically identified COL actions for each of these TMI issues in Items 18.8.13 and 18.8.14, respectively, of SSAR Section 18.8. This approach is acceptable.

18.7.3 Finding

The staff concludes that detailed resolution of the TMI, USI, and GSI technical issues are adequately addressed by GE's design and implementation process as reflected in SSAR Appendix 18E and COL license information given in SSAR Section 18.8.

18.8 Emergency Procedure Guidelines

The staff approved Revision 4 of the BWROG EPGs, in an NRC letter (A.C. Thadani to D. Grace) dated September 12, 1988. This revision of the EPGs formed the basis for the ABWR EPGs. In Appendix 18B of the SSAR, GE submitted to the staff a list of differences between the ABWR EPGs and the BWROG EPGs, Revision 4. The following is a summary of the major differences and a description of the unresolved item, F18.8.4-1, when the advance version of the SER was issued. The resolution of this item is addressed in Sections 18.8.2 and 18.8.4 of this report.

18.8.1 Containment Temperature

Section CN/T of the BWROG EPGs contains guidelines for monitoring and controlling containment temperature by using available containment cooling. The control functions specified in this section - operation of containment cooling, initiation of suppression pool sprays, and RPV depressurization when containment temperature cannot be maintained below a prescribed limit - are specified in Sections SP/T and DW/T of the ABWR EPGs. The steps in these sections are carried out concurrently. Step SP/T-1 directs the operator to initiate all available suppression pool cooling. Step SP/T-3 directs the operator to depressurize the RPV in accordance with the heat capacity temperature limit (HCTL) curve. Step DW/T-2 calls for the initiation of containment sprays.

Section CN/T of the BWROG EPGs was developed specifically for the BWR/6 Mark III containment where temperature can be controlled by the previously stated control functions. Although the ABWR containment design incorporates the concept of a Mark III suppression pool, it is analogous to a Mark II BWR containment design for the purpose of controlling the wetwell space temperature.

Because Section CN/T is design specific for the BWR/6 Mark III containment and the control functions specified in this section are carried out concurrently in other sections in the ABWR EPGs, this section has been eliminated from the ABWR EPGs.

18.8.2 Venting

Revision 4 of the BWROG EPGs directs the operator to manually vent the containment before the primary containment pressure limit (PCPL) is reached in order to prevent uncontrolled containment failure. The PCPL is defined as the lowest pressure of the following: (1) pressure capability of the containment, (2) maximum containment pressure for vent valves to open and close, (3) maximum containment pressure at which safety/relief valves (SRVs) can open and remain open, and (4) maximum containment pressure for RPV vent valves to open and close for containment flooding. In the staff's SER for Revision 4 of the BWROG EPGs, the staff's stated goal was to limit venting to a "last resort" action. The major staff concern was centered on the appropriate containment pressure for venting. As a result, the venting pressure should be established to be as high as reasonably achievable.

In the ABWR, primary containment overpressure protection will not be accomplished through manual venting. This will be the function of the passive containment overpressure protection system (COPS), which is described in Section 19.2.3.3, "ABWR Containment Vent Design," of this report. The COPS is a passive system that is designed to actuate at 0.62 MPaG (90 psig) at 93 °C (200 °F) before the primary containment reaches a pressure corresponding to the ASME Service Level C of 0.67 MPaG (97 psig) at 260 °C (500 °F).

The COPS meets the primary containment overpressure protection philosophy described in the SER for Revision 4 of the BWROG EPGs because the set point is set as high as is reasonably achievable. Therefore, this change in vent design and the associated changes to the EPGs are acceptable.

The low pressure venting issue following the November 4, 1993, conference call with GE required GE to address the following items:

(1) Revise EPGs (PC/P) to show that venting is restricted to the 5 cm (2 in.) line in the drywell.

(2) Address suppression pool bypass mechanism through interconnection in the ACS and show the effect on the existing bypass analysis. Ensure that no other bypass pathways exist that have not been accounted for.

(3) Address containment isolation configuration of interconnection in the ACS between the wetwell and drywell. GE should justify automatic control of the ACS over a normally closed penetration ensuring containment integrity.

(4) Address suppression pool level issue in EPGs relating to the wetwell to drywell interconnection level. The EPGs appear to be inconsistent with the design.

(5) Address suppression pool level and pressure control EPGs for injection from sources outside of containment. The EPGs appear to direct conflicting actions in that SP/L-3.3 directs operators to stop injection from sources outside containment when the suppression pool level reaches 27.2 m (89.5 ft). Whereas, PC/P-6 directs operators to spray the containment when the water level reaches 27.2 m (89.5 ft) with use of sources external to the containment.

Resolution of Items 1 and 5 was provided in Amendment 33. GE provided revised design information for Items 2, 3, and 4 in SSAR Amendment 34. For Item 2, the suppression pool bypass mechanism was shown to be insignificant when compared to the suppression pool bypass capability discussed in Section 6.2.1.1.5 of the SSAR. For Item 3, GE provided a description of the isolation provisions. For Item 4, GE modified the EPGs to specify the correct water level. The staff finds the information acceptable; therefore, the low pressure venting portion of Open Item F18.8.4-1 is resolved.

18.8.3 Drywell Spray Initiation Limit

The drywell spray initiation limit (DSIL) curve, used in Revision 4 of the BWROG EPGs, is defined to be the highest drywell temperature at which initiation of drywell sprays will not result in an evaporative cooling pressure drop to below either (1) the drywell-below-wetwell differential pressure capability or (2) the high drywell pressure scram set point. The curve contains a single peak. The curve to the left of the peak is a function of the high drywell pressure scram set point, while the right-hand side of the curve is limited by the drywell-below-wetwell differential pressure capability.

The DSIL curve used in the ABWR EPGs differs from the one used in Revision 4 of the BWROG EPGs. The curve to the right of the peak has been eliminated for the ABWR because a large pressure differential between the wetwell and the drywell at the onset of drywell sprays is not likely since the drywell and wetwell sprays are designed to actuate simultaneously. There will be some differential pressure between the wetwell and the drywell as spraying proceeds because of different injection line-filling times and rates of spray flow into the two volumes.

GE calculated the differential pressure between the wetwell and drywell with concurrent spraying in the two volumes using the GE computer code SHEX. For the SHEX calculations, only six of eight vacuum breakers were assumed operable. Initial drywell pressure varied from maximum normal operating to maximum design, and relative humidity varied from 0 to 100 percent. Wetwell pressure varied from one atmosphere to normal operating and a wetwell temperature of 35 °F (95 °F) was assumed along with 100-percent relative humidity. The maximum drywell-to-wetwell pressure differential did not exceed -3.45 kPaD (-0.5 psid), which is less than the design valve of -13.7 kPaD (-2.0 psid) design value.

Actuation of drywell spray only is possible through a series of operator actions or following failure of the wetwell spray injection valve to open when containment sprays (both wetwell and drywell) are actuated. In a letter dated May 26, 1994, GE analyzed the effect of drywell spray actuation alone on the negative pressure capability of the containment and drywell/wetwell interface. GE concluded that the analysis supports all modes of spray actuation on the right side of the curve.

The staff finds the revised DSIL curve acceptable because the drywell and wetwell sprays will normally actuate simultaneously in the ABWR, thus eliminating the possibility of a significant pressure differential between the wetwell and the drywell at the onset of drywell sprays. This effect was confirmed by calculations performed by GE using the SHEX computer code. In the event of drywell spray actuation alone, GE provided analysis to demonstrate that the negative differential pressure capability of the containment would not be exceeded.

18.8.4 Heat Capacity Temperature Limit

The advance version of the SER identified Open Item F18.8.4-1 which consisted of two issues - the heat capacity temperature limit and the low pressure venting. The low pressure venting issue is discussed in Section 18.8.2 of this report.

For the ABWR, GE proposes the use of HCTL which would require reactor vessel depressurization when the suppression pool temperature reaches 103.9 °C (219 °F). Increasing the allowable suppression pool temperature before reactor vessel depressurization would begin permits the operation of the RCIC system for vessel injection when all other dedicated plant systems would be postulated to fail. This could occur during a SBO. This proposal raises several phenomenology issues related to hydrodynamic loads.

Concerns were raised regarding the suppression pool with steam discharges from SRV or the RCIC turbine exhaust during pump operation, since the suppression pool temperature has traditionally been restricted by HCTL curves to ensure reactor vessel depressurization at 66 °C (151 °F). With a steam discharge from a SRV quencher or RCIC turbine exhaust sparger at suppression pool temperatures approaching 103.9 °C (219 °F), should a unstable steam condensation process occur, the containment liner may be subjected to an excessive buckling load from a low pressure region occurring at the containment liner/suppression pool water interface. Also, a suppression pool bypass issue arises if a steam plume extends from the quencher to the suppression pool surface. This was HCTL part of Open Item F18.8.4-1.

To resolve the above issue on unstable collapse for an extended plume where the steam jet extends beyond the quencher condensation zone, GE relied on testing performed by Drs. Chun and Sonin as described in GE's submittal of January 20, 1994. In current generation reactors, steam discharge from a SRV quencher is condensed within a cylindrical region about the quencher arms called a condensation zone. The radius is in part of function of the amount of sub-cooling which exists within the suppression pool during a discharge, with the basis for sub-cooling being set forth in NUREG-0783, "Suppression Pool Temperature Limits for BWR Containments."

With the proposal of permitting steam discharge in the suppression pool at a higher pool temperature than what has been traditional discussed with the NRC, the staff pursued the potential consequences, as follows;

(1) Potential generation of a high quality steam plume extending beyond the quencher condensation zone with the plume being ingested by the ECCS inlet piping,

(2) Potential for sudden collapse and an unacceptably high condensation oscillation (CO) load, should the steam plume discussed above become sufficiently buoyant to detach from the quencher source,

(3) Potential extension of a steam plume from the quencher to the pool surface, thereby leading to a pool bypass path for particle scrubbing,

(4) Potential for a steam discharge from the RCIC turbine exhaust sparger, causing a CO or chugging load higher than the CO or chugging load for LOCA or discharge of all SRVs.

The ABWR ECCS inlet piping is located approximately 1 m (3.3 ft) below the SRV quencher devices. The staff concluded that steam plume injection by the ECCS is not possible due to the buoyant nature of the steam plume.

In the January 20, 1994, submittal, GE presented a discussion by Dr. A. Sonin addressing the potential large steam plumes drifting into cooler region of the pool, thereby creating the initial conditions for sudden collapse of the plume. The conclusion that was reached in the above stated paper was that the cooler regions (with respect to the local temperature about the quencher which is discharging) of the suppression pool are at a low elevation and azimuthally away from the quencher. During the quencher discharge, a circulatory drift motion occurs as the surrounding water is entrained into the plume. As the pool temperature increases during an extended discharge, the area about the expanding steam plume is expected to be relatively well mixed horizontally. Thermal stratification will be primarily vertical, with the highest temperature being in the warm buoyant layer near the surface and the colder temperature near the bottom of the pool. The staff finds that GE's position that a condition where the steam plume could move from a warm region of the pool to a significantly colder region to be implausible is justified based on the above stated paper and experiments performed by Drs. Chun and Sonin.

A question was also discussed concerning a steam plume extending from the SRV quencher to the pool surface and creating a potential suppression pool bypass pathway, which would negate any scrubbing action by the water. This issued appears to be unfounded based on the discussion in the January 20, 1994, submittal. The argument against the notion of a long continuous high quality steam plume extending to the pool surface appears unlikely due to turbulence about the buoyant high velocity jet formed at the quencher hole. The turbulence caused at the plume in close proximity to the quencher entrains water into the plume from the sides causing rapid loss of plume temperature and steam volume fraction with increasing distance from the quencher. In addition, independent calculations by the staff show that the wetwell airspace pressurization during pool heatup produces sufficient pressure to maintain a minimum of 40 degrees K subcooling in the pool, based on bulk pool temperature. The staff concludes that a pool bypass is not a concern based on the proposed HTCL curve.

RCIC turbine exhaust discharge during suppression pool heat was reviewed for the potential of producing pool boundary loads which may exceed LOCA loads. This issued was raised because the turbine exhaust is discharged into the pool via a sparger which may not have the same performance features for condensing steam as a X-

Quencher. GE evaluated the sparger and has determined that the potential for producing CO and chugging loads greater than a LOCA seems unlikely based on a steam mass flux of about 48 kg/m2-sec (9.83 lbm/ft^2-sec). At a mass flux of this magnitude, it is unlikely that CO and chugging loads could be produced which would be of the same magnitude as LOCA loads. In addition, the ABWR SSAR specifies a bounding asymmetric load case which assumes that half the drywell vents are 180 degrees out of phase with remaining vents for chugging. Based on the asymmetric loading requirement for chugging, the low mass flux at the sparger and that the sparger design has been in use on current-generation BWR without a reported failure or problem, the staff finds that the HCTL curve as drawn would not produce higher loads on the containment than LOCA loads currently assumed. This is acceptable and resolved HCTL portion of Open Item F18.8.4-1.

18.8.5 Primary Containment Flooding

An override statement has been placed in front of ABWR EPG Step C6-2. This step directs the operator to terminate all injection into the primary containment when drywell water level reaches the bottom of the RPV if containment radiation is greater than the core damage radiation level (CDRL) and RPV water level is below the top of the active fuel.

GE stated that containment flooding is to be terminated when the drywell water level reaches the bottom of the RPV during severe accident conditions when the core has melted through the vessel and dropped to the lower drywell. Flooding will be terminated to avoid covering the wetwell vent path, which has the containment rupture diaphragms. The wetwell vent is located at an elevation above the bottom of the RPV.

GE stated that it had set the CDRL at a level that will differentiate between an accident that has led to the melting of most of the fuel and an accident that results in damage of a few fuel pins. Once most of the fuel has melted the possibility of an ex-vessel event that leads to pressurization of the primary containment becomes much more likely. This pressurization may require actuation of the COPS. The staff finds this approach to primary containment flooding acceptable.

18.8.6 ATWS Stability Strategy

GE, in SSAR Amendment 32, submitted changes to the EPGs (departed from the Rev. 4 BWROG EPGs) incorporating ATWS stability strategy related to initiation of the SLC system and lowering of the RPV water level. This strategy is similar to that proposed by the BWROG for current BWRs and, in that context, is still a subject of discussion between the staff and the BWROG. On the basis of its review (see Section 4.4 of this report), the staff concludes that the proposed processes for boron insertion and lowering of the RPV water level below the feedwater sparger level are acceptable for reducing large power oscillations to acceptable levels, should they occur, and for reducing power level in general for ATWS.

18.9 Design and Implementation Process

The final CR design is an area of rapidly changing technology and it is important that the Tier 1 certified DD and the ITAAC do not "lock in" a CR design that would be obsolete at the time of construction. The staff's approach for ensuring CR human factors considerations for design certification is to "lock in" a design process and specific DAC that, if met, would result in a design that is acceptable. GE provides an overview of the design process in SSAR Sections 18.3.4 and 18.7 and gives details in SSAR Appendix 18E. Section 3.1 of the CDM gives the HFE ITAAC and DAC associated with the design and implementation process. At the time of the DFSER review, the ITAAC and DAC "Design Description" for Section 3.1 of the CDM were not complete. This was identified as DFSER Open Item 18.9.1-1. The material was submitted by GE in SSAR Amendment 32 and is reviewed below in this section.

The DAC will be in the same format as the ITAAC used for other systems in that they will specify the certified design process commitment and the method of demonstration that the commitment has been met. The method of demonstration will be inspection, test, or analysis against established acceptance criteria. The CR acceptance criteria will describe a formal design implementation process with test, analysis, and acceptance criteria.

General acceptance criteria are specified in the Tier 1 ITAAC and DAC material for each of the program elements shown, along with specific criteria for submitting several COL applicant technical reports. The general criteria are derived from accepted HFE practices. The Tier 2 SSAR (Appendix 18E) material contains applicable guidance documents for the development of the material for each of the program elements.

The staff performed the safety evaluation of the design and implementation process using Tier 1 HFE ITAAC and DAC described in Table 3.1 of the CDM and the Tier 2 criteria that appear in SSAR Tables 18E-1 through 18E-4. Each table provides the Tier 2 acceptance criteria. The design and implementation process review is described in Section 18.9.2, which follows. The Tier 2 material contains all the requirements identified in the Tier 1 material and gives additional detail regarding how Tier 1 requirements are satisfied. The evaluation of the Tier 1 material is provided in Section 18.9.3, which follows.

The Tier 2 commitments described in the SSAR provide methods and descriptions of the implementation of the Tier 1 requirements. The determination that the plant has been constructed in accordance with the design certification will require the use of the information contained in both the Tier 1 and Tier 2 documents. The Tier 2 material contained in SSAR Appendix 18E was used to support the safety finding with regard to the design and implementation process. Thus, any change to the SSAR Appendix 18E commitments by the COL applicant would involve an unreviewed safety question and, therefore, would require NRC review and approval before implementation. Any requested change to SSAR Appendix 18E commitments shall either be specifically described in the COL application or be submitted for license amendment after COL issuance.

18.9.1 General Discussion in the Standard Safety Analysis Report

In GE's design and implementation process (as described in SSAR Appendix 18E and Table 3.1 of the CDM), the following HFE activities are defined:

- HFE design team
- HFE program and implementation plans
- system functional requirements analysis
- allocation of functions
- task analyses
- HSI design
- human factors V&V

GE's key HFE design activities and their acceptance criteria were developed to address the staff's HFE PRM. It is important to note that the ITAAC and DAC description as presented in Table 3.1 of the CDM and the Tier 2 criteria that appear in SSAR Tables 18E-1 through 18E-4 have a scope limited to the main CR and the RSS (as agreed to by the NRC), whereas the HFE PRM has a broader scope. In addition, although procedures are within the scope of the HFE PRM as Element 7, GE has not included procedures within the scope of SSAR Appendix 18E. Procedure development is addressed as a COL responsibility in SSAR Section 13.5. In addition to scope, there are other differences between the staff's HFE PRM and GE's design and implementation process. However, a high degree of similarity exists between the two. The relationship between the PRM elements and the SSAR Tier 2 criteria is given below.

HFE PRM Element	SSAR Tier 2
1	Table 18E-1
2	Table 18E-1
3	Tables 18E-1, 18E-2
4	Tables 18E-1, 18E-2
5	Tables 18E-1, 18E-2
6	Tables 18E-1, 18E-3
7	Section 13.5
8	Tables 18E-1, 18E-4

The following review focuses on (1) the evaluation of the acceptability of the differences between GE's process and the staff's PRM and (2) the closure of DSER (SECY-91-320) issues. Since the GE process was evaluated against the criteria depicted in the PRM, the organization of the report design and implementation plan review follows the organization of the PRM (i.e., Elements 1 to 8).

18.9.2 Analysis

18.9.2.1 DSER (SECY-91-320) Issue Resolution

The initial SSAR provided little detailed information about the ABWR HSIs. As part of the general resolution of the lack of design detail, GE committed to provide a detailed HFE design and implementation process through which the HSIs will be designed and evaluated. This became DSER (SECY-91-320) Issue 18.25. However, because many other DSER (SECY-91-320) issues addressed design detail and were, therefore, beyond the scope of the certified design review, they too became incorporated into the design process review for subsequent consideration by the COL applicant as design development proceeds. These DSER (SECY-91-320) issues are identified in Table 18.2. As indicated previously in Section 18.1 of this report, an eight-element HFE PRM was developed to provide review criteria for the process. Table 18.2 shows which HFE PRM element addresses each DSER (SECY-91-320) issue. The issue is discussed in the section of this report identified in the table.

18.9.2.2 Evaluation of the Current SSAR

One general open issue pertained to Tier 2 guidance descriptions for each of the GE process elements. Element 1 (in the HFE PRM), for example, specifies the documents that are to be used as guidance. Each element has a similar specification. When the DFSER was issued, GE had not incorporated the specific documents, to serve as guidance, from the staff's review model into its process. The list of guidance documents was identified as DFSER Open Item 18.9.2.2-1.

Table 18.2 DSER (SECY-91-320) HFE issues to be addressed in the process plan

Issue Number	Issue	SER Section	HFE PRM Element
18.04	Operator and system reliability	18.9.2.2.8	Element 8 - Verification & Validation
18.05	Operator workload analysis	18.9.2.2.8	Element 8 - Verification & Validation
18:06	Tests and analysis to support design implementation	18.9.2.2.8	Element 8 - Verification & Validation
18.07	ABWR HFPP	18.9.2.2.1	Element 1 - HFE Program
18.08	Control room prototype	18.9.2.2.8	Element 8 - Verification & Validation
18.10	Detailed task analyses	18.9.2.2.5	Element 5 - Task Analysis
18.13	HSI design requirements for CRT, flat panel, and large-screen displays	18.9.2.2.6	Element 6 - Interface Design
18.15	CRT display information	18.9.2.2.6	Element 6 - Interface Design
18.16	PGCS display reliability	18.9.2.2.8	Element 8 - Verification & Validation
18.17	Alarm suppression criteria, alarm points	18.9.2.2.6	Element 6 - Interface Design
18.21	Procedure development	18.9.2.2.7	Element 7 - Procedure Development

GE has incorporated the list into SSAR Appendix 18E, and the staff finds it acceptable based on a comparison to the HFE PRM; therefore, all of the basic documents necessary to conduct the design and implementation process that are identified in the HFE PRM have been incorporated into SSAR Appendix 18E. Therefore, this item is resolved.

For each of the following elements, the staff compared the criteria in the HFE PRM with GE's element criteria. If GE's criterion differs from the "general criterion" described in the HFE review model, the analysis of the difference is given.

18.9.2.2.1 Element 1 - Human Factors Engineering Program Management

GE addresses human factors engineering program management in its HFE program plan of the HFE ITAAC and DAC described in Table 3.1 of the CDM and in the Tier 2 description in SSAR Table 18E-1(II). GE's description of this element is substantially the same as the HFE PRM, Element 1 description. However, the staff identified and evaluated the following three exceptions. It also evaluated DSER (SECY-91-320) Issue 18.07 as part of this element:

(1) Operating Experience Review (OER)

GE incorporated OER (HFE PRM Element 2) into its HFE program plan instead of presenting it as a separate element. The main purpose of this element is to ensure that the designer identifies HFE issues from current and past operating experience to be incorporated into the HFE issue tracking system. The merger of these elements does not compromise the contribution of the OER and is acceptable.

(2) Absence of System Safety Engineering Expertise on the Design Team

GE's design team does not include system safety engineering expertise as specified in the HFE PRM. In the May 1992 meeting, GE stated that the system safety engineers will be included as needed and not as full permanent members of the HSI design team. The staff finds GE's approach acceptable, because this area of engineering expertise is applicable to the HFE design rather than the other HFE elements of the process.

(3) Absence of Reliability, Availability, Maintainability, and Inspection Expertise on the Design Team

GE's design team does not include reliability, availability, maintainability, and inspection expertise as specified in the HFE PRM. In the May 1992 meeting, GE agreed to include this expertise in the description of the HSI design. This is acceptable because this area of engineering expertise is applicable to the HSI design rather than the other HFE elements of the process.

DSER (SECY-91-320) Issue 18.07: ABWR Human Factors Program Plan (HFPP)

In the DSER (SECY-91-320), the staff stated that SSAR Section 18.3 gave an outline for systems analysis and HSI design, but because the ABWR HFPP was not provided or referenced, the section contained little detail about actual analysis steps and procedures and discussion of results. The staff needed additional detailed information to complete its review.

Evaluation: The design and implementation process described in the SSAR and in the HFE certification design material DD document adequately addresses the HFPP and the types of analyses to be performed. All relevant portions of the HFE model were incorporated into the GE documents. Therefore, DSER (SECY-91-320) Issue 18.07 was to be resolved subject to receipt of final ITAAC and DAC. This was DFSER Confirmatory Item 18.9.2.2.1-1. GE has provided an acceptable plan as described in the ITAAC and in SSAR Appendix 18E (the following section addresses the technical justification of the plan's acceptability).

18.9.2.2.2 Element 2 - Operating Experience Review

As indicated above, GE addresses OER as part of the HFE program plan of the HFE ITAAC and DAC described in Table 3.1 of the CDM and in the SSAR Tier 2 description in SSAR Table 18E-1 (II).

In the DFSER the staff identified an open item with regard to the OER. The OER for a specific list of issues still needed to be performed. The list of items had not been identified when the DFSER review was conducted. The list of issues was DFSER Open Item 18.9.2.2.1-1. GE has provided an acceptable approach to OER in Table 18E-1(II) in SSAR Appendix 18E. The main treatment of OER is included as Article e in that SSAR section. For the first ABWR implementation, a list of issues is identified and organized into topical areas such as CR design, computers, CRT displays, and anthropometrics. These issues were identified through a preliminary OER performed by GE and are required, according to the design and implementation process, to be included in the COL HFE issues tracking system. The issues were identified on the basis of a review of pertinent industry experience literature and detailed control room design reviews of predecessor plants. The basis for the identification of operating experience is consistent with the requirements of the PRM. In addition, experience reviews are required in six selected areas, which GE has identified as ones for which further industry development is anticipated; thus, new issues are likely to emerge. These areas include on-screen controls, wide display panels, alarm prioritization systems, automation, VDU design, and workstation integration. The staff agrees that these are significant areas of HSI design and areas where significant technology development and operating experience are likely to occur. COL reviews in these areas include source material consistent with the PRM. New issues identified will be incorporated into the HSI tracking system. Subsequent ABWR COL applications would use the OER first implemented if no design changes were made. When changes are to be made, an OER is to be conducted using operator interviews and LERs of previous ABWR implementations. The staff interprets this requirement as also including a review of the documentation identified under Article 2e.(i)(b) of SSAR Appendix 18E; that is, the reviews will include industry experience, design, and research reports applicable to the areas of the design being modified. Where the changes represent a significant departure from the previous ABWR implementation, the staff expects applicable elements of the certified design and implementation process to be followed.

OER also is included as part of Article d of Appendix 18E for applications of HSI technologies that are different from those specified in SSAR Section 18.4.3.

On the basis of GE's commitment in SSAR Table 18E-1(II) for an OER to be conducted, this item is resolved.

18.9.2.2.3 Element 3 - System Functional Requirements Analysis

GE addresses system functional requirements analysis as part of the HFE ITAAC and DAC described in Table 3.1 of the CDM and in the Tier 2 description in Table 18E-1(III). GE's description of this element is substantially the same as the HFE PRM description for Element 3. However, the staff identified and evaluated the following three exceptions:

(1) Modification of General Criterion 2

This criterion defined critical functions as "those functions required to achieve major system performance requirements; or those functions which, if failed, could degrade system or equipment performance or pose a safety hazard to plant personnel or to the general public." GE has deleted from the definition the words "degrade system or equipment performance." The staff agrees with this change because the initial model definition was so broad that, if applied, it could define all functions as critical. GE's later definition of critical functions is acceptable.

(2) Elimination of General Criterion 8

This criterion stated: "The function analysis shall be kept current over the life cycle of design development." GE justified the deletion of this criterion by stating that an iterative approach to design development does not affect the review process. The staff review focuses on the acceptability of the end products of the design process as defined by the ITAAC requirements in Table 3.1 of the CDM, regardless of the number of iterations the designers went through to derive the requirements. Therefore, GE's deletion of this criterion is acceptable.

(3) Elimination of General Criterion 9

This criterion addressed verification that "all the functions necessary for the achievement of safe operation are identified" and that "all requirements of each function are identified." GE's justification for eliminating this criterion was that it was covered by the quality assurance (QA) requirements of Appendix B to 10 CFR Part 50. The staff finds this change acceptable because (1) general verification is covered by QA and is addressed in the Tier 2 analysis report criteria for this element and (2) the HFE design team evaluation report will adequately address the verification aspects of this element. Therefore, GE's deletion of this criterion is acceptable.

18.9.2.2.4 Element 4 - Allocation of Function

GE addresses function allocation as part of the HFE ITAAC and DAC described in Table 3.1 of the CDM and in the SSAR Tier 2 description in SSAR Table 18E-1(IV).

GE's description of this element is substantially the same as the HFE PRM description for Element 4. However, the staff identified and evaluated the following two exceptions:

(1) Elimination of General Criterion 5

This criterion stated: "Functions shall be re-allocated in an iterative manner, in response to developing design specifics and the outcomes of on-going analyses and trade studies." As in the case of the systems requirements analysis, as the design is modified as a result of design tradeoffs and analyses, it is important to reevaluate the functionassignments; however, an iterative approach to design does not affect the review process since the staff review focuses on the acceptability of the end products of the design process regardless of the number of iterations the designers went through to derive the results. Therefore, GE's deletion of this criterion is acceptable.

(2) Elimination of General Criterion 6

This criterion stated: "Function assignment shall be evaluated." Although this was eliminated as a general criterion (Tier 1) it was not removed from the Tier 2 requirements that elaborate on the evaluation specification. Because the general criterion did not provide any specific information and the evaluation of function assignment is maintained in the Tier 2 description, GE's deletion of this criterion is acceptable.

18.9.2.2.5 Element 5 - Task Analysis

GE addresses task analysis as part of the HFE ITAAC and DAC described in Table 3.1 of the CDM and in the SSAR Tier 2 description in SSAR Table 18E-1 (V). GE's description of this element is substantially the same as the HFE PRM description for Element 5. However, the staff identified and evaluated the following exceptions; it also evaluated DSER (SECY-91-320) Issue 18.10 as part of this element.

(1) Elimination of Part of General Criterion 4

This criterion stated: "The task analysis shall be iterative and become progressively more detailed over the design cycle. The task analysis shall be detailed enough to identify information and control requirements to enable specification of detailed requirements for alarms, displays, data processing, and controls for human task accomplishment." GE deleted the first sentence dealing with the iterative aspects of the analysis. This change is acceptable because an iterative approach to design does not affect the review process since the staff review focuses on the acceptability of the end products of the design process regardless of the number of iterations the designers went through to derive the results. Therefore, GE's partial deletion of this criterion is acceptable.

(2) Definition of PRA Critical Tasks

In SSAR Table 18E-1(V), "Task Analysis Implementation Plan," Item (1)(c) states: "Human actions which are identified through PRA sensitivity analyses to have significant impact on safety shall also be considered critical tasks." The staff expects that the definition of critical tasks will include all PRA-defined human actions that are critical and that the definitions will not be limited to sensitivity analyses alone. SSAR Section 19D.7.6 defines important operator actions from the ABWR PRA as derived from the ABWR Level 1, Level 2, fire, flood, seismic, and shutdown analyses. These actions are to be included in the definition of critical tasks for task analysis and the HFE efforts associated with subsequent elements - HSI design, procedure development, and V&V. Therefore, the definition of critical tasks is acceptable.

DSER (SECY-91-320) Issue 18.10: Detailed Task Analyses

In the DSER (SECY-91-320), the staff stated that detailed task analyses, which should cover the full range of normal and off-normal plant operations, had not been performed. GE stated that the task analysis will be performed as part of the hardware and software procurement and design implementation activities.

Evaluation: Although the design commitment, ITAAC, and general criteria for task analysis in Table 3.1 of the CDM and in the SSAR Tier 2 description adequately address detailed task analyses, DSER (SECY-91-320) Issue 18.10 was to be resolved subject to receipt of the final ITAAC and DAC. This was DFSER Confirmatory Item 18.9.2.2.5-1. Task analysis is described in Table 3.1 of the CDM, Item 4. The analyses will address "the range of plant operating modes, including startup, normal operations, abnormal operations, transient conditions, low power and shutdown operations." This approach is acceptable and this item is resolved.

18.9.2.2.6 Element 6 - Human-System Interface Design

GE addresses HSI design as part of the HFE ITAAC and DAC described in Table 3.1 of the CDM and in the Tier 2 description in SSAR Table 18E-1(Item VI). The description of this element is different from the HFE PRM Element 6 description. The staff identified and evaluated the following two exceptions. In addition, it evaluated DSER (SECY-91-320) Issues 18.13, 18.15, and 18.17 as part of this element.

(1) Elimination of General Criterion 6

This criterion stated: "The selection and design of HSI hardware and software approaches shall be based upon demonstrated criteria that support the achievement of human task performance requirements. Criteria can be based upon test results, demonstrated experience, and trade studies of identified options." GE eliminated this element from its process description. It states that the list for Element 6 of specific documents to serve as guidance will be used by the COL applicant for the selection and design of HSI hardware and software approaches. The staff finds this to be an acceptable approach; therefore, GE's deletion of this criterion is acceptable.

(2) Elimination of General Criterion 7

This criterion stated: "HFE standards shall be employed in HSI selection and design. Human engineering guidance regarding the design of particular features, shall be developed by the HSI designer to (1) insure that the HSIs are designed to currently accepted guidelines and (2) insure proper consideration of human capabilities and limitations in the developing system. This guidance shall be derived from sources such as expert judgement, design guidelines and standards, and quantitative (e.g., anthropometric) and qualitative (e.g., relative effectiveness of differing types of displays for different conditions) data. Procedures shall be employed to ensure HSI adherence with standards." GE states that the specific documents listed in SSAR Appendix 18E will be used by the COL applicant to comply with this general criterion. The staff finds that the information in SSAR Appendix 18E related to this issue and GE's deletion of this criterion is acceptable.

DSER (SECY-91-320) Issue 18.13: HSI Design Requirements for Cathode Ray Tube (CRT), Flat Panel, and Large-Screen Displays

In the DSER (SECY-91-320), the staff stated that additional detailed information was needed on the ABWR HSI design requirements for the control station CRT, flat panel, and large-screen displays.

Evaluation: Although the design commitment, ITAAC, and general criteria for HSI design described in the CDM and in the SSAR Tier 2 description adequately address the detailed design of the HSI, DSER (SECY-91-320) Issue 18.13 was to be resolved subject to receipt of the final ITAAC and DAC. This was DFSER Confirmatory Item 18.9.2.2.6-1. Design Commitment 5 of Table 3.1 of the CDM and SSAR Appendix 18E states that the HSI will be based on requirements derived from task analyses and designed using HFE criteria and guidance. The staff has compared GE's Design Commitment 5 and SSAR Appendix 18E with the HFE PRM and found GE's description to be acceptable. Therefore, this item is resolved.

DSER (SECY-91-320) Issue 18.15: CRT Display Information

In the DSER (SECY-91-320), the staff stated that no details of the CRT displays were provided to permit visualization of the actual information available to the operator. The staff needed this information to complete its review.

Evaluation: The design commitment, ITAAC, and general criteria for HSI design described in the DD of the CDM and in the SSAR Tier 2 description adequately address the detailed design of the HSI. DSER (SECY-91-320) Issue 18.15 was to be resolved subject to receipt of the final ITAAC and DAC. This was DFSER Confirmatory Item 18.9.2.2.6-2. Design Commitment 5 of Table 3.1 of the CDM and SSAR Appendix 18E state that the HSI will be based on requirements derived from task analyses and designed by the COL applicant using HFE criteria and guidance. This is acceptable based on a comparison of the design commitments with the HFE PRM. Therefore, this item is resolved.

DSER (SECY-91-320) Issue 18.17: Alarm Suppression Criteria, Alarm Points

In the DSER (SECY-91-320), the staff stated that additional detailed information about the ABWR alarm suppression criteria and rationale used to determine the limit number of alarm points that operators can simultaneously recognize was necessary for the staff to complete its review.

Evaluation: Although the design commitment, ITAAC, and general criteria for HSI design described in the CDM and in the SSAR Tier 2 description adequately address the detailed design of the HSI, DSER (SECY-91-320) Issue 18.17 was to be resolved subject to receipt of the final ITAAC and DAC. This was DFSER Confirmatory Item 18.9.2.2.6-3. Design Commitment 5 of Table 3.1 of the CDM and SSAR Appendix 18E state that the HSI will be based on requirements derived from task analyses and designed using HFE criteria and guidance. This is acceptable based on a comparison of Design Commitment 5 with the HFE PRM. Therefore, this item is resolved.

18.9.2.2.7 Element 7 - Plant and Emergency Operating Procedure Development

As stated in Section 18.9.1 above, GE has not included procedure development in the scope of its HFE design and implementation plan. Since procedure development is addressed as a COL applicant responsibility in SSAR Section 13.5, it is evaluated in Chapter 13 of this report.

18.9.2.2.8 Element 8 - Human Factors Verification and Validation (V&V)

GE addresses human factors V&V as part of the HFE ITAAC and DAC described in Table 3.1 of the CDM and in the SSAR Tier 2 description in Tables 18E-1(VII). GE's description of this element is the same as the HFE PRM description for Element 8. However, the staff identified and evaluated the following exception. It also evaluated DSER (SECY-91-320) Issues 18.04, 18.05, 18.06, 18.08, and 18.16 as part of this element.

Elimination of PRA/HRA-Defined Critical Actions From General Criterion 8

This criterion stated: "A verification shall be made that all critical human actions as defined by the task analysis and PRA/HRA have been adequately supported in the design. The design of tests and evaluations to be performed as part of HFE V&V activities shall specifically examine these actions." Risk-critical human actions (those to which the plant design is especially sensitive in a risk model) should receive special attention in the V&V process. GE committed to having all critical tasks confirmed as part of the V&V process and defined critical tasks under the task analysis element to include all PRA/HRA items included in SSAR Appendix 19D, Section D.7. These would include those operator actions that had significant safety impact. The staff considered this approach acceptable. The incorporation of the commitment was DFSER Confirmatory Item 18.9.2.2.8-1. SSAR Table 18E-1 states that critical tasks will include those human actions identified through PRA sensitivity analyses to be critical. Table 3.1 of the CDM, Item 6a, (4)(a), states that one V&V objective will be the "confirmation that the identified critical functions can be achieved using the integrated HSI design." GE's deletion of PRA/HRA-defined critical actions from this criterion is acceptable; therefore, this item is resolved.

DSER (SECY-91-320) Issue 18.04: Evaluation of Operator and System Reliability During Shift From Normal to Abnormal Operations

In the DSER (SECY-91-320), the staff stated that the net effect on operator and system reliability should be evaluated for normal operations and for the shift, or transition, from normal to emergency operations.

Evaluation: Incorporation of this analysis into the V&V ITAAC was DFSER Confirmatory Item 18.9.2.2.8-2. Table 3.1 of the CDM, Item 6a, states that the dynamic performance evaluations conducted as part of validation "shall be conducted over the range of operational conditions and upsets." The staff interprets this as including transitions from normal to emergency operations to permit the evaluation of the crew's ability to assume plant control under abnormal conditions, and notes that a detailed examination of operator transition from normal to abnormal operations will be performed during the V&V of the main CR and RSS by the COL applicant. Therefore, this item is resolved.

DSER (SECY-91-320) Issue 18.05: Operator Workload Analysis

In the DSER (SECY-91-320), the staff stated that GE did not make it clear what analyses had been performed in support of the design and development of the ABWR and what tests and analyses are yet to be done by the COL applicant.

Evaluation: In the DFSER, the staff stated that the V&V analyses conducted by the COL applicant as part of the HFE ITAAC and DAC described in Table 3.1 of the CDM specifically would address this issue, which would be resolved subject to receipt of the final ITAAC and DAC. This was DFSER Confirmatory Item 18.9.2.2.8-3. Design Commitment 6, Acceptance Criteria 6a(6)(d) of Table 3.1 of the CDM, and SSAR Appendix 18E state that workload will be used as a performance measure in dynamic performance tests. On the basis of a comparison of Design Commitment 6 and SSAR Appendix 18E with the HFE PRM, this item is resolved.

DSER (SECY-91-320) Issue 18.06: Tests and Analysis To Support Design Implementation

In the DSER (SECY-91-320), the staff stated that additional detailed information was needed about the methods, criteria, and results of analyses that support the level and type of staffing, automation, and function allocation to achieve the goals of safe and reliable performance of the operating crew and overall system. The staff further stated that the design bases in SSAR Section 18.2 would be more appropriate as design requirements if they had been derived and justified on the basis of the systems analysis. The staff considered it more appropriate to develop design bases that are stated in terms that would help achieve the primary goal of developing interfaces (and a system) that make possible safe, efficient, and reliable operator performance. The bases could be described in "operator-centered" terms that can objectively be linked with achieving the design goals and serve as criteria for test and evaluation activities. There were two aspects of this issue to consider: (1) the analyses conducted to date and (2) the analyses that will be done in the future.

Evaluation: The analyses conducted to date were included in SSAR Appendix 18G and evaluated by the staff as acceptable (see detailed discussion of this review in Section 18.3.2.1 above). Although the V&V analyses to be conducted by the COL applicant as part of the HFE ITAAC and DAC described in Table 3.1 of the CDM specifically address this issue, it was to be resolved subject to receipt of the final ITAAC and DAC. This was DFSER Confirmatory Item 18.9.2.2.8-4. Design Commitment 6 of Table 3.1 of the CDM and SSAR Appendix 18E include a commitment that the V&V analyses will be performed. These analyses are acceptable (as stated above) on the basis of a comparison of Design Commitment 6 and SSAR Appendix 18E with the HFE PRM. Therefore, this item is resolved.

DSER (SECY-91-320) Issue 18.08: CR Prototype

In the DSER (SECY-91-320), the staff stated that development of a fully functional CR prototype of the standard design was appropriate to demonstrate acceptable human performance.

Evaluation: Although V&V analyses conducted by the COL applicant as part of the HFE ITAAC and DAC described in Table 3.1 of the CDM specifically require prototype evaluation, this issue was to be resolved subject to receipt of the final ITAAC and was DFSER Confirmatory Item 18.9.2.2.8-5. Design Commitment 6 of Table 3.1 of the CDM requires dynamic task performance test evaluations to be performed for HSI validation. SSAR Table 18E-1(VII), Item (1)(d), states that the dynamic task performance evaluations will be performed using "dynamic HSI prototypes, i.e., prototypical HSI equipment which is dynamically-driven by real time plant simulation computer models." On the basis of a comparison of Design Commitment 6 and SSAR Appendix 18E with the HFE PRM, the staff finds that GE's proposal is acceptable and, therefore, this item is resolved.

DSER (SECY-91-320) Issue 18.16: Power Generation Control System (PGCS) Reliability

In the DSER (SECY-91-320), that staff stated that additional detailed information was necessary regarding the reliability of the PGCS and the effect on operator performance and workload should it malfunction.

Evaluation: Since system malfunctions are required to be analyzed as part of the V&V effort under HFE model Element 8, the COL applicant will analyze malfunctions of the PGCS. This issue was resolved as DFSER COL Action Item 18.9.2.2.8-1 to be addressed as part of the HFE issue tracking system. GE has included this analysis in Item 18.8.10 of SSAR Section 18.8, which the staff finds acceptable. Therefore, this item is resolved.

18.9.3 Design Description Tier 1 ITAAC and DAC Review

While the above review was directed toward the acceptability of the design and implementation process as a whole, the staff performed a separate evaluation of the HFE ITAAC and DAC provided in Table 3.1 of the CDM to ensure that significant features of the design certification application contained in the SSAR were captured by Table 3.1 of the CDM. It should be noted that the materials reviewed in SSAR Appendix 18E were used to support the safety finding with regard to the design and implementation process. Thus, any change to the commitments in SSAR Appendix 18E would involve an unreviewed safety question and, therefore, require NRC review and approval before implementation. Any requested change to commitments in SSAR Appendix 18E shall either be specifically described in the COL application or be submitted for license amendment after COL issuance.
The review of the SSAR using the HFE PRM led to the staff's conclusion that the design and implementation process contained the necessary and sufficient aspect of an HFE program that were sufficient to result in an acceptable HSI design. The general guidance in SECY-92-287 was used to support the review of Table 3.1 of the CDM. Table 3.1 of the CDM was compared to the major PRM elements to determine whether they were captured. No omissions were identified. Table 3.1 of the CDM was then evaluated to ensure that they accurately reflected the design and implementation process and that they were at a level of detail consistent with the staff's intent to not constrain the use of state-of-the-art, proven technology at the time the HSI is designed (one of the stated intents of the DAC process). All necessary and sufficient ITAAC were identified based upon comparison with the HFE PRM, and no concerns were identified.

Therefore, the staff concludes that the design commitments in the HFE ITAAC and DAC accurately summarize the DD for HFE; that the inspections, tests, and analyses identified are acceptable methods for determining whether the design commitments have been met; and that the acceptance criteria are sufficient to establish, if they are met, that the design commitments have been met.

18.10 Conclusion

The staff has reviewed the HSI design development and implementation process presented by GE in SSAR Sections 18.0 through 18.8 and Appendices 18A, 18B, 18D, 18E, 18F, 18G, and 18H, up through SSAR Amendment 34. SSAR Appendix 18C represents only one illustration of a possible CR design for the ABWR; therefore, it is NOT a part of the certified design and was NOT subject to the review process. The staff concludes that the design and implementation process discussed in the SSAR describes an acceptable HFE program, and if applied, will result in an acceptable HSI designs for the main CR and RSS. In addition, the design commitments and ITAAC in Table 3.1 of the CDM accurately summarize the minimum HFE requirements for an acceptable design, development, implementation, and V&V process for the main CR and RSS. All previously identified DSER (SECY-91-320) and DFSER issues are resolved.

19 SEVERE ACCIDENTS

19.1 Probabilistic Safety Assessment

19.1.1 Executive Summary

As part of its advanced boiling water reactor (ABWR) design certification application, GE Nuclear Energy (GE) has performed a design-specific probabilistic risk assessment (PRA) as required by 10 CFR 52.47(a)(1)(v). GE submitted a Level 3 PRA (i.e., the PRA calculated core damage frequencies, conditional containment failure probabilities, and conditional offsite consequences) that addresses internal initiating events. The PRA also evaluates seismic, internal flood, and fire-initiating events.

The staff reviewed the ABWR PRA to investigate design insights and to determine its quality. The staff concluded that the quality of the ABWR PRA is adequate for supporting and improving the ABWR design process; providing relative importance of sequences (as well as identifying important structures, systems, and components (SSCs)) leading to core damage or containment failure; and searching for design and procedure vulnerabilities that could be eliminated on a cost-benefit basis.

The draft safety evaluation report (DSER) (SECY-91-309) and the draft final safety evaluation report (DFSER) included a number of unresolved issues. Since both evaluations were too detailed and extensive to be repeated in this safety evaluation, the staff totally revised its evaluation as reflected in this report. As stated herein, those issues that were individually listed in the previous two evaluations have been adequately addressed by GE in its application.

The staff concludes, based on its review of the ABWR PRA, that the ABWR is of a robust design, that the design is an improvement over existing designs, and that the design meets the Commission's safety goals described in 51 FR 28044 and 51 FR 30028 published August 21, 1986, for internal events (see FSER Section 19.1.3.8.3). The Commission has determined that it is acceptable for GE to submit external event analyses that provide insights needed to identify design and procedure vulnerabilities; provide insights needed for inclusion in areas such as the reliability assurance program (RAP); and inspections, tests, analyses, and acceptance criteria (ITAAC); but do not provide core damage frequency estimates suitable for use in comparison to the Commission's safety goals or in comparison to the Electric Power Research Institute's (EPRI's) Public Safety Requirement, which states

> "The design is considered to have met the EPRI risk requirement if the mean complementary cumulative distribution function (CCDF) for whole-body dose developed for a 0.8 km (one-half mile) radius falls outside the region bounded by a lower limit for frequency at 1E-6 per year and has a lower limit for consequences of 25 rem whole-body dose at 0.8 km (one-half mile). The EPRI goal is based on consideration of both internal and external initiators"

Although direct comparison of external-event results to these goals is not possible, the ABWR design has significant margins above the design bases for seismic, fire, and internal flood-initiating events and, where computed, has low estimated core damage frequencies from these bounding analyses. The staff believes that the ABWR design meets the Commission's safety goals.

The staff finds that there is an acceptable balance of preventative and mitigative features in the ABWR design. The core damage frequency estimates for internal events reported in the ABWR PRA are on the order of 1E-7 per year. Table 19.1-1 lists the most important internal initiating events and Table 19.1-2 lists the top 20 internal event sequences leading to core damage. Station blackout (SBO) contributes about 70 percent of the internal events core damage frequency (i.e., its absolute value is about 1E-7 per year, which is low when compared to the figures in most recent boiling water reactor (BWR) PRA studies). Table 19.1-3 lists the most important sequences leading to core damage from seismic, internal flood, and fire event initiators.

Table 19.1-1 ABWR PRA initiating event contributors to CDF (Level 1, internal events)

Initiating Event	Events Per Yr.	CDF x 1E-8	Percent CDF
Station blackout < 2 hrs	1.2E-6	6.7	43%
Station blackout 2 < X < 8 hrs	4.5E-7	2.6	16%
Station blackout > 8 hrs	1.6E-8	1.7	11%
Isolation/loss of feedwater	0.18	1.7	11%
Unplanned manual reactor shutdown	1.0	1.2	7%

Table 19.1-2 Important sequences leading to core damage (internal events)

Sequence Description (Top 20 sequences)	CDF (Per Year)	Percent CDF	Plant Damage Class
SBO from 0.5 to 2 hrs, RCIC unavail. because of test or maintenance (T/M)	2.4E-8	15.6	ID
SBO more than 8 hrs	1.6E-8	10.4	IB-2
SBO from 0.5 to 2 hrs, RCIC turbine mech. failure	1.3E-8	8.6	ID
SBO from 2 to 8 hrs, RCIC unavail. because of T/M	8.9E-9	5.7	IB-1
SBO from 0.5 to 2 hrs, RCIC pump fails	8.3E-9	5.3	ID
SBO from 0.5 to 2 hrs, RCIC lubrication system fails	5.1E-9	3.3	ID
SBO from 2 to 8 hrs, RCIC turbine mech. fails	4.9E-9	3.1	IB-1
Loss of feedwater/isolation, failure to inject with feedwater, conditional containment failure (CCF) of MUX, operator fails to manually initiate feedwater after 30 min.	4.6E-9	2.9	IA
Reactor trip, failure to inject feedwater, CCF of MUX, operator fails to manually initiate feedwater after 30 min.	3.1E-9	2.0	IA
SBO from 2 to 8 hrs, RCIC pump fails	3.0E-9	1.9	IB-1
Loss of feedwater/isolation, failure to inject feedwater, CCF of system logic unit, operator fails to manually initiate feedwater after 30 min.	2.3E-9	1.5	IA
Loss of feedwater/isolation, failure to inject feedwater, operator fails to manually initiate feedwater after 30 min., CCF of remote MUX	2.3E-9	1.5	IA
SBO from 0.5 to 2 hrs, valve E51-F011 fails to close after RCIC pump has started	2.2E-9	1.4	ID
SBO from 0.5 to 2 hrs, valve F037 fails closed (NCFC)	2.1E-9	1.3	ID
SBO from 0.5 to 2 hrs, valve F004 fails closed (NCFC)	2.1E-9	1.3	ID
SBO from 0.5 to 2 hrs, valve E51-F011 fails to open when RCIC pump starts	2.1E-9	1.3	ID
Loss of feedwater/isolation, failure to inject feedwater, battery CCF, loss-of-offsite line 1 power	2.0E-9	1.3	IA
SBO from 2 to 8 hrs, RCIC lubrication fails	1.9E-9	1.2	IB-1
Turbine trip, failure to inject feedwater, CCF of MUX, operator fails to manually initiate feedwater after 30 min.	1.8E-9	1.2	IA
SBO from 0.5 to 2 hrs, isolation signal logic fails	1.7E-9	1.1	ID
Totals	1.1E-7	71.9	

Table 19.1-3 Important sequences for external events (seismic, fire, and internal floods)

Seismic Events

Sequences chosen for having low HCLPF[1] values or needing few SSC failures	Sequence HCLPF
Failure of emergency dc power	0.74 g
Emergency ac/emergency SW and fire water	0.62 g
Emergency ac/emergency SW and scram	0.62 g
Reactor or control building, containment, RPV pedestal, or RPV supports	1.11 g
ATWS and SLCs failure	0.62 g
ATWS and SLCs and high-pressure core flood	0.62 g

Internal Flood Events

Sequences chosen as being most challenging and having worst consequences	Estimated CDF
Control building: large pipe break in RSW, operator fails to isolate flooding, auto RSW pump trip fails, water flows to remaining RSW pump rooms, operator fails to respond to flooding alarm, RSW fails.	2E-9 per year
Reactor building: break in fire water standpipe or line from CST, operator does not respond to alarm to isolate flood, overfill lines to corridor are clogged, all three electrical rooms on floor B1F flood, ac power is lost to all make-up systems.	2E-10 per year
Turbine building: break in CWS system, isolation valves in CWS lines fail to close, water fills up and runs out of the condenser pit, fire door between the turbine building and the service building is either open or fails open allowing water into service building, service building floods and a door between the service building and the control building fails open or is open, water entering the control building causes electrical power supplies and all three divisions of RCW to fail.	3E-9 per year

Fire

Sequences chosen that had a core damage frequency of 1E-6 or higher before the plant design was improved	Estimated CDF (considered conservative)
Fire in the control room causes its evacuation, feedwater fails, RCIC or one high-pressure core flood train fails, either one train of low-pressure core flood train or manual depressurization fails.	less than 1E-6 per year

1 - High confidence with low probability of failure that the structure, system, or component will fail at the given peak ground acceleration.

Note: Because of the assumptions and methods used in the ABWR shutdown risk evaluation, no dominant sequences leading to core damage could be determined. NUREG-1449 did identify various important scenarios that were potential contributors to core damage in BWRs during shutdown. Conclusions from NUREG-1449 are based on actual nuclear power plant operating experience.

The internal events core damage frequency estimate is very low and is a reflection of GE's efforts to systematically minimize the effect of sequences or initiators that have been important contributors to core damage frequency in previous BWR PRAs. A brief discussion is necessary on the implications of the low estimated core damage frequency for internal events for the ABWR. Estimated core damage frequency values with absolute values less than 1 in a million years should not be taken literally as the expectation of the "true" core damage frequency of the design. Rather, this value should be taken as a reflection of the conscientious engineering and design effort to reduce or eliminate the contributors to core damage frequency found in previous PRAs. When core damage frequencies of one in a hundred thousand or a million years are estimated in a PRA, it is the areas of the PRA where modeling is least complete, supporting data are sparse, or even nonexistent that actually could be the more important contributors to risk. Areas not modeled or incompletely modeled include errors of commission, sabotage, rare initiating events, construction errors, design errors, control systems, ageing, systems interactions, human interaction with smart control rooms, and human errors.

For seismic initiating events, GE submitted a PRA-based seismic margins analysis. This method eliminates the uncertainties associated with picking an appropriate seismic hazard curve, while still providing the insights needed to judge the ability of the design to withstand beyond-design-basis earthquakes. With a PRA-based seismic margins analysis, rather than developing an estimated core damage frequency, the method estimates the margin the design has beyond the design basis safe shutdown earthquake (SSE) (which is 0.3*g* for the ABWR) and identifies any weak links in the design. GE reported that all sequences leading to core damage from a purely seismic event were found to have a high confidence with low probability of failure (HCLPF) value of 0.6*g* or higher. An HCLPF roughly represents the g-level acceleration at which a SSC is believed to fail 5 percent or less of the time with a 95 percent confidence level. The staff finds that the ABWR's HCLPF ($\geq$ 0.5*g*) demonstrates that a significant margin exists beyond the design basis earthquake level.

For internal floods that occur at power, GE performed a PRA internal flood analysis that assumed that once flood water reaches a level high enough to cause the failure of any piece of equipment in an area, then all the equipment in that area instantly fails and is unrecoverable. This analysis estimated that the core damage frequency from internal floods was on the order of 1E-8 per year. This number was particularly low because the ABWR design has three safety divisions that are physically separated. For internal floods during shutdown, GE developed guidelines for the COL applicant for configuring shutdown cooling divisions such that one division would be in operation, one isolated and in standby, and one in maintenance. It is believed that the core damage frequency from this configuration should not exceed 1E-6 per year and probably can be at least an order of magnitude lower, given the conservatism of the assumptions in the analysis.

For fires, GE submitted a fire analysis that was a combination of the Fire Induced Vulnerability Evaluation (FIVE) methodology developed by EPRI and the internal events PRA. This analysis assumed that, if a fire occurred in any portion of a fire area, all equipment in the area failed instantly. The GE fire analysis estimated the core damage frequency from fires to be on order of 1E-6 per year.

The design basis analysis for tornados in the ABWR is such that the ABWR is designed to be able to withstand tornados that occur with a frequency in the range of 1E-7 per year. Since the plant is designed to handle these low frequency tornados and is already analyzed for loss-of-offsite power events, the staff does not consider it necessary to analyze design basis tornados probabilistically.

The staff finds that the ABWR design is adequate to limit exposure to risk when the plant is operated in Modes 3, 4, and 5 (hot shutdown, cold shutdown, and refueling, respectively). The staff finds that the ABWR design includes enhanced features that reduce risk during shutdown operations when compared to operating BWRs. These features specifically address the more risk-significant operations during shutdown identified in NUREG-1449 including three independent residual heat removal (RHR) divisions, three emergency diesel generators (EDGs), an ac-independent water addition (ACIWA) system, an alternate onsite combustion turbine generator (CTG), and proper plant electrical and physical separation and layout.

The results of the Levels 2 and 3 portion of the ABWR PRA indicate that the ABWR containment is quite robust and able to accommodate severe accidents with a low attendant probability of containment failure. Both GE and staff estimates of the ABWR conditional containment failure probability (CCFP) are within the Commission's containment performance goal (0.10). Using the structural integrity definition, GE's estimate of CCFP is 0.005, whereas the staff's estimate is 0.026.

Based on the Level 3 PRA, the estimated total risk to the public for the ABWR is extremely small. GE's analysis indicates a total dose of about 0.3 person-rem over the

60-year plant life. The staff's estimate is about 1 person-rem. The staff estimated that total risk is dominated by events that lead to early containment failure and containment bypass. This is consistent with results from PRAs for operating plants.

GE made a number of design modifications to the ABWR both early in the design and later during the staff's review of the ABWR PRA that were motivated by the results of the PRA. Table 19.1-4 describes some of the modifications made to the design by GE.

Table 19.1-4 Examples of cost-effective PSA-inspired design/procedure modifications to the ABWR design

Area Modified	Modification
Core cooling systems	GE found it only needed three, not four, ECCS divisions.
Reactivity control	GE found alternate rod insertion reliability was such that less expensive ATWS mitigation system was acceptable.
Instrumentation	GE found that it could eliminate 60 percent of sensor instrumentation in the reactor safety systems without affecting plant safety.
AC-independent water addition (ACIWA) system	System added to ABWR design. Staff believes that it is the most important system for helping to prevent severe accidents.
Combustion turbine	System added to ABWR design. In combination with ACIWA, it virtually eliminates SBO as a consideration.
Lower drywell flooder (LDF)	System added to ABWR design. Floods lower drywell in event that vessel fails and corium enters lower drywell.
Containment Overpressure Protection System (COPS)	System added to ABWR design. If an accident occurs that pressurizes containment, the COPS allows for release of the pressure (90 psig set point) with the capability to reclose vent path.
Control for fourth SRV on remote shutdown panel	Extra SRV control added based on ABWR fire analysis. Mitigates control room fire.
Water-level sensors, pump trip and valve isolation circuits	Pump trips added for floods in turbine and control building as well as valve isolation signals on high water level. Pipe length between first RSW isolation valve and control building limited to help assure pipe break will not cause unacceptable results.
Containment concrete	Basaltic concrete used rather than limestone concrete to limit production of noncondensible gases from a core on floor event.

GE and the staff have drawn a substantial number of significant safety insights from the ABWR PRA that have or will affect the design, construction, operation, maintenance, and regulation of the ABWR. These insights are discussed in more detail in Section 19.1.3 in the final safety evaluation report (FSER) and in Section 19.8 in the SSAR. Appendix 19K in the SSAR lists those SSCs that are to be included in a COL action item that lists proposed inputs for the COL applicant's operational reliability assurance process (O-RAP) and design reliability assurance program (DRAP) based on the ABWR PRA. Appendix K in the FSER lists those safety insights that were motivated by the PRA. The disposition of these insights is documented in the appendix and indicates if the insights are in ITAAC, Tier 2 information, technical specifications (TS), COL action items, Interface Items, or RAP.

It is the staff's view that the mean core damage frequency for the ABWR from internal, external, and shutdown events is probably on the order of 1E-6 or less assuming the plant is constructed, maintained, and operated in accordance with the SSAR. This judgment is based on the staff's understanding of operating plant experience and PRA results, design improvements in the ABWR relative to operating reactors, and insights from the staff's review of the ABWR PRA. Furthermore, it should be emphasized that there are large uncertainties in internal event core damage frequency estimates. The external event analyses for the ABWR were quasi-probabilistic and were designed to uncover vulnerabilities in the design rather than generate specific core damage frequency estimates. Also, the shutdown risk evaluation was quasi-probabilistic, using PRA-based techniques to determine the reliability of shutdown cooling and to examine the possibility of determining plant configurations that would limit exposure to risk. The staff's review of the shutdown risk evaluation reflected the insight from previous PRAs that human error was the greatest contributor to shutdown risk. However, human error analysis methods still cannot accurately predict human response to various circumstances. This fact increases the uncertainty of the estimated bottom-line core damage frequency numbers for shutdown events.

GE conducted uncertainty analyses for the Level 1 portion of the ABWR PRA. GE reported that the internal event core damage frequency distribution had a mean value of 1.6E-7 per year and an error factor (EF) of about 4.2 (where the EF is the ratio of the 95th percentile to the median of the lognormal distribution). The 95th and 5th percentiles reported by GE were 4.5E-7 per year and 3.8E-8 per year, respectively. GE used a lognormal distribution for most random variables in the ABWR PRA, which is a mathematical simplification and assumption that is commonly used in PRA evaluations. The actual distribution for most variables is not known. Use of the lognormal distribution in lieu of the "actual" distribution adds an unquantifiable uncertainty to the evaluation, particularly in the bottom-line numbers.

GE performed importance analyses in order to determine the most important structures, systems, and components (SSCs) to be added to the RAP. GE used two importance measures: risk achievement worth ratio and fussell-vesely importance. The analysis identified scram function and its attendant equipment as very important. Station batteries are similarly important, since dc power controls the automatic depressurization system (ADS) as well as many pumps and valves. Another insight is the importance of reactor core isolation cooling (RCIC). This is because it is ac-independent, reliable, and provides high-pressure injection. These attributes are important for mitigating SBOs.

The probabilistic shutdown risk evaluation performed by GE concluded that the ABWR design could be maintained in configurations during Modes 3, 4, and 5 so that the estimated conditional core damage frequency was very low. The staff noted that estimates of core damage frequency and risk at shutdown for the ABWR design have much larger uncertainties than do such estimates for internal events in Modes 1 and 2. In the SSAR, GE presented sample plant configurations that help limit risk when shutdown occurs. The staff finds that the ABWR design, through appropriate shutdown planning, contingency planning, and operator instrumentation, can be configured and maintained in Modes 3, 4, and 5 in a manner that helps to reduce the chances of core damage or releases to the environment to a point where shutdown risk does not represent a disproportionate risk to the public.

19.1.2 Introduction

As part of the ABWR design certification application, GE submitted the ABWR PRA in response to 10 CFR 52.47, the Commission's Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants, described in FR Vol. 50, No. 153, dated August 8, 1988, p. 32138 dated August 8, 1991, and the ABWR Licensing Review Bases. The staff's assessment included the traditional evaluation of events that could lead to core damage and offsite consequences as well as an evaluation of what the ABWR PRA revealed about the ABWR design.

The general objectives of the staff's review of the ABWR PRA were (a) to identify safety insights based on the performance of systematic risk-based evaluations of the ABWR design; (b) to determine in a quantitative manner whether the ABWR design represents a reduction in risk over existing plants; (c) to examine the balance of preventive and mitigative features of the design; (d) to assess the reasonableness of the risk estimates documented in the PRA and other risk-related documents submitted as part of the FDA application package, and (e) to support pre- and postcertification activities such as ITAAC, RAP, TS, and completion of site-specific design details (e.g., ultimate heat sink). In addition, the ABWR PRA was used to both determine how the ABWR design related to various safety goals and discover design and procedural vulnerabilities.

The objectives are drawn from 10 CFR 52.47, the Commission's Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants, the Commission's Safety Goal Policy Statement, the Commission approved positions concerning the analyses of external events contained in SECY-93-087, and the Commission's interest in the use of PRA to help improve future reactor designs. In general, these objectives have been achieved by the ABWR PRA and the staff's review. The staff's proposed applicable regulation for the analysis of external events for the ABWR PRA is as follows:

> The application for design certification must contain a probabilistic risk assessment that includes an assessment of internal and external events. Simplified probabilistic methods and margins methods may be used to assess the capacity of the standard design to withstand the effects of external events such as fires and earthquakes. Seismic margin analysis must consider the effects of earthquakes with accelerations approximately one and two-thirds the acceleration of the safe-shutdown earthquake.

The staff believes that if its review were to concentrate on bottom-line numbers or merely on the quality of a PRA, the most important insights from a PRA could receive inadequate attention. In Section 19.1.3, the staff reported on its investigation of those safety insights that are revealed by the ABWR PRA about the ABWR design. These insights include those that are to be passed on to the COL applicant, insights into the balance of prevention and mitigation, design vulnerabilities, and aspects of the design that tend to reduce or exacerbate risk estimates. The results of this broadened perspective are multifold. GE's systematic evaluation of the ABWR design has accomplished the following:

- Identified important areas where minor design modifications will help confirm that the potential for severe accidents is maintained at a low level

- Helped GE to identify the most safety-important equipment to be included in the RAP

- Provided the COL applicant with information about what is most important about the ABWR design and operation from a safety standpoint

- Helped confirm for the staff that the ABWR design is robust against internal and external events.

GE provided a list of important safety insights from the ABWR PRA including internal and external events and events during all modes of operation. This list, which was developed from a systematic process, is described in detail in SSAR Section 19.9 and is detailed in Appendix K in the FSER. As indicated in this table, GE recommended and staff agreed that a subset of these insights be included in the design control document (DCD) as ITAAC, Interface Items, or Tier 1 or Tier 2 material.

During the construction stage, the COL applicant will be able to consider as-built information. The staff concludes that updated PRA insights, if properly evaluated and utilized, could strengthen programs and activities in areas such as training, development of emergency operating procedures, reliability assurance, maintenance, and 10 CFR 50.59 evaluations. The staff recommends that the design-specific PRA developed to meet 10 CFR 52.47 be revised to account for site-specific information, as-built (plant-specific) information refinements in the level of design detail, and design changes. These updates are the responsibility of the COL applicant. As plant experience data accumulate, failure rates (taken from generic data bases) and human errors assumed in the design PRA are to be updated and incorporated, as appropriate, into ORAPs.

19.1.3 Advanced Boiling Water Reactor Probabilistic Risk Assessment-Based Safety Insights

Insights gleaned from a PRA can provide significant perception into the design and operation of a nuclear power plant. This section documents the insights derived by GE and the staff about the ABWR design based on the ABWR PRA.

19.1.3.1 Technical Insights Summary

In developing the ABWR design, GE significantly reduced the dominant contributors to core damage frequency found in most current BWR plant-specific design PRAs. The success of this attempt is substantiated by the low estimated core damage frequency for internal events recorded in the ABWR PRA (1.6E-7 per year) and the staff's conclusion documented in Section 19.1.4 that, on balance, the ABWR PRA was performed in an acceptable manner. The estimated core damage frequency from internal floods is about 7E-9 per year. The fire analysis performed by GE for the ABWR produced a core damage frequency estimate of about 1E-6 per year. Because of conservatism in the analyses, the staff does not believe that the fire and internal flood core damage frequency estimates should be compared to those of internal events. The seismic analysis performed by GE was a PRA-based margins analysis that does not provide core damage frequency estimates. The margins analysis demonstrated that the ABWR design is robust for seismic events well beyond the design basis. The staff finds that the ABWR design, if built, maintained, and operated as assumed in the ABWR PRA, represents a reduction in risk when compared to the current design of BWRs.

It is important to understand what contributes to core damage frequency and risk in the ABWR design. Internal initiating events that are dominant contributors to core damage frequency include SBO, loss of offsite power, and vessel isolation or loss of feedwater. Failure of RCIC, the multiplexing transmission network, the station batteries, and the trip logic units are the most important contributing failures. SSAR Section 19K discusses the SSCs found to be most important in the PRA for internal and external events and when the plant is in modes other than full power. Table 19.1-2 provides a list of important sequences leading to core damage or risk for internal events. Table 19.1-3 provides a list for external events sequences.

For events occurring during modes other than full power, the ABWR design provides enhanced protection over the designs of many operating plants in that it has three fully separated safety divisions. To make use of this redundancy for maintenance purposes and still maintain an appropriate level of protection when in modes other than full power, GE has developed tables (See SSAR Tables 19Q.7-2 to 19Q.7-4) that list combinations of equipment that, if kept operable while in Modes 3, 4, and 5, will help ensure that a severe accident does not occur. These tables have the goal of maintaining a conditional core damage frequency of less than 1E-5 per year should the operating train of RHR cooling become unavailable. Since GE assumed that a loss of RHR cooling (i.e., of the operating train) has an occurrence rate of once per 10 years (GE's assumption is that during a 10-year period with all its included shutdowns there will be one loss of shutdown cooling event — 0.1 per year), this gives a frequency of core damage of less than 1E-6 per year when operating in Modes 3, 4, and 5. In the shutdown analysis, it is conservatively assumed that all equipment not specifically referred to in the tables as "operable" is "inoperable." During shutdown, it is conservatively assumed that any core damage would result in a large release, since the containment would be open much of the time. The staff and GE noted that separation of safety divisions is not always maintained during maintenance outages. To help assure that fires and floods cannot become common cause failures of all three divisions when in modes other than full power, GE has developed guidelines for plant operation. These guidelines would have one division isolated and in standby, another division operating in the shutdown cooling mode but not necessarily isolated, and another in maintenance. The staff believes that a safety-oriented approach to planning and controlling an outage is needed and that such an approach will reduce risk during Modes 3, 4, and 5. It recommends that COL applicants make use of the guidance provided by GE in Section 19Q of the SSAR regarding outage planning and control.

For seismic events, fires, and internal floods, the ABWR design has specific advantages over many current designs. The seismic design bases for the ABWR is a 0.3*g* SSE. In simple terms, the ABWR design can be built at any site that has its site-specific spectrum bounded by the design bases spectrum. Such a site might normally have an SSE of 0.2*g* assigned to it. However, the ABWR will be built to the 0.3*g* SSE standard, regardless. This creates an additional explicit seismic robustness at most potential sites east of the Rocky Mountains. For fires and internal floods, the existence of three separated safety divisions along with a diesel-driven fire water pump and an alternative water supply (that will remain functional following a design bases earthquake) external to the reactor building provide design improvements that significantly reduce potential core damage.

External events such as external flooding and hurricanes may be analyzed by a designer in a bounding manner in order to minimize the chances that the design has vulnerabilities to site-specific external events. GE did not chose to evaluate such external events at the Design Certification stage. The staff worked with GE to assure that GE was aware of the potential for vulnerabilities to such external events. It is possible that some sites may not be appropriate for the ABWR design because they could introduce vulnerabilities into the ABWR design that were not taken into account in the Design Certification process. For each site, the COL applicant must provide a site-specific PRA-based analysis to help determine if the ABWR design has any vulnerabilities to previously unanalyzed external events applicable to the site (e.g., river flooding or soil liquefaction from seismic events).

Human reliability analyses (HRAs) of the ABWR design show that it is not particularly sensitive to operator errors during operation in Modes 1 and 2. This is true, in part, because of the multitude of paths by which water can be provided to the core and the inherent attributes of the design (in most sequences of actions, it takes a long time before an operator needs to make a critical decision, if such a decision is needed at all). GE conducted a sensitivity analysis to investigate the influence of variations in the PRA-modeled human error rates of the ABWR core damage frequency estimate. The analysis suggests that, while there is little room for reducing core damage frequency by improving human performance, core damage frequency can increase if human performance significantly degrades. The potential increase in core damage frequency appears to be about two orders of magnitude above the base case frequency under an assumption that all human error probabilities are simultaneously increased by a factor of 30. However, such a magnitude of increase is unrealistic because the human error probabilities used in the ABWR human performance analyses were either screening values or reasonable values for the actions defined based on previous HRAs for existing PRAs. Even though some recovery actions (such as offsite power and emergency ac) were not included in this analysis, the results of the sensitivity analysis provide a proper indication of the ABWR tolerance of human errors.

Since the ABWR design was not detailed in a number of areas important to evaluating human actions and potential errors (e.g., control room design or plant-specific emergency procedures), GE's HRA was essentially a scoping analysis, based largely on a generalization of results from previous HRAs (which reflect conventional BWR human-machine interface designs) and the use of screening-type human error probabilities collected from various sources. GE contends that this treatment is conservative for the ABWR because of the significant improvements envisioned for the ABWR human-machine interface design relative to earlier designs. However, the validity of the scoping analysis for the as-built ABWR design will need to be confirmed as part of the implementation of the detailed control room design process. This process and associated ITAAC/DAC are described in Chapter 18 of this report. The focus of that effort with regard to the HRA will be on confirming that the final control room design has not introduced any human engineering deficiencies that would 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.

Every nuclear power plant PRA is incomplete to some extent. Ordinarily, a PRA is performed on a plant for which the site is known, the equipment has been procured, and the plant is nearly or completely built. For the ABWR PRA, the PRA practitioners had to develop their models using a design that lacks many of the details available for an existing plant. The lack of detail was recognized by the staff as well as by GE. For this reason, the staff developed several processes to which the COL applicant must comply, including (a) the ITAAC process, (b) the reliability assurance process and program (O-RAP and DRAP) that the COL applicant should implement to confirm that the as-built plant conforms to the assumptions of the ABWR PRA, and (c) interface items that the COL applicant must address in its application for a COL. To help confirm that the assumptions in the PRA are realized in the as-built design, GE provided a systematic list of SSCs that are to be included in the COL applicant's RAP, PRA-based insights into the ITAAC, and a systematic list of safety insights derived from the ABWR PRA that will be passed on to future COL applicants. One of the significant benefits of having a PRA at early stages in the design is that GE took advantage of PRA insights to improve the final design and provide guidance to COL applicants or holders.

Based on the information provided in the SSAR, this design has achieved a significant reduction in expected core damage frequency and risk compared to operating plant PRA results. The staff compared the numerical results of internal events in the ABWR PRA to the Commission's safety goals and found that the ABWR design meets each safety goal.

As part of its investigation of the ABWR design, the staff endeavored to determine if a balance had been achieved between the prevention of accidents and accident mitigation capabilities. The staff concludes that the design has an appropriate balance of prevention and mitigation. Details of this discussion are provided in Section 19.1.3.9.2 of this report.

GE searched for design and procedure improvements that were prudent to include in the ABWR. The staff concludes that the search made by GE was adequate to identify design and procedure vulnerabilities for the ABWR design. Details of the design improvements motivated by the ABWR PRA are discussed in Section 19.1.3.2.3 of this report.

The staff believes that the ABWR PRA is capable of supporting pre- and post- certification activities such as ITAAC, RAP, and TS. The PRA can be modified to include site-specific design details for areas outside the design certification such as the ultimate heat sink.

19.1.3.2 Level 1 — Internal Events

19.1.3.2.1 Dominant Accident Sequences

GE estimated the total core damage frequency from internal events for the ABWR design to be 1.6E-7 per year. The internal events initiators that contribute the most to core damage frequency are loss of offsite power, SBO, and loss of feedwater/isolation of the vessel (See

Table 19.1-1). Of these, SBO is the largest contributor to core damage frequency. This is consistent with risk profile estimates for many other BWR PRAs that have identified SBO as one of, if not the leading, contributor to core damage frequency. For the ABWR, SBO and loss of offsite power sequences have an estimated core damage frequency of about 1E-7 per year.

The top seven sequences that in aggregate contribute 52 percent of the internal event core damage frequency are SBO events in which RCIC fails or is unavailable or where the blackout outlasts the capacity of the emergency batteries. Of the top 20 sequences, 14 involve SBO. The reason SBO events in the ABWR have a very low estimated absolute core damage frequency value is the low overall core damage frequency estimate for the ABWR design. This is primarily because the ABWR design has three independent EDGs and because of the addition of the onsite CTG. With respect to SBO events, the staff noted that the removal of a steam-driven high-pressure system (such as the high- pressure core injection system (HPCI) in earlier designs) is well compensated for by a substantial improvement in the reliability of backup power resulting from the addition of a CTG in the ABWR design.

Anticipated transients without scram (ATWS) has a very low absolute value for the ABWR design for several reasons, including the following: (1) the ABWR design has diverse means of inserting control rods into the core (by hydraulic or electric means), (2) initiation of the standby liquid control system (SLCS) is automated, (3) the hydraulic system that inserts control rods includes additional backup scram valves to relieve scram air header pressure, and (4) the ABWR design does not have a scram discharge volume.

19.1.3.2.2 PRA as a Design Tool

The ABWR design was influenced by PRA insights. Table 19.1-4 summarizes design changes that GE made to the ABWR that were motivated by PRA insights. GE used PRA insights and mini-PRA studies to help decide on a number of important design options. The performance of the ABWR PRA and several mini-PRAs early in the ABWR design process made important contributions to improving the ABWR design. The PRA influenced the design positively, not only early in the design process, but also during the design certification.

GE indicated that the use of PRA evaluations in the early stages of the ABWR design helped GE to simplify the design in a manner that maintains or improves core damage frequency estimates compared to estimates for operating plants. Examples of the design simplification by GE include the following:

- The reduction in the number of emergency core cooling system (ECCS) divisions from four to three by upgrading the RCIC system and by modifying the automatic depressurization system (ADS) logic to begin operation when a low water level is reached (improved design for transient response)
- The elimination of 60 percent of the sensor instrumentation in the reactor safety systems without its affecting plant safety.

In its initial ABWR PRA submittal, GE concluded that the Commission's safety goals and goals proposed by EPRI for evolutionary reactors (core damage frequency, CCFP, and dose at the boundary of the plant following an accident) were met and that the design had many means of severe accident prevention and mitigation. After discussions with the staff and in order to come to more complete agreement with EPRI evolutionary plant design guidelines, GE decided to make several design improvements that significantly added to defense in depth and reduced uncertainty that the design would meet its intended goals. GE made the following design changes:

- Added an AC-independent water addition (ACIWA) system. This system provided benefits for SBO, fires, internal floods, and seismic events. It has both preventative and mitigative capabilities in that it can either inject to the vessel or spray the drywell. It can provide water from a seismically robust diesel-driven pump or a fire truck.
- Added a non-safety grade combustion turbine generator that starts automatically with safety grade loads added manually. This system used in conjunction with the ACIWA system significantly reduces the estimated core damage frequency from SBO.
- Made recommendations for the improvement of emergency procedures following examination of the dominant severe accident sequences.
- Changed the lower drywell basemat concrete composition from limestone to basaltic concrete to limit production of noncondensible gases.
- Added a containment overpressure protection system (COPS).
- Increased the pressure capacity of the drywell head.
- Surveillance testing of the microprocessor-based controllers was increased to quarterly to improve the ability to detect failures left undetected by the continuous self-test feature. This action was taken

based on a study of potential failures of the safety system logic and control.

19.1.3.2.3 Plant Features and Operator Actions Important to Risk

Insights as to what features, procedures, or operator actions are important to risk have been gathered from several areas of Chapter 19 of the SSAR, including Appendices 19K, 19L, 19M, 19Q, and 19R as well as from discussions with GE. The staff considers the following plant features, procedures, and operator actions to be the most important to risk reduction or prevention for Level 1 internal events.

Plant Features Important to Risk

Combustion Gas Turbine Generator — The combustion gas turbine generator (CTG) provides non-safety grade power that is diverse and independent from the normal and emergency ac power sources. It is capable of providing ac power to any of the three safety divisions or to a condensate pump. The CTG requires no plant support systems to start or run. The gas turbine, in conjunction with the ACIWA system, is a significant factor in reducing the likelihood of an SBO (a leading contributor to core damage).

ACIWA System — The ACIWA system has two pumping sources available: a diesel-driven pump and a fire truck. The diesel-driven pump is designed to survive a design bases earthquake. Its pumping sources are located outside the reactor building. Manual valves to direct the flow from the ACIWA system into the RHR system and then on to either the core or to the drywell or wetwell spray are located in the reactor building and can be operated successfully following an internal event.

Lack of Recirculation Piping — The ABWR design has done away with external reactor recirculation piping and pumps. This means that there are no large pipes that penetrate the vessel below the core. This design detail has significantly reduced the already low chances of a loss-of-coolant accident (LOCA) leading to core damage.

Three ECCS Trains — The redundancy of the divisions improves the chances of the design preventing transients from developing into core damage events.

Reactor Core Isolation Cooling (RCIC) — RCIC is a safety-grade system that provides the ABWR with a diverse high-pressure system that can delay the onset of core damage following an SBO event to about 8 hours. Importance analyses from the ABWR PRA indicate that RCIC is one of the most important systems in preventing core damage accidents.

Multiplexing System — The high reliability of the multiplexing system is very important in the ABWR PRA. Significant degradation of this function could result in a large increase in the likelihood of the plant's having an event that leads to core damage.

High-pressure Core Flooder (HPCF) Pump — One of the high-pressure core flood pumps can be operated independently of the essential multiplexing system. This design feature is an important factor in reducing the chances of the plant experiencing core damage, since this design should reduce the chance of a common cause control system failure, that would disable all ECCS pumps.

Electrically Driven Control Rod Insertion — The diverse ability to drive in rods with electrical motive power as well as hydraulic power significantly reduces the chances of an ATWS occurring.

Automatic Depressurization System — The ADS is needed to enable the use of low-pressure core flooder pumps and the ACIWA system. ADS is important in SBO, small LOCAs, and transients.

Reactor Water Cleanup (RWCU) Isolation Valves — The isolation valves in the RWCU system must be capable of isolating against a differential pressure equal to the operating pressure of the reactor coolant system (RCS) in the event that there is a LOCA in the RWCU.

Operator Actions Important to Risk

- Five specific human actions have been identified as most important to the Level 1 PRA analysis because of their impact on core damage frequency. All five of these relate to makeup of reactor inventory — the first four with the reactor at high pressure and the last with the reactor at low pressure (the acronyms in parentheses correspond to the basic event identifier in the ABWR PRA):
  - Backup manual initiation of HPCF (HOOBOPHL)
  - Recovery of feedwater in events without MSIV closure (Q)
  - Recovery of feedwater in events with MSIV closure (Q2)
  - Reopening of HPCF injection valves following maintenance (HBMAER1)

- Use of condensate injection following scram with reactor depressurized (COND).

A sixth human action — control of reactor water level in an ATWS (LPL) — was also identified as critical to the Level 1 analysis based on its impact on offsite consequences rather than on its impact on core damage frequency. Because of their importance, these human actions are included as "critical tasks" that will be evaluated further by the COL applicant as part of the advanced plant design (see SSAR Section 19.9).

- Although miscalibration of sensors is not a significant contributor to core damage frequency (partly because of the low assigned probabilities), instrument calibration is an important maintenance action requiring that a special maintenance procedure be developed by the COL applicant.

- A limited number of human actions were identified as having a significant impact on containment performance and the results of the Level 2 analysis. These involve the following:

  - Emergency depressurization of the reactor to mitigate core damage in-vessel and reduce the potential for containment failure from direct containment heating (OP)

  - Alignment and initiation of firewater (ACIWA mode of RHR) for injection into the depressurized reactor pressure vessel (ARV)

  - Alignment and initiation of firewater for injection to the drywell sprays to prevent drywell overtemperature (HTF)

  - Initiation of drywell sprays in response to suppression pool bypass

  - Use of a fire truck as a backup to the diesel-driven firewater pump.

Because of their importance, the actions are included in COL Action Items (See SSAR Section 19.9). Some of these actions would also be addressed by the COL applicant as part of its Accident Management Program discussed in Section 19.2.

Appendix 19K in the FSER lists those PRA-based safety insights motivated by the PRA. The disposition of these insights is documented in the table and indicates if the insights are in ITAAC, Tier 2 information, TS, COL action items, Interface Items, or the RAP.

19.1.3.2.4 Interface and COL Applicant Action Items

The Design Certification Rule 10 CFR Part 52 does not require a vendor to provide complete design details, especially in areas where there is evolving technology (e.g., the control room) or where it is very difficult to bound the possible design options (e.g., the ultimate heat sink and the service water pump house). Other areas such as accident management are the responsibility of the COL applicant. GE identified Interface Items (hardware) and COL Action Items (primarily procedural or analytical) that define the areas derived from performance of the ABWR PRA that the COL applicant needs to perform or complete. The following COL action item addressed in Section 19.9 in the SSAR is of particular interest to the staff:

- The COL applicant should update the design-specific ABWR PRA to include site-specific information (i.e., a site-specific PRA) and additional design details (e.g., once the COL applicant designs the structures, systems, and components that were not part of the design certification). The PRA is to be maintained by the COL holder (living PRA) so that the PRA can be useful in helping to make 50.59-like decisions as well as helping to determine the safety significance of operational events or data from ABWR operation or other nuclear power plants.

These actions are detailed in Section 19.9 and Table 1.9.1 in the SSAR. The staff finds the Interface Items in Section 1.9 in the SSAR to be adequate for performing the ABWR design-specific PRA. The staff finds the COL action items in Section 19.9 of the SSAR to be acceptable and finds that GE identified those actions that the COL applicant must take in order to help make the PRA assumptions come true in the as-built, as-operated plant.

19.1.3.2.5 Insights From Uncertainty, Importance, and Sensitivity Analyses

GE has conducted a traditional uncertainty analyses for the Level 1 portion of the ABWR PRA. GE calculated importance analyses in order to determine the most important SSCs to be added to the RAP. GE also conducted selected sensitivity analyses to determine the robustness of the design to biases in numerical assumptions.

The uncertainty analyses started with GE's assuming that all basic events (in the fault trees) have lognormal distributions. These distributions were propagated using Monte Carlo techniques. As shown in Figure 19.1-1, the core damage frequency distribution had a mean value of 1.6E-7 per year and an error factor (EF) of about 4.2

(where the EF is the ratio of the 95th percentile to the median of the lognormal distribution). The 95th and 5th percentiles reported by GE were 4.5E-7 per year and 3.8E-8 per year, respectively.

The uncertainty studies revealed that the majority of basic events that were part of the dominant core damage sequences (i.e., involved in the failure of SSCs needed to prevent core damage) were also identified by the importance analyses as the top contributors to core damage frequency. Similarly, the relative importance of the top-ranked sequences and top-ranked basic events was shown to change only slightly when the input data were biased by a factor of two.

GE performed importance analyses on internal events using "change in core damage" as the underlying variable. GE used the importance analyses results to help determine SSCs to be included in Appendix 19K of the SSAR that deals with SSCs that are to be included in the COL applicant's RAPs. GE used two importance measures: risk achievement worth ratio and fussell-vesely importance. The risk achievement worth ratio is calculated by taking the ratio of (a) the core damage frequency with the particular SSC always failed to (b) the base core damage frequency. Risk achievement identifies those SSCs for which it is particularly important to do good maintenance, since poor reliability or availability of this equipment would increase estimated core damage frequency significantly. The Fussell-Vesely importance measure is calculated by (a) finding the difference between the base case core damage frequency and the core damage frequency with the SSCs operating perfectly, and (b) dividing by the base case core damage frequency. The Fussell-Vesely importance measure identifies which SSCs would benefit the most from improved testing and maintenance that would minimize equipment unavailability and failures. The staff believes that the risk achievement worth is a particularly important measure when deciding which SSCs to include in a RAP.

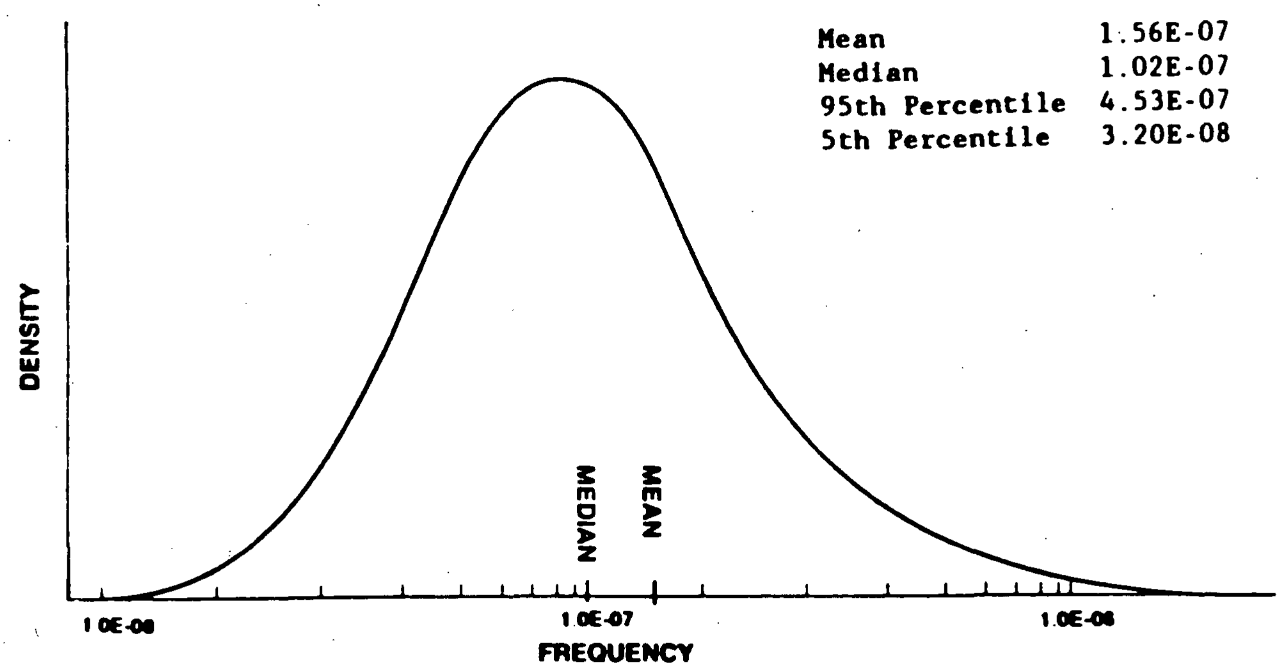


Figure 19.1-1 ABWR core damage frequency distribution

Table 19.1-5 in this report provides a listing of the most important SSCs and their importance. The table highlights the importance of the scram function and its attendant equipment. Similarly important are the station batteries, since dc power controls ADS as well as many pumps and valves. The table also points out the importance of the CTG and RCIC. This is because SBO is the largest contributor to internal event core damage frequency estimates.

Table 19.1-6 in this report shows the sensitivity of core damage frequency to test and maintenance outage assumptions for ECCS equipment. Increasing RCIC unavailability was found to cause the greatest increase in estimated core damage frequency. This results in large part from RCIC's contribution to mitigation of SBO sequences. This outcome supports the results of importance analyses discussed above. GE found that increasing high-pressure core flooder B (HPCFB) unavailability resulted in the second highest core damage frequency increase. This is because HPCFB is important in mitigating a common-cause failure of the essential multiplexing system.

GE performed a sensitivity study on the effect of increasing the test and maintenance outage times for ECCS equipment (i.e., the period each piece of ECCS equipment is assumed to be unavailable during the year because of testing or maintenance). RCIC was found to be the ECCS system most sensitive to increased outage time. This would be expected based on the Fussell-Vesely importance measure results. Second in importance is high-pressure core flooder "B" that includes a diverse (resulting from the multiplexing system), hard-wired manual-initiation backup in the control room. The sensitivity study indicated that other individual systems are not sensitive to test and maintenance outage time. GE noted that even with RCIC out of service all the time, the ABWR core damage frequency estimate is still below 1E-5 per year. It should be noted that the CTG and the EDGs have the largest Fussell-Vesely importance measure value, since 100-percent availability of the CTG or an EDG would eliminate SBO as an estimated core damage contributor.

19.1.3.3 Level 1 — External Events

Three external events were analyzed in the ABWR PRA: seismic, fire, and internal flood. In many PRAs performed to date, these external events have had combined core damage frequencies that are the same magnitude as internal events. It is not unusual to see the combined core damage frequencies for these events in the area of 1E-4 per year. The varying methods used in the ABWR PRA to evaluate external events are acceptable to the staff because they provide the insights necessary to determine if any design or procedural vulnerabilities exist for these external events and because the methods provide insights needed for the RAP, ITAAC, TS, and other important programs.

To help confirm that no vulnerabilities are introduced when a site is chosen for an ABWR (e.g., site-specific external floods can flood higher than assumed in the design basis so a vulnerability, such as an external, nonwatertight door, could exist), the following steps need to be taken. The staff requires COL applicants to perform a site-specific PRA-based analysis that searches for vulnerabilities from site-specific external events not evaluated in the ABWR PRA and that are not enveloped by the ABWR PRA assumptions (e.g., seismic-induced soil liquefaction). This PRA-based analyses must also search for vulnerabilities in the parts of the ABWR design that were not part of the design certification (e.g., the service water pump house).

The ACIWA system is very important to preventing and mitigating severe accident external events. GE and the staff understand the importance of the ACIWA system being able to function following various external initiators. For this reason, GE developed a COL action item (detailed in Section 19.9 and Table 1.9 in the SSAR) that outlines the following:

- The COL applicant should design the building to house the ac-independent water addition (ACIWA) pumps in such a manner that the building is capable of withstanding high seismic events, river flooding, and other site-specific external events such as high winds (e.g., hurricanes). The capability of the building housing the ACIWA pumps should be evaluated in the site-specific PRA to assure that vulnerabilities do not exist for the specific site.

19.1.3.3.1 Seismic Events

A PRA-based margins analysis systematically evaluates the capability of SSCs in the design to withstand an earthquake, but does not estimate the core damage frequency from seismic events. The margins analysis is a way of estimating how much larger an earthquake than the SSE the design should be able to withstand without sustaining core damage.

Table 19.1-5 SSCs identified by importance measure values

Structure, System, or Component (SSC)	Risk Achievement Worth	Fussell-Vesely Importance Measure (%)
Multiplex Transmission Network (CCF)	204,400	12.1
Trip Logic Units	204,300	6.0
Remote Multiplexing Units	204,300	6.0
Station Batteries	13,160	3.3
Digital Trip Modules (CCF)	281	0.1
Level 2 Sensors (CCF)	273	0.1
SRV (CCF)	189	0.1
RHR Flow Transmitters (CCF Miscalibration	32	0.2
Level 8 Sensors (CCF Miscalibration)	28	0.1
Combustion Turbine Generator (CTG)	14	69.6
Both Offsite Power Sources	14	1.3
Division 1 Transmission Network Failure (EMS)	13	0.70
1st ESF RMU Division 1 Fails	13	0.34
2nd ESF RMU Division 1 Fails	13	0.34
RCIC Turbine	12	12.0
RCIC Pump	12	7.4
RCIC System (Unavailable, Test or Maintenance)	12	21.8
RCIC Flow Sensor E51-FT007-2 Fails	12	0.32
RCIC Isolation Valve F036 Fails (NOFC)	12	0.18
RCIC Isolation Valve F035 Fails (NOFC)	12	0.18
RCIC Isolation Valve F039 Fails (NOFC)	12	0.18
RCIC Check Valve E51-F003 Fails to Open	12	0.15
RCIC Check Valve F038 Fails to Open	12	0.15
RCIC Outboard Check Valve F005 Fails to Open	12	0.15
NBS Isolation Check Valve B21-F003B (FW Isolation) Fails Closed	12	0.15
NBS Isolation Check Valve B21-F004B (FW Isolation) Fails Closed	12	0.15
NBS Manual Valve B21-F005B (FW Isolation) Fails Closed (NOFC)	12	0.14
RCIC Pres Sensor PIS-Z605 Miscalibrated	12	0.054
RCIC Flow Sensor FT-007-2 Miscalibrated	12	0.054
RCIC Pressure Sensor E51-PIS-Z605 Fails	12	0.013
Failure of Division I Distribution Panel	12	0.0064
SLU/EMS Link for Division I SLU 1 Fails (RCIC Fails)	12	0.00046
SLU/EMS Link for Division I SLU 2 Fails (RCIC Fails)	12	0.00046
RCIC Turbine Exhaust Isolation Valve F039 Limit Switch Fails	12	0.74
RCIC Steam Supply Bypass Valve F045 Limit Switch Fails	12	0.74
RCIC Turbine Lubrication System	12	4.6
RCIC Min. Flow Bypass Valve E51-F011 (NOFO)	12	2.0
RCIC Min. Flow Bypass Valve E51-F011 (NCFC)	12	1.9
RCIC Injection Valve E51-F004 (NCFC)	12	1.9
RCIC Steam Supply Valve E51-F037 (NCFC)	12	1.9
RCIC Isolation Signal Logic	12	1.5
All 3 Diesel Generators, CCF	11	unknown
SP Temp High (Loss of Pump Head)	6.6	0.00055
SRVs	4.3	1.0
Single Offsite Power Line	4.1	3.1
HPCF Maintenance Valve E22-F005B	2.7	1.7
HPCF Pump	2.6	1.1

Table 19.1-6 Sensitivity to test or maintenance (T/M) outages

Single system perturbations (factor of five) to base case T/M of two percent unavailability

System							
RCIC	0.02	**0.1***	0.02	0.02	0.02	0.02	0.02
HPCFB	0.02	0.02	**0.1**	0.02	0.02	0.02	0.02
HPCFC	0.02	0.02	0.02	**0.1**	0.02	0.02	0.02
RJRA	0.02	0.02	0.02	0.02	**0.1**	0.02	0.02
RHRB	0.02	0.02	0.02	0.02	0.02	**0.1**	0.02
RHRC	0.02	0.02	0.02	0.02	0.02	0.02	**0.1**
Core Damage Frequency	1.6E-7	2.9E-7	1.8E-7	1.6E-7	1.6E-7	1.6E-7	1.6E-7
Percent Increase	N.A.	86	13	< 1	< 1	< 1	< 1

* Unavailabilities in bold type represent perturbations to the base case in the first column.

Effect of single systems completely removed from service

System							
RCIC	0.02	**1.0***	0.02	0.02	0.02	0.02	0.02
HCPFB	0.02	0.02	**1.0**	0.02	0.02	0.02	0.02
HPCFC	0.02	0.02	0.02	**1.0**	0.02	0.02	0.02
RHRA	0.02	0.02	0.02	0.02	**1.0**	0.02	0.02
RHRB	0.02	0.02	0.02	0.02	0.02	**1.0**	0.02
RHRC	0.02	0.02	0.02	0.02	0.02	0.02	**1.0**
Core Damage Frequency	1.6E-7	1.8E-6	4.1E-7	1.6E-7	1.6E-7	1.6E-7	1.6E-7
Percent Increase	N/A	1073	162	4	5	2	< 1

* Unavailabilities in bold type represent single systems that have been removed from service.

Table 19.1-6 Sensitivity to test or maintenance (T/M) outages (continued)

Multiple system perturbations (factor of five) to base case

System							
RCIC	**0.1***	0.02	0.02	**0.1**	0.02	**0.1**	**0.1**
HPCFB	0.02	**0.1**	0.02	**0.1**	**0.1**	**0.1**	**0.1**
HPCFC	0.02	0.02	**0.1**	0.02	**0.1**	**0.1**	**0.1**
RHRA	**0.1**	0.02	0.02	0.02	0.02	0.02	**0.1**
RHRB	0.02	**0.1**	0.02	0.02	0.02	0.02	**0.1**
RHRC	0.02	0.02	**0.1**	0.02	0.02	0.02	**0.1**
Core Damage Frequency	2.9E-7	1.8E-7	1.6E-7	3.1E-7	1.8E-7	3.2E-7	3.2E-7
Percent Increase	88	14	< 1	101	13	102	104

* Unavailabilities in bold type represent perturbations to a base case value of 0.02.

The capability of a particular SSC to withstand beyond design bases earthquakes is measured in terms of an HCLPF given a certain input acceleration. The HCLPF has units of acceleration. An HCLPF value represents the acceleration that approximates the concept of having about 95 percent confidence that the SSC will fail less than about 5 percent of the time. The ABWR is designed to withstand a 0.3*g* SSE. Since the analysis used in evaluating the capability of SSCs to withstand the SSE has significant margin in it, the staff expects that a plant built to withstand a 0.3*g* SSE actually will be able to withstand a much larger earthquake. The staff indicated that it expects that a plant truly designed to withstand a 0.3*g* SSE should have a plant HCLPF at least 1.67 times the SSE. In SSAR Section 19H.5, GE provided a COL action item to assess how the as-built facility corresponds to the assumptions in the seismic margins analysis. As discussed below, the ABWR design as analyzed meets the 1.67 times the SSE expectation. Thus, GE and the staff believe that the ABWR can be designed and built in a reasonable, cost-effective manner to achieve a plant HCLPF of 0.5*g* or higher. The staff notes that the ABWR seismic margins analysis assumes that no soil liquefaction will occur (regardless of the g-level) or that liquefaction would not affect the plant HCLPF. COL applicants will need to confirm (with their site-specific seismic risk assessment) that soil liquefaction does not introduce any vulnerabilities. The staff finds this acceptable.

19.1.3.3.1.1 Dominant Accident Sequences

In the PRA-based margins method, the event trees and fault trees for internal events are modified to accommodate seismic events. In this way, the random failures and human errors modeled in the internal events portion of the PRA are captured in the seismic analysis.

In Table 19.1-7 in this report, 25 sequences are identified that lead to core damage in the ABWR PRA-based seismic margins analysis, listed by plant damage class. The first HCLPF value given for each sequence is the HCLPF assuming that only seismic failures can occur (i.e., without any random failures or human errors). Additional HCLPF entries under a sequence include various combinations of random failures and human errors. The underlying assumption that earthquakes exceeding the SSE will happen less frequently than once in a 1,000 years allows us to exclude random failures or human errors less than 1E-3. This is because the combination of seismic events more than 1.67 times the SSE with random failures lower than 1E-3 would result in core damage frequency estimates much less than 1E-6 per year. We also exclude seismic-random combinations where the seismic portion has an HCLPF at least as high as the seismic-only HCLPF.

Further details are provided in Table 19.1-8 in this report, which provides the "dominant" cut sets for each plant damage state. The word "dominant" appears in quotes to emphasize that the use of this terminology in the context of a margins study should not be taken in the same way as it would be for a conventional PRA. While these sequences and cut sets dominate the HCLPF values for the plant, the margins approach does not permit a determination that these are the dominant contributors to seismic risk in a probabilistic sense.

Table 19.1-7 Results of GE HCLPF quantification using the MIN/MAX approach (by plant damage class)

Damage Class	Sequence Number	Sequence Description	Seismic HCLPF	Seismic/Random HCLPF
IA	5	/SI*LOP*APW*/DP*/C*/UR*X*/HX	0.74 g	None
	9	/SI*LOP*APW*/DP*/C*UR*X*/HX	0.74 g	None
	13	/SI*LOP*AW*DP*/HX	0.74 g	None
	18	/SI*LOP*/APW*/C*/PC*UR*UH*X	0.74 g	None
	20	/SI*LOP*/APW*/C*PC*UH*X	0.74 g	None
	Total		0.74 g	None
IB-2	3	/SI*LOP*APW*/DP*/C*/UR*/X*FA*/HX	0.62 g	None
	Total		0.62 g	None
IC	21	/SI*LOP*/APW*C*/C4*/LPL*/PC*/PA*UH*UR	0.70 g	0.62g * 6E-2
	23	/SI*LOP*/APW*C*/C4*/LPL*PC*/PA*UH	0.74 g	0.62g * 1E-1
	27	/SI*LOP*/APW*C*C4*UH	0.62 g	None
	Total		0.62 g	None
ID	7	/SI*LOP*APW*/DP*/C*UR*/X*FA*/HX	0.70 g	0.62g * 6E-2
	17	/SI*LOP*/APW*/C*/PC*UR*UH*/X*V1*V2	0.70 g	0.62g * 6E-2
	19	/SI*LOP*/APW*/C*PC*UH*/X*V1*V2	0.74 g	0.62g * 2E-3
	Total		0.70 g	0.62g * 6E-2
IE	15	SI*/HX	1.11 g	None
	Total		1.11 g	None
IV (incl. IV-P)	11	/SI*LOP*APW*/DP*C*/HX	0.62 g	None
	12	/SI*LOP*APW*/DP*C*HX	0.70 g	None
	22	/SI*LOP*/APW*C*/C4*/LPL*/PC*PA	0.74 g	0.62g * 2.4E-3
	24	/SI*LOP*/APW*C*/C4*/LPL*PC*PA	0.74 g	None
	25	/SI*LOP*/APW*C*/C4*LPL	0.74 g	0.62g * 1E-2
	26	/SI*LOP*/APW*C*C4*/UH	0.62 g	None
	Total		0.62 g	None

Table 19.1-7 Results of GE HCLPF quantification using the MIN/MAX approach (by plant damage class) (continued)

Damage Class	Sequence Number	Sequence Description	Seismic HCLPF	Seismic/Random HCLPF
IA-P through IE-P	4 (IB2-P)	/SI*LOP*/APW*DP*/C*/UR*/X*FA*HX	0.70 g	None
	6 (IA-P)	/SI*LOP*APW*/DP*/C*/UR*X*HX	0.74 g	None
	8 (ID-P)	/SI*LOP*APW*/DP*/C*UR*/X*FA*HX	0.70 g	None
	10 (IA-P)	/SI*LOP*APW*/DP*/C*UR*X*HX	0.74 g	None
	14 (IA-P)	/SI*LOP*AW*DP*HX	0.74 g	None
	16 (IE-P)	SI*HX	1.11 g	None
	Total		0.70 g	None

NOTE: An entry of "none" in the seismic/random HCLPF column means that either (1) there were no combinations in which the random portion was greater than 1E-3 or (2) there were no combinations in which the HCLPF of the seismic portion of the combination was less than the HCLPF from seismic only.

LEGEND:

APW = Failure of Emergency AC Power or Service Water
C = Failure of Reactivity Control System
C4 = Failure of Standby Liquid Control System
DP = Failure of dc Power
FA = Failure of Fire Water System
HX = Rupture of RHR Heat Exchanger
LOP = Loss of Offsite Power
LPL = Failure of Primary Level and Pressure Control
PA = Failure to Inhibit ADS Actuation
PC = Failure of SRVs to Close
SI = Collapse of Plant Essential Structures
UH = Failure of High Pressure Core Flooder
UR = Failure of Reactor Core Isolation Cooling System
V1 = Failure of Low Pressure Core Flooder
V2 = Failure of Condensate Injection
X = Failure of Primary Depressurization

A slash (/) appearing before a designator means the occurrence of the opposite condition (i.e., success rather than failure).

Table 19.1-8 "Dominant" contributors to HCLPF using the MIN/MAX approach (by plant damage class)

<table>
<tr><th>DAMAGE CLASS</th><th colspan="4">SEISMIC ONLY "DOMINANT CUT SETS"</th><th>SEISMIC/RANDOM "DOMINANT CUT SETS"</th></tr>
<tr><td rowspan="2">IA</td><td colspan="4">Dc Cable Trays (0.74 g)</td><td rowspan="2">None</td></tr>
<tr><td colspan="2">SW HVAC (0.63 g)
or
SW Pump (0.62 g)
or
SW Pump House (0.60 g)
or
Diesel Gen. (0.62 g)
or
Transformer (0.62 g)
or
MCCs (0.62 g)
or
Switch (0.63 g)
or
HPCF Pump (0.62 g)</td><td colspan="2">S/Rvs (0.74 g)</td></tr>
<tr><td>IB-2</td><td colspan="2">SW HVAC (0.63 g)
or
SW Pump (0.62 g)
or
Diesel Gen. (0.62 g)
or
Transformer (0.62 g)
or
MCCs (0.62 g)
or
Switch (0.63 g)</td><td colspan="2">Fire Pump (0.62 g)</td><td>None</td></tr>
<tr><td>IC</td><td>Fuel Ass.
(0.62 g)
or
HCU
(0.6 3 g)</td><td>SLC Tank
(0.62 g)
or
SLC
Pump
(0.62 g)</td><td>HPCF
Pump
(0.62 g)</td><td></td><td>None</td></tr>
</table>

How to read this table:

The columns for seismic only and seismic random are independent. Within each of those columns, the or function is obvious, and the and function is represented by a horizontal dotted line. Thus, for Damage Class IA, the seismic only column represents the Boolean expression [(SW HVAC + SW Pump + Diesel Gen. + Transformer + MCCs + Switch + HPCF Pump) * S/Rvs] and the same column for Damage Class IC represents the Boolean expression [Fuel Ass. * (SLC Tank + SLC Pump) * HPCF Pump].

Table 19.1-8 "Dominant" contributors to HCLPF using the MIN/MAX approach (by plant damage class) (continued)

<table>
<tr><th>DAMAGE CLASS</th><th colspan="4">SEISMIC ONLY "DOMINANT CUT SETS"</th><th colspan="3">SEISMIC/RANDOM "DOMINANT CUT SETS"</th></tr>
<tr><td rowspan="2">ID</td><td colspan="2">SW HVAC (0.63 g)
or
SW Pump (0.62 g)
or
SW Pump House (0.60 g)
or
Diesel Gen. (0.62 g)
or
Transformer (0.62 g)
or
MCCs (0.62 g)
or
Switch (0.63 g)
or
HPCF Pump (0.62 g)</td><td>RCIC Pump (0.70 g)</td><td>Fire Pump (0.62 g)</td><td rowspan="2">SW HVAC (0.63 g)
or
SW Pump (0.62 g)
or
SW Pump House (0.60 g)
or
Diesel Gen. (0.62 g)
or
Transformer (0.62 g)
or
MCCs (0.62 g)
or
Switch (0.63 g)
or
HPCF Pump (0.62 g)</td><td rowspan="2">Fire Pump (0.62 g)</td><td rowspan="2">RCIC (6E-2)</td></tr>
<tr><td>RCIC Pump (0.70 g)</td><td>HPCF Pump (0.62 g)</td><td>LPCF Pump (0.56 g)</td><td>V2 (0.62 g)</td></tr>
<tr><td>IE</td><td colspan="4">Reactor Building (1.12 g)
or
Control Building (1.11 g)</td><td colspan="3">None</td></tr>
<tr><td>IV (incl. IV-P)</td><td colspan="2">SW HVAC (0.63 g)
or
SW Pump (0.62 g)
or
SW Pump House (0.60 g)
or
Diesel Gen. (0.62 g)
or
Transformer (0.62 g)
or
MCCs (0.62 g)
or
Switch (0.63 g)
or
SLC Tank (0.62 g)
or
SLC Pump (0.62 g)</td><td colspan="2">Fuel Assemblies (0.62 g)
or
HCU (0.63 g)</td><td colspan="3">None</td></tr>
<tr><td>IA-P through IE-P</td><td colspan="2">SW HVAC (0.63 g)
or
SW Pump (0.62 g)
or
SW Pump House (0.60 g)
or
Diesel Gen. (0.62 g)
or
Transformer (0.62 g)
or
MCCs (0.62 g)
or
Switch (0.63 g)</td><td>RHR HtEx (0.70 g)</td><td>Fire Pump (0.62 g)</td><td colspan="3">None</td></tr>
</table>

How to read this table:

The columns for seismic only and seismic random are independent. Within each of those columns, the or function is obvious, and the and function is represented by a horizontal dotted line. Thus, for Damage Class IA, the seismic only column represents the Boolean expression [(SW HVAC + SW Pump + Diesel Gen. + Transformer + MCCs + Switch + HPCF Pump) * S/Rvs] and the same column for Damage Class IC represents the Boolean expression [Fuel Ass. * (SLC Tank + SLC Pump) * HPCF Pump].

GE used two methods to calculate its HCLPFs: min/max and convolution. The min/max method provides more safety insights. The convolution method may more appropriately (in a pure mathematical sense when a margins analysis is performed) take into account the fact that when one combines HCLPFs, one is working with the tails of probability distributions. If random failures and human errors are ignored when the min/max method is used, the plant HCLPF is 0.6*g*. With the min/max method, if random failures and human errors are included, the most significant combination of HCLPF, human errors, and random failures for the plant is also 0.6*g*. When the convolution method is used, the plant HCLPF is about 0.69*g* if random failures and human errors are ignored. All of these values exceed the 1.67 times SSE that represents the seismic robustness the staff expects to find.

For earthquakes that generate higher accelerations than the plant HCLPF, the staff no longer has the same high degree of confidence that core damage will not occur. However, the staff does not believe that a cliff-effect will exist for the ABWR design at or near the plant HCLPF and expects that the plant will have margin (perhaps quite a bit) above the HCLPF value.

The staff defines the most important seismic core damage sequences as those that have the lowest HCLPFs (seismic only) or the lowest combination of HCLPF with random failure or human error. Using these as the selecting criteria, none of the most important seismic core damage sequences involve random failures or human errors. The following are the most important sequences:

- (loss of emergency ac power or loss of emergency service water) and (fire water injection fails)
- (loss of emergency ac power or loss of emergency service water) and (scram fails)
- (scram fails) and (standby liquid control fails)
- (scram fails) and (standby liquid control fails) and (high-pressure core flood fails).

These results are similar to those from seismic margins analyses and PRAs performed previously.

The staff focused on the performance of and insights drawn from the ABWR seismic margins analysis. Because of the large uncertainty in hazard curves and the fact that seismic PRA results are dominated by the tails of the site hazard and SSC's fragility curves, it is expected that if a seismic PRA had been performed, it would have been one of the largest contributors to core damage frequency (though still a low absolute value). This is particularly true for the ABWR design since design changes or improvements have greatly reduced the estimated core damage frequency from internal events. As it is, the ABWR design should be better able to resist seismic events than most, if not all, existing nuclear power plants east of the Rocky Mountains because of its built-in safety margins.

The staff did not require GE to examine the HCLPFs of paths by which the containment could be bypassed, fail to isolate, or fail, since the containment structure was considered to be very rugged and none of the sequences that led to core damage had HCLPFs less than 1.67 times the SSE. Nevertheless, GE performed an evaluation of beyond-design-basis earthquakes to see how the containment would perform under high g-levels. This evaluation showed that no bypass paths are expected to occur with an HCLPF of less than 1.67 times the SSE. The lowest min/max HCLPF for bypass reported by GE was 0.74*g*.

19.1.3.3.1.2 PRA as a Design Tool

The following are examples of ways in which the ABWR design or procedures were modified by GE, based on the ABWR PRA-based seismic margins analysis:

- GE switched the seismic qualifications of the fire water pumps (i.e., the diesel-driven fire water pump took the place of the motor-driven pump as the pump capable of surviving an SSE) because failure of the diesel generators following a seismic event (by seismic or random failure) would otherwise have left the plant with no fire water addition capability.
- GE chose to lower the capacities of a few SSCs since it might have been difficult for a COL applicant to achieve the capacities in a cost-effective manner. The staff found this acceptable, since the lowered capacities did not affect the overall HCLPF of the plant and the original capacity assumptions seemed to be slightly higher than normally assumed in eastern U.S. seismic fragility analyses.
- GE developed recommendations for improving emergency operating procedures by instructing the operator to manually operate heat removal system valves if seismic-induced transformer loss should make power operation of these valves impossible.

19.1.3.3.1.3 Plant Features and Operator Actions Important to Risk

The margins approach does not allow a determination of which plant features are most important to risk. It does allow one to determine which plant features are important to the plant-level HCLPF and the redundancy or diversity available in achieving that HCLPF. In order to make this determination, the staff examined each sequence that led to core damage on the seismic event trees. None of the sequences has a seismic-only HCLPF of less than 0.5*g*. The sequences were examined to determine if lowering the HCLPF value of a single SSC (to a much lower HCLPF value) or increasing the demand failure rate of a single system (to a much higher demand failure rate) would result in a plant HCLPF of less than 0.5*g*. A review of the cut sets in Table 19.1-8 shows that most of the important sequences require at least two failures of SSC with HCLPFs above 0.5*g*. The two cases where this is not true are discussed below:

- Structural integrity — Most of the safety-significant structures (except for the service water pump house) are assumed to have an HCLPF of 1.11g or higher. If any of these structures were to be built with an HCLPF much lower than 0.5*g*, it would result in a much lower plant HCLPF. These structures include the control building, reactor building, containment, service water pump house, reactor pressure vessel (RPV) supports, and RPV pedestal.

- Batteries — In a seismic event, it is assumed that offsite power will be lost and ac power must be supplied by the EDGs. For the EDGs to start and load, they require dc power. The battery chargers or inverters, which provide dc power when ac power is available, are lost on loss of offsite power, until the EDGs start and load. However, the EDGs will not start and load unless dc power is available. Therefore the EDGs rely on the batteries to start and load in these circumstances. If the batteries or the dc cable trays were to have an HCLPF of less than 0.5*g*, it would lower the plant HCLPF accordingly.

All other seismic sequences require multiple failures of SSC the HCLPF of which is greater than 0.5*g* in order to cause the plant to experience core damage. As noted in SSAR Sections 19.9.5 and 19.H.5, a check of the capacity of as-built SSCs to meet the HCLPFs assumed in the ABWR PRA will be provided by a seismic walkdown. Details are to be developed by the COL applicant.

19.1.3.3.1.4 Human Reliability Insights and Important Human Actions

In the margins analysis, GE used the same human error rates and random failure rates that were used in the ABWR internal events analysis. The PRA-based seismic margins analysis did not identify any human reliability insights that were not already identified in the internal events analyses. There were no human actions or random failures that contributed to the plant HCLPF.

19.1.3.3.1.5 COL Action Items

GE identified COL actions that define the areas derived from performing the ABWR PRA-based seismic margins analysis that the COL applicant needs to perform or complete. These actions, including plant walkdowns, are detailed in SSAR Section 19.9 and are acceptable.

19.1.3.3.1.6 Insights from Uncertainty, Importance, and Sensitivity Analyses

One of the reasons for performing an uncertainty analysis is to help to display the range of values within which the results of an analysis could reasonable be expected to fall. The use of a PRA-based seismic margins analysis inherently makes use of the breadth of information being considered. This is because HCLPF values can be thought of as the g-level at which one has a 95 percent confidence that less than 5 percent of the time the equipment will fail (i.e., we are dealing with the tails of the curves). In addition, the staff does not require that a seismic hazards analysis be convoluted with equipment fragilities, since hazard curves have a large uncertainty that reduces their value in helping to make judgements about the seismic risk. From seismic PRA analyses, it is clear that uncertainties in the hazard curves would dominate the uncertainties in equipment or structure fragilities. For the ABWR PRA-based seismic margins analysis, no uncertainty analysis was performed because uncertainty is directly reflected in the margins method. Similarly, sensitivity studies were not performed on the margins analysis. Finally, the margins method does not result in either core damage frequency or risk results. Therefore, importance analyses were not performed.

19.1.3.3.2 Fire

In a number of PRAs for operating plants, fires have shown up as important contributors to core damage frequency and risk. GE has taken a unique and effective approach to analyzing beyond-design-basis fires. In performing its fire analysis (a combination of the FIVE methodology and the ABWR internal events PRA), GE chose to simplify its analysis by assuming that any fire that started in a divisional fire area while the plant was at power would immediately cause all equipment to fail in that divisional fire area. While this assumption greatly simplifies the performance of a fire analysis, it biases the numerical results in a conservative manner. GE suggests and the staff agrees that it would be inappropriate to compare the numerical results from the ABWR fire analysis to the results of the more realistic internal events analyses. Analysis of the potential consequences of a fire while in Modes 3, 4, and 5 are investigated in the shutdown risk analysis discussed in Section 19.1.3.4.

The ABWR design has a number of attributes that have significantly improved the detection, suppression, and confinement potential of fires. The most important improvement is that the ABWR was designed with fire prevention and mitigation in mind. The ABWR does not need to rely on spatial separation as a barrier between safety divisions. The three safety divisions are separated by 3-hour fire barriers throughout the plant, with few exceptions (See SSAR Section 9A.5). Detection systems and suppression systems will be placed for optimal advantage.

19.1.3.3.2.1 Dominant Accident Sequences

Because GE did not perform a fire PRA, but rather used a combination of the FIVE methodology and the ABWR internal events PRA, one cannot determine dominant sequences in the sense normally used when working with a PRA. For the fire analysis, the word "dominant" appears in quotes to emphasize that the use of this terminology in the context of a fire analysis should not be taken in the same way as it would be for a PRA (i.e., no cut sets were developed for the fire analysis and therefore importance analyses cannot be performed). The staff identified the "dominant" sequences leading to core damage in the systematic fire analysis performed by GE on the basis of engineering judgement, core damage frequency point estimates, and descriptions provided by GE of what it considers the most important sequences. The staff believes this approach is adequate to identify fire vulnerabilities.

GE identified the most important sequences as those that begin with a fire in the control room. In these "dominant" sequences, a control room fire starts, feedwater is lost, RCIC or one high-pressure core flood train is lost, and either one train of low-pressure core flooder or manual depressurization fails. The fire analysis screened out all additional sequences and fire areas because the estimated screening core damage frequency associated with each fire area was less than 1E-6 per year.

The staff finds 1E-6 per year to be an acceptable screening level for the ABWR since the fire analysis methodology used by GE assumes that any fire that starts in a physically separated divisional area instantaneously causes all the equipment in that division to fail. Although this assumption is conservative, the separation of divisions in the ABWR design still results in low core damage frequency estimates.

19.1.3.3.2.2 PRA as a Design Tool

On the basis of the ABWR fire analysis, GE made a design change to assure that certain fire protection capabilities existed in the as-built plant. The change was made because GE determined that unless the capability to control ADS valves from the remote shutdown control panel was improved (controls for a fourth safety relief valve (SRV) were added) and unless it was possible to control the RCIC system locally (RCIC could not be controlled from the remote shutdown panel), the control room would not pass the 1E-6 per year screening value.

19.1.3.3.2.3 Plant Features and Operator Actions Important to Risk

Several plant features and operator actions are important to reducing the consequences of a fire in an ABWR.

Design Features

- Separation of the three safety divisions — The most important is the 3-hour fire barrier that is to be in place among the three safety divisions. The fire analysis assumes that there will be no spread of fire or suppressants among the divisions, once a fire begins.

- Smoke control system — This system helps prevent the migration of smoke to other divisions if a fire occurs. This is accomplished by pressurizing the surrounding areas so that the smoke will be contained in the fire zone. Smoke control is particularly important in secondary containment.

- Inerting of containment at power — The inerting of containment precludes the need to analyze fires in containment while at power. Containment is one of the few areas in the plant where all four divisions (three safety, one non-safety) are in the same fire area. The control room is another area where multiple divisions coexist.

- Capability to operate RCIC from outside the control room — The largest contributor to core damage frequency among fire-initiating events is a control room fire. For control room fires, it is important to be able to control RCIC locally, since it cannot be controlled directly from the remote shutdown panel. This is both a design feature and an important operator action.

- Capability to operate four SRVs from the remote shutdown panels — As with RCIC, for control room fires, it is important to be able to operate four SRVs from the remote shutdown panel. Successful depressurization is achieved by manual operation of three of the four SRVs. If only three SRVs were on the panel, a single failure of one of the SRVs would prevent successful mitigation of the control room fire. This is both a design feature and an important operator action.

- Remote shutdown panel — The panel is required for shutdown of the plant in the event that the control room needs to be evacuated.

19.1.3.3.2.4 Human Reliability Insights and Important Human Actions

The ABWR fire analysis considered random failures and human errors modeled in the internal events analysis. Operator control of the RCIC system from outside the control room was found to be an important human action, should the control room needs to be evacuated.

19.1.3.3.2.5 Insights From Uncertainty, Importance, and Sensitivity Analyses

Because the methodology used in performing the ABWR severe accident fire analysis (a combination of the FIVE analysis, which is primarily a deterministic evaluation of fires, and portions of the ABWR PRA internal events analysis) is bounding in nature, the results of uncertainty, sensitivity, or importance analyses would be biased. Since the purpose of performing an uncertainty analysis is to better understand the subject being investigated and since in the case of the fire analysis it is unclear what the results of an uncertainty analysis would represent physically given the bounding nature of the assumptions in the analysis, the staff finds that an uncertainty analysis was not required. The staff concludes that the fire methodology used by GE provides the insights needed to determine whether fire vulnerabilities exist for severe accident fires and whether fires represent a disproportionate risk.

19.1.3.3.3 Internal Floods

At some plants, PRAs have shown that internal floods are leading contributors to core damage frequency. Utilities that have performed a systematic internal floods analysis have generally found design modifications that they deemed prudent to implement in order to maintain adequate prevention and mitigation capability for internal floods. GE performed a comprehensive internal flood PSA for ABWR power operation. In the shutdown heat removal analysis, GE performed a separate flooding analysis that evaluated the flood protection provided by the plant during shutdown by controlling barriers between divisions and by controlling equipment configurations.

19.1.3.3.3.1 Dominant Accident Sequences

The ABWR Probabilistic Flooding Analysis (flood analysis) in Appendix 19R of the SSAR estimates that the chances of an internal flood causing core damage is very small (about 7E-9 per year).

For the internal flooding analysis, the word "dominant" appears in quotes to emphasize that the use of this terminology in the context of the flooding analysis should not be taken in the same way as it would be for a PRA performed using more realistic assumptions or for an analysis that has cutsets on which importance analyses can be performed. The ABWR internal flooding analysis made several conservative assumptions that could bias the results. GE has identified the "dominant" sequences leading to core damage in the internal flooding analysis on the basis of sequences with the largest estimated core damage frequency and on engineering judgment.

The "dominant" sequences that are initiated in the turbine building are those associated with the circulating water system (CWS) or the turbine service water (TSW) system. The following sequence is "dominant" for the turbine building: a large pipe breaks in the CWS system, the isolation valves in the CWS lines fail to close, water fills up and runs out of the condenser pit, and the fire door between the turbine building and the service building is either open or fails open allowing water into the service building. The service building floods and a door between the service building and the control building fails open or is open. Water enters the control building and causes electrical power supplies and all three divisions of reactor building cooling water (RCW) to fail. The estimated frequency of this event is 3E-9 per year.

The most important internal floods initiated in the control building are those associated with the reactor service water (RSW) system. This sequence involves the following: a large pipe breaks in the RSW piping in the RSW/RCW room and the operator fails to isolate the flooding. The automatic RSW pump trip fails and the water flows into the remaining RSW pump rooms. The operator fails to respond to the flooding alarm and the RSW fails. The estimated frequency of this event is 2E-9 per year.

Floods that begin inside the reactor building are divided into two parts: those inside and those outside secondary containment. The most important internal flood sources that initiate inside secondary containment are the suppression pool, condensate make-up, and fire water. The lowest floor of the reactor building is entirely within the secondary containment. On this level, each of the three safety divisions has a separate ECCS room that has an alarmed watertight door and a sump pump. Floods inside these rooms would cause the HPCF, RCIC, RHR, and control rod drive (CRD) pumps, and the CRD hydraulic control unit to fail. A common hallway into which the watertight doors open runs around the perimeter of the division rooms. There are three worst- case flood sequences for the reactor building. The three flood sequences are developed in event trees in SSAR Appendix R, Figures 19R.5-4, -5, and -6. The largest estimated contributor to core damage among these three flood sequences occurs inside an ECCS room resulting from a leak in the suppression pool suction line upstream of the isolation valve. This is a nonisolable break.

In the review of the risk resulting from internal floods, several concerns were identified. The main concerns identified for turbine building floods were (1) the assumptions regarding the likelihood of the truck door failing and allowing turbine building flood waters to exit the turbine building without challenging the fire door between the turbine building and the service building and (2) the justification for the assumed reliability for the fire door. GE satisfactorily addressed these staff concerns in Section 19R of the SSAR. In the reactor building, the chief concern identified was the lack of any common-cause failure analysis for the failure of the isolation measures credited in the analysis. GE performed an improved common-cause failure analysis that resulted in no change in internal flood insights. For control building floods, the staff was concerned that failure of the discharge or intake valves following a service water system pipe break in an RSW or RCW room would result in an unisolable flood that could affect all three divisions of the RSW system. GE added design requirements for antisiphon capability to prevent continued flooding in the event the RSW pump is tripped but the isolation valves do not close. In addition, GE stated that the ABWR would be designed so that the ABWR ultimate heat sink (UHS) cannot gravity drain into the control building.

19.1.3.3.3.2 Comparison of Dominant Sequences

Conservatism in the ABWR flooding analysis biases the results and may make it unsuitable for comparison to the more realistic internal events results. Conservatisms include the assumptions that all piping breaks are double-ended shears and once a flood in a division causes one piece of equipment to fail, all equipment in that division is assumed to fail.

19.1.3.3.3.3 PRA as a Design Tool

In performing its internal flooding analysis, GE determined that several areas of the design needed to be modified or strengthened to help make sure that internal floods do not present an unacceptable level of risk for the ABWR design. These modifications include the following:

- GE modified the motor control centers (MCCs) to have NEMA Type 4 enclosures to protect the MCCs from water spray from a pipe break or leak.

- GE added four water-level sensors in the condenser pit. When actuated, these sensors send an alarm to the control room, trip the CWS and TSW pumps, and close isolation valves in both systems. This isolates floods in the turbine building caused by pipe breaks in the TSW and CWS.

- GE added four room-floor water-level sensors, which send an alarm to the control room when water is first detected in the room, and four diverse sensors, which automatically trip the reactor water service water/RCW pumps at a higher water level in the RCW/RSW room. The sensors will alert the operators to RCW leakage and will isolate the flood.

- GE added to the Tier 1 description of the control building in the design control document (DCD) the design requirement that no more than 4,000 meters (4,374 yds) of pipe exist between the isolation valves at the RSW pump house and the control building (2,000 meters each, for supply and return lines). This limits the amount of water that can be drained into the RCW/RSW room following RSW pump trip during flooding.

- GE added antisiphon capabilities to the RSW system to end a flood if the RSW pump trips but the isolation valves do not close.

19.1.3.3.3.4 Plant Features and Operator Actions Important to Risk

Several plant features are important in reducing the chances or consequences of an internal flood. These features are discussed below:

- Separation of the three safety divisions — Whenever the three safety divisions are in a building, they are separated by barriers. The most important barrier is the 3-hour fire barrier that must be in place between the three safety divisions. The flooding analysis assumes that there is a limited spread of water among the divisions, once a flood begins. The primary assurances that flooding will not get high enough to affect SSCs in multiple divisions are that (1) the 3-hour barriers are in place, (2) there are elevator shafts with no sills that carry water to lower reactor building levels, (3) stairwells on all floors have doors that provide a leak path (up to a 1.9 cm (0.75 in.) gap at bottom of fire door) down the stairs to the lowest building level, (4) there is at least one drain in every room, (5) all pipes penetrating floors have 200 mm (8 in.) sills that help prevent sneak-path leaks, (6) all hatches and electrical penetrations through floors are to be able to withstand 200 mm (8 in.) of standing water without leaking, and (7) except as noted in SSAR Section 19R.4.2.3, equipment from multiple divisions does not occupy the same divisional area. Barriers include concrete fire barrier floors, ceilings, and walls; partitions; rated watertight doors; penetration seals for process pipes and cable trays; and special assemblies and constructions.

- Auto trip of the CWS and TSW pumps — The three CWS pumps and the two TSW pumps provide unlimited supplies of water to the turbine building. If a flood were to occur, level sensors would alert the control room operators, trip all five pumps, and close all isolation valves in both systems.

- Maximum length of RSW pipe — The COL applicant will have to limit the RSW piping distance between the control building and the RSW isolation valves to 4000 meters (4,374 yds)(2,000 meters each, for supply and return lines). A longer length would provide enough water to flood one RCW room, overflow, and start to flood another RCW room.

- Auto trip of the RSW pumps — The RSW pumps trip and the RSW isolation valves close in the affected division should flooding occur in the control building. The trips limit the volume of water that can be added to a control building flood. Trip sensors are diverse from sensors that alert the operators early on (i.e., at a lower level) to a flood in the room.

- Antisiphon capability — If the RSW pumps trip on a high water level caused by a line break, but the isolation valves fail to close, antisiphon capabilities will be needed to prevent further flooding because otherwise the siphon effect will continue to draw water from the ultimate heat sink into the rooms.

- ECCS room watertight doors — The doors between the ECCS rooms and the outside corridor in the reactor building are watertight. These doors have dogs and alarms. The doors do not send an alarm if they are physically closed, but not "dogged." The doors are designed to limit flooding in a division to the particular ECCS divisional area.

- Reactor building corridor volume — The volume of the reactor building corridor surrounding the three ECCS divisions at the lowest level of the reactor building is sufficiently large to handle large breaks. The corridor can hold water from a suppression-pool-driven flood at the equilibrium water level with the suppression pool and have the water level not exceed the ceiling of the corridor. This limits the flood potential to one ECCS division and the corridor.

- Drip-proof designs and NEMA Type 4 enclosures — All electric motors have drip-proof designs and all motor control centers have NEMA Type 4 enclosures that protect electrical equipment from water spray from above.

Further details of the ABWR design features that help to prevent or mitigate internal flooding are given in SSAR Appendix 19R, Table 19R.6-2. In addition, the following are important insights into the ABWR internal flooding analysis:

- The only buildings modeled in the PRA for flooding where internal flooding could damage safety-related equipment or cause plant transients are the turbine building, control building, and reactor building.

- The service water pump house, which is outside the ABWR certification scope, is a building that must be designed to prevent internal floods from impairing multiple safety trains. Flooding of the pump house would cause all ECCS pumps to fail except for RCIC. For injection to the vessel, the operators also would have the fire water pumps and the condensate pumps.

- Secondary containment in the reactor building is designed to mitigate internal floods that begin at elevations above the lowest elevations by directing flood water to floor drains and stairwells that are routed to the lowest elevation. It is important that the drains be sized to drain at a rate that does not permit a flood on a particular level to rise high enough to damage safety-related equipment, pressurize the volume within which it is contained, or spill over to other divisions.

- From SSAR Appendix 19Q, the recommended shutdown configuration is as follows: one RHR division and its support systems should be operating, the second safety division should be administratively controlled to not be in maintenance and its barriers should be intact, and the third safety division may be undergoing maintenance.

- Fire doors generally are not considered to be capable of holding back a large head of water.
- If a watertight door is physically shut but not "dogged," it will not alarm to indicate it is incapable of holding back a flood inside the door.
- The service water system and the CWS are assumed to be designed and located so that they cannot gravity feed to the plant.
- The ACIWA system can provide water to the core or spray the drywell in the event of a catastrophic internal flood. Manual valves to direct the flow to either the core or to the drywell are located in the reactor building and can be operated successfully following an internal flood.

19.1.3.3.3.5 Insights Into Human Reliability and Important Human Actions

Although postulated floods can be mitigated from a risk perspective with few operator actions, because of the inherent ABWR flooding capability (the frequency of internal floods leading to core damage without taking any credit for operation action is still quite low), timely implementation of the following operator actions can limit potential flood damage:

- Isolation of flood sources following detection by sump pump operation and alarms or floor water-level detectors (for floods in the turbine building, the operator should attempt to isolate the leak and shut down the plant without losing condenser vacuum to avoid a "turbine trip without bypass" scenario)
- Closure of watertight doors to prevent damage to equipment in more than one safety division
- Opening of certain nonwatertight doors or hatches to divert water from safety-related equipment.

19.1.3.3.3.6 Combined License Applicant Action Items

GE has identified COL Action Items that define the actions derived from performing the ABWR internal flood analysis that the COL applicant needs to perform or complete. These actions are detailed in SSAR Section 19.9. The staff finds these COL Action Items to be acceptable.

19.1.3.3.3.7 Insights From Uncertainty, Importance, and Sensitivity Analyses

The ABWR internal flooding analysis methodology made bounding assumptions to simplify the task of evaluating floods. This resulted in an evaluation that cannot readily be manipulated to provide uncertainty, importance, or sensitivity insights because of its internal biases. Therefore, GE did not perform any uncertainty, importance, or sensitivity analyses on its internal flooding analysis nor did the staff require these analyses to be performed. GE identified important design features through engineering judgement. The staff finds this approach to be acceptable.

19.1.3.3.4 External Floods

In SECY-93-087, the staff identified the need for a site-specific probabilistic safety analysis and analysis of external events. GE did not perform an analysis (PRA or bounding) of the capability of the ABWR design to withstand external flooding. Instead, GE assumed that the ABWR standard design would be sited such that its grade level would be 30.5 cm (1 ft) higher than the probable maximum flood level as stated in Section 2.6.2 of the SSAR. However, estimates of the return periods of river floods at various nuclear power plant sites that would exceed the probable maximum flood level range from probable to very improbable. For some sites where the return period of large floods is high, the ABWR design may have vulnerabilities to external flooding.

Therefore, the staff will require, where applicable to the site, that the COL applicant perform a site-specific PRA-based analysis for external flooding to search for site-specific vulnerabilities.

19.1.3.4 Operation in Modes Other Than Full Power and Startup — Level 1

This section details the staff's safety insights drawn from the review of the ABWR shutdown heat-removal reliability study performed by GE. Although the staff found that the most significant events to date have occurred at pressurized water reactors (PWRs), the potential vulnerability of BWR plants to shutdown and low-power events cannot be ignored. GE submitted a shutdown risk evaluation of the ABWR design (SSAR Appendix 19Q). The evaluation covered Modes 3 (hot shutdown), 4 (cold shutdown), and 5 (refueling). It included all aspects of the nuclear steam supply system (NSSS), containment, and all systems that support the NSSS and containment. It did not address fuel handling outside the primary containment or fuel storage in the spent fuel pool.

The evaluation covered important aspects of draft NUREG-1449, "NRC Staff Evaluation of Shutdown and Low Power Operation," such as decay heat removal (DHR), inventory control, containment integrity, electrical power, reactivity control, and instrumentation. The analysis examined 70 loss-of-power and more than 100 loss-of-DHR precursor events at operating BWRs. For new features of the ABWR design, GE developed a failure-modes-and-effects-analysis to determine if the new features would introduce additional shutdown vulnerabilities. GE performed a detailed reliability study of the ABWR DHR function. This study included fault and event trees for all DHR and make-up systems. The analysis separately investigated fire and flooding at shutdown, along with the potential effect of increased maintenance of important equipment.

The staff finds that the ABWR design includes enhanced features that reduce risk during shutdown operations when compared with operating BWRs. These features, including three independent RHR divisions, three EDGs, an ACIWA system, an alternate onsite CTG, and proper plant electrical and physical separation and layout, specifically address the more risk-significant operations during shutdown identified in NUREG-1449. GE provided appropriate safety guidance for effective outage planning and control and provided TS to ensure adequate systems are available to respond to events that may occur during shutdown. Implementation of these recommendations by a COL applicant will be reviewed by the staff.

The staff concludes, based on previous shutdown analyses (both international and U.S. operating reactors) and the ABWR Shutdown Risk Evaluation, that the chances of a core damage event occurring when in Modes 3, 4, or 5 is probably on the same order of magnitude as that of internal events occurring in Modes 1 and 2.

19.1.3.4.1 Dominant Accident Sequences

GE and the staff determined that it would not be useful to attempt to identify dominant accident sequences during Modes 3, 4, and 5. This is because, among all possible plant shutdowns and even on a minute-by-minute basis during any shutdown, the plant configuration could and probably would change. During each of these different configurations, one could conceivably have different dominant sequences and therefore different insights. Because of this, no dominant sequences based on the ABWR Shutdown Risk Evaluation were identified. NUREG-1449 identified that the majority of shutdown precursors and actual shutdown events were caused by human error. Operating, administrative, and emergency procedures as well as aspects of the ABWR design that minimize the chances of human error are discussed below.

19.1.3.4.2 PRA as a Design Tool

GE made no design changes specifically based on the ABWR shutdown risk evaluation. However, it did develop procedural guidance for outage planning and did develop for COL applicants a short list of acceptable system configurations, minimum combinations of systems to ensure adequate shutdown safety margins, when in Modes 3, 4, and 5. This guidance is located in SSAR Section 19.Q.7.9, Tables 19Q-3 through 19Q-5 and is repeated in Table 19.1-9 in this report. The tables were developed based on a single initiating event during shutdown (the failure of the operating DHR division), but are valid for initiating events during shutdown including loss of offsite power and the loss of the operating service water division.

19.1.3.4.3 Vulnerabilities

The staff believes that the results of the ABWR shutdown risk evaluation provide the information needed to decide if there are shutdown vulnerabilities and whether operation in modes other than full power represent a disproportionate risk in the ABWR. GE did not identify any shutdown vulnerabilities. With the use by a COL applicant of the tables (and accompanying or similarly acceptable methodology) that define sets of equipment that should remain operable for the ABWR to meet GE's proposed goal for conditional core damage frequency, 1E-5 given the loss of a DHR train, when in Modes 3, 4, and 5, the staff finds that operation of the ABWR in Modes 3, 4, and 5 does not represent a disproportionate risk. In SSAR Appendix 19Q.9, GE has investigated whether the new features in the ABWR design might have introduced vulnerabilities when in modes other than full power. GE concluded that they did not introduce new vulnerabilities, and the staff concurs with GE's conclusion. The staff reviewed the ABWR shutdown evaluation and found no unreported shutdown vulnerabilities.

19.1.3.4.4 Plant Features and Operator Actions Important to Risk

A detailed list of ABWR features that are important to risk in Modes 3, 4, and 5 are provided in Tables 19Q.4-1 in the SSAR. The following list outlines those areas of the ABWR design that are important to maintaining risk during shutdown operations at a low level.

<u>Features and Actions Minimizing Loss of DHR</u>

- Having three divisions of RHR that are physically separated lowers the frequency of loss of DHR.

Table 19.1-9 Examples of configurations that meet 1E-5 CCDF goal in Modes 3, 4, and 5

Example Minimum Sets of Systems for Modes 3 and 4

	RHRB	Main Cond.	CUW	HPCFB	CRD	ADS	RHRB (CF)	Condensate	Fire Water
1	X				X	X	X		X
2	X		X			X	X		X
3	X					X	X	X	X
4	X	X				X	X		X
5	X			X	X	X		X	

Examples of Minimum Sets of Systems for Mode 5 (Unflooded) (2 to 3 Days After Shutdown)

	RHRB	HPCFB	CRD	Condensate	Fire Water	RHRB (CF)
1	X		X		X	X
2	X			X	X	
3	X	X	X	X		

Examples of Minimum Sets of Systems for Mode 5 (Flooded) (3 or More Days After Shutdown)

	RHRB	FPC	CUW	HPCFB	CRD	RHR (CF)	Condensate	Fire Water
1	X						X	X
2	X				X	X		
3	X			X	X		X	
4	X					X	X	X
5*			X		X			X
6*			X				X	X
7**		X			X		X	
8**		X					X	X
9**		X			X			X

* After 8 days
** After 10 days

- If all RHR loops were unavailable, (a) steam from the reactor could be directed to the main condenser (if available) and make-up could be supplied to the vessel by many sources discussed in Inventory Control below, or (b) the suppression pool could be used as a heat sink via the SRVs, or (c) the reactor water cleanup system could be a heat sink, or (d) the spent fuel pool inventory could be a heat sink if the reactor water level were raised to the refueling level, or (e) if the vessel head were removed, bulk boiling of reactor coolant in the vessel with adequate make-up would prevent fuel damage.

- There is no isolation of shutdown cooling on the loss of reactor protection system (RPS) logic power.

Features and Actions Minimizing Loss of Inventory

- The vessel level is displayed for the operator in the control room during all shutdown configurations, including refueling.

- Multiple sources of make-up exist such as the suppression pool, condensate storage tank, main condenser hotwell, and ac-independent water addition system.

- RHR system valves are interlocked with reactor system pressure to help ensure that RHR system low-pressure piping is not subject to full system pressure. However, even if it were exposed to full system pressure, the RHR piping should withstand full reactor pressure without rupture.

- Make-up can be provided by the CRD hydraulic system, the reactor water cleanup system, the condensate pumps in conjunction with the hotwell, the ACIWA system, and the RHR system.

- The mode selector switch automatically realigns the valves, as required, for the RHR mode selected. In the past, operator errors led to sending water to the wrong place. Now, all the operator has to do is change the mode switch to realign the system automatically.

- The shutdown cooling piping connects to the nozzle in the vessel above the level of the active fuel, so fuel cannot be uncovered by a siphoning effect.

- Suppression pool drain down has been identified as a period during which there is a diminished level of protection against core damage. It is important that a COL applicant properly coordinate suppression pool drain down with TS equipment configuration requirements and the DHR core damage frequency goal (1E-5 conditional core damage frequency).

Features Minimizing Loss of Containment Integrity

- During shutdown with the drywell head removed, the ABWR has a secondary containment that will automatically be isolated on high radiation from a radiological boundary breach or fuel-handling accident. If the radiological accident is one that does not pressurize secondary containment, the filtering function of standby gas treatment system (SGTS) will be a benefit. If the radiological breach is caused by boiling in Mode 5, secondary containment will overpressurize and fail. In this circumstance, the SGTS filtering function will be lost; however, plate out via a tortuous path will tend to reduce consequences. If the core can continue to be covered by water, then boiling is a relatively benign event (with respect to radiological consequences). If, however, the core subsequently becomes uncovered for an extended period with the head off and secondary containment breached, there would be a massive radiological release. It takes approximately 3 hours to boil down the water above the core and longer to heat up the core. This should be an extremely low probability event since the ABWR design includes multiple means to provide water to the core in all modes of operation.

ABWR Features Minimizing Loss of Electrical Power

- There are three physically independent 1E diesel generators.

- There is a CTG that can be used to power any of the Class 1E or non-Class 1E buses. This generator can start a feedwater or other pump for DHR or inventory make-up if required.

- There are four divisions of dc power.

Other Design Features and Operator Actions Minimizing the Chances of Core Damage When Not at Power

- Fires and floods during shutdown can be mitigated by ensuring, through administrative procedures, that at least one safety division is not in maintenance and its physical boundaries remain intact. If it is decided to breach the boundaries of two safety divisions to complete maintenance tasks, an evaluation could indicate if the minimum set of systems capable of meeting the shutdown safety criteria would be available if a fire or flood were to occur. The analysis would indicate if a minimum set were available should a flood or fire occur in the intact division or the breached divisions.

- The staff believes that the most important element in control of shutdown risk is adequate planning of maintenance on systems and support systems that can be used to remove decay heat or supply inventory make-up to the vessel. Maintenance planning is the responsibility of the COL applicant, but GE has provided guidance in its shutdown risk evaluation (SSAR Appendix 19Q.10) to help ensure that the planning will not place the plant in an unfavorable configuration from the standpoint of expected core damage frequency.

Appendix K of this report lists those PRA-based safety insights that were drawn from the PRA. The appendix documents the disposition of these insights and indicates if the insights are in ITAAC, Tier 2 information, TS, COL Action Items, Interface Items, or the RAP.

19.1.3.4.5 Insights Into Human Reliability and Important Human Actions

A total of eight human actions were identified as important for controlling risk during shutdown. Five actions were treated probabilistically in the evaluation, and three were treated deterministically (i.e., they were assumed to be taken if needed):

- Recognition of failure of an operating RHR system
- Initiation of standby RHR following loss of the operating division
- Use of non-safety grade equipment for DHR (e.g., reactor water cleanup (CUW) and main condenser)
- Use of non-safety grade equipment for inventory make-up (e.g., CRD, feedwater, condensate)
- Use of boiloff for DHR with the RPV head removed
- Implementation of fire or flood watches during periods of degraded safety equipment integrity
- Firefighting during shutdown operations, possibly with part of the fire protection system in maintenance
- Use of the remote shutdown panel during shutdown operation.

Because these actions and instrument requirements are considered significant, they are included in COL Action Items (See SSAR Section 19.9).

19.1.3.4.6 Combined License Applicant Action Items

COL Action Items based on the shutdown evaluation are identified in SSAR Appendix 19Q.12.3. These actions are contained in the detailed discussion of Chapter 19 COL Action Items given in SSAR Section 19.9.11. The staff finds these COL Action Items to be acceptable.

19.1.3.4.7 Insights From Uncertainty, Importance, and Sensitivity Analyses

The greatest uncertainty in a shutdown evaluation is the actual configuration of the plant. This is particularly true since there is no agreed-upon method of modeling plant configurations during shutdown. To bound this concern, the ABWR shutdown evaluation made a conservative assumption that all equipment not included in the minimum combinations of equipment needed to meet the goal of conditional core damage frequency was unavailable. GE and the staff believe that this conservatism would overshadow any uncertainty analysis or sensitivity analysis that might be performed. No importance analyses were performed since the importance of an SSC will vary depending on the particular configuration the plant is in.

19.1.3.5 Level 2 Analysis

19.1.3.5.1 Containment Performance and CCFP

The results of the Level 2 and 3 portions of the ABWR PRA indicate that the ABWR containment is quite robust and able to accommodate severe accidents with a low attendant probability of containment failure. In assessing the probability of containment failure, two alternative definitions of containment failure were considered: (1) loss of containment structural integrity, and (2) releases that result in doses of 25 rem or greater at a distance of 0.8 km (0.5 mile) from the reactor. The GE and staff estimates of CCFP are presented in Table 19.1-10. The staff estimates are based on a "staff-adjusted" ABWR risk profile that reflects staff views on selected issues. The most significant staff adjustment was to increase the frequency of early containment failures to account for uncertainty in the magnitude of direct containment heating loads and the contribution to CDF from unisolated LOCAs outside containment and ATWS events. Adjustments to account for uncertainties in source terms and consequence modelling were also made but affect only the dose definition of CCFP.

Table 19.1-10 GE's point estimates and the staff's mean estimates of the internal events risk

Performance Measure	GE Updated PRA		Staff-Adjusted Result	
	CFP	CCFP	CFP	CCFP
Structural Integrity	7.7E-10 2/	0.005 2/	4.1E-9 1/	0.026 1/
Dose Definition	3E-10	0.002	1.6E-8 3/	0.10 3/

Notes:

1/ Staff estimates reflect: (1) contribution from unisolated LOCAs outside containment and (2) increased probability of containment failure because of DCH.

2/ Based on GE's estimates of early and late containment failure frequency.

3/ Staff estimates include all frequency from Cases 7, 8, and 9, plus 58 percent of the frequency of Case 1 based on staff calculations using MACCS.

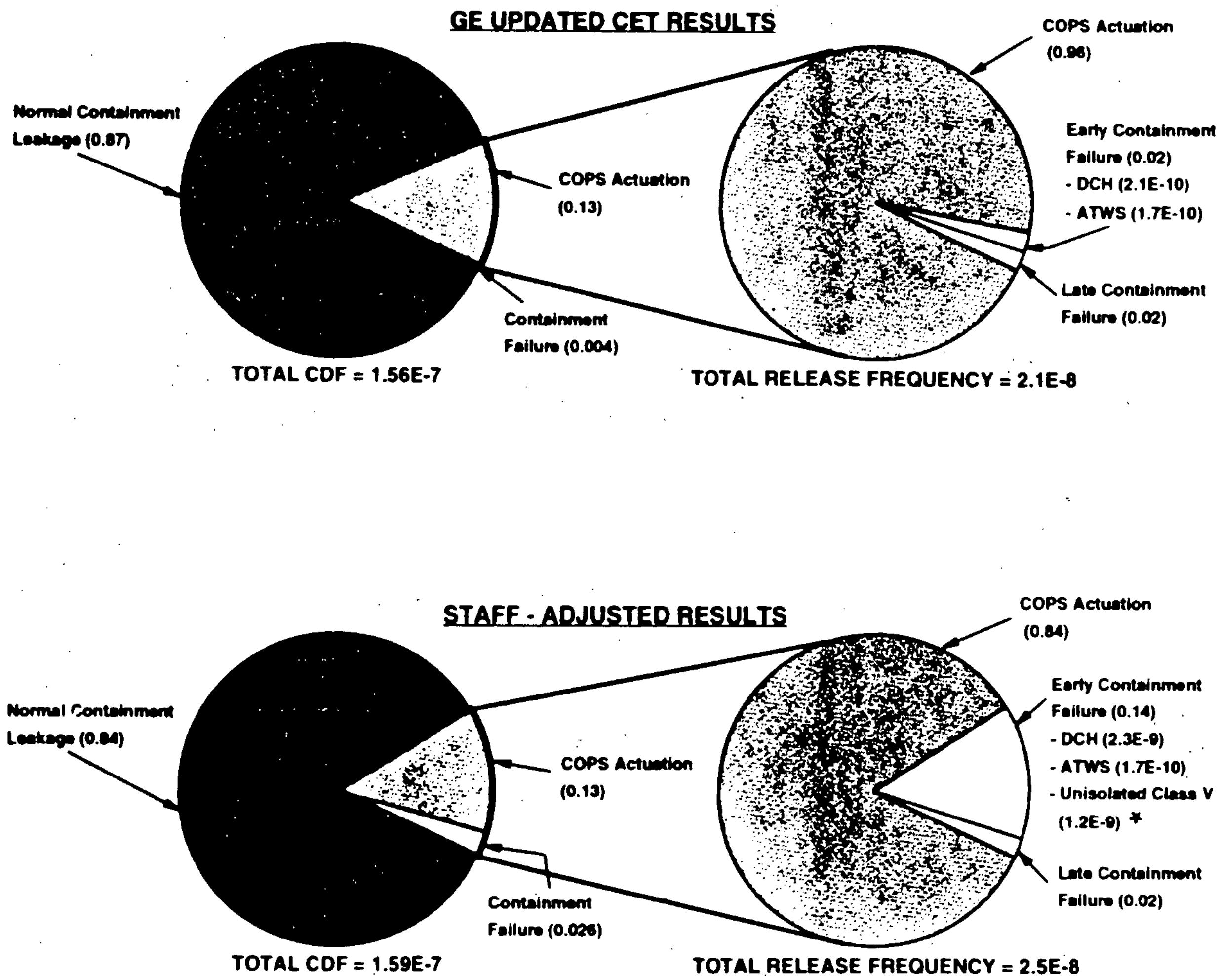
Both the GE and the staff estimates of the ABWR CCFP are within the Commission's containment performance goal (0.10) using either containment failure definition. Combined with the core damage frequency estimate from the Level 1 portion of the PRA (1.6E-7), this results in an extremely low likelihood of containment failure in absolute terms (i.e., on the order of 1E-8 to 1E-9 per year). GE's estimate of CCFP using the structural integrity definition is 0.005. The staff's estimate is 0.026, and is higher due to the treatment of LOCAs outside containment as containment failures, and the higher loads associated with DCH in the staff's assessment. GE's estimate of CCFP based on the dose definition is 0.002. In contrast, the staff's estimate of CCFP using the dose definition of containment failure is about 0.1. The staff's estimate is higher because of primarily (1) a higher source term used by the staff to represent scenarios with vented containment and suppression pool scrubbing, and (2) the use of the Melcor Accident Consequence Code System (MACCS) code for the consequence in calculations. However, the staff concludes that CCFP does not exceed 0.1, even under conservative staff assumptions, and therefore, meets the Commission's safety goal.

19.1.3.5.2 Leading Sequences for Containment Failure

The breakdown of contributors to containment failure for internal events is presented in Figure 19.1-2 in the form of a pie chart. For internal events, the bulk (about 85 percent) of the core melt events in the ABWR are successfully contained, with the releases to the environment limited to leakage on the order of the design-basis containment leak rate. A small fraction (about 13 percent) of core damage events result in actuation of the COPS and releases to the environment via the stack. While this is the dominant release mode for the ABWR, the consequences of these releases are significantly reduced because of the relatively late time of the release (about 20 hours from accident initiation), and fission product removal by the suppression pool. As a result, these releases do not dominate the ABWR risk profile.

Based on the staff assessment, only about 3 percent of core melt accidents result in containment failure. The bulk of these failures (about 90 percent of the 3 percent) are classified as early failures, relative to the time of core melt. Major contributors are DCH, LOCAs outside containment that proceed to core melt, and overpressure as a result of ATWS (The DCH frequency is based on a GE sensitivity study, as discussed in FSER Section 19.1.3.5.4; the LOCA and ATWS frequencies are based on GE's baseline Level I PRA). Late failures constitute only about 10 percent of the containment failure frequency. Drywell head or penetration overtemperature and rupture disk failure (i.e., failure of the drywell before rupture disk actuation) are the major contributors.

Figure 19.1-2 Breakdown of ABWR containment release frequency from CET analysis



* Based on GE results before Amendment 32. GE values for unisolated LOCAs outside containment reported in Amendment 32 are about a factor of 10 lower, however this does not affect the overall staff findings.

19.1.3.5.3 Important Design Features for Containment Performance

GE added a passive flooder system to the lower drywell to improve the chances that a core-on-the-floor (following a postulated reactor vessel failure) would be covered by water. This is designed to help quench the corium and reduce the drywell temperature and pressure from noncondensable gas generation.

GE added a COPS to protect the integrity of the containment (i.e., to assure that fission products must pass through the pool rather than bypass it by preventing failure of the drywell head) in slow containment overpressure events. The COPS allows isolation of the containment after the opening of the COPS rupture disks.

Suppression Pool — The suppression pool is an important containment feature to prevent severe accident progression and to promote fission product removal, since releases from the reactor vessel are either directly routed to the pool (e.g., transients with actuation of ADS) or pass through the pool via the drywell-wetwell connecting vents. However, the suppression pool function can be compromised in the ABWR design by a single failure of a wetwell or drywell vacuum breaker (i.e., a stuck open vacuum breaker) or by excessive leakage of one or more vacuum breakers. While a similar containment challenge exists for all operating BWRs, the frequency of suppression pool bypass is expected to be somewhat greater for the ABWR since the ABWR has only a single vacuum breaker in each wetwell-drywell connecting path, in contrast to two valves in series in each path in operating BWRs. The impact of suppression pool bypass is minimized in the ABWR by assuring: (1) a low probability of vacuum breaker leakage and failure through periodic surveillance, and (2) availability of drywell or wetwell sprays to condense steam that bypasses the suppression pool.

Vacuum Breaker Position Indication — Each vacuum breaker will be equipped with a position indication switch that will be sensitive enough to detect the allowable suppression pool bypass capability of the containment. Vacuum breaker position indication and associated alarms will be provided in the main control room. This reduces the potential for suppression pool bypass by assuring that the plant is not operated with a stuck open vacuum breaker, and that preexisting leakage paths will be limited to small flow areas.

Drywell Sprays — The drywell spray system is critical for mitigating the consequences of severe accidents in the ABWR. The drywell spray system serves to: (1) reduce containment overpressure and delay the time to actuation of COPS, (2) eliminate the potential for drywell failure resulting from overheating in those events in which debris may be dispersed to the upper drywell, and (3) mitigate the consequences of suppression pool bypass by condensing steam produced in the drywell. The ACIWA mode of RHR as a backup source of water to the sprays is perhaps the single-most important feature for reducing the consequences of severe accidents in the ABWR.

Lower Drywell Design — The design of the ABWR lower drywell or reactor cavity is such that there is a low probability that the cavity will be flooded at the time of reactor vessel failure, but a high probability that the cavity will be flooded after vessel failure. A dry cavity at the time of vessel failure reduces the potential for large ex-vessel steam explosions, whereas the subsequent flooding of the cavity helps minimize the impact of core concrete interactions (CCIs). The following ABWR design features are important to assuring a dry cavity at the time of vessel failure: (1) lack of any direct pathways by which water from the upper drywell (e.g., from drywell sprays) can drain to the lower drywell, other than by overflow of the suppression pool, (2) negligible probability of premature or spurious actuation of the passive flooder valves at temperatures lower than 260 °C (500 °F) or under differential pressures associated with reactor blowdown and pool hydrodynamic loads, and (3) a capability to accommodate approximately 7.2E5 kg of water in the suppression pool from external sources before the pool overflows into the lower drywell. Additional design features are included to increase the chances of a flooding a cavity following vessel failure, as discussed below.

Lower Drywell Flooder (LDF) System — The LDF system in the ABWR provides a passive means of adding water to the lower drywell following reactor vessel breach. This water would cover the core debris, thereby enhancing debris coolability, cooling the drywell, and providing fission product scrubbing. The passive flooder system is a backup to other means of lower drywell water addition in the ABWR, including: (1) continued water addition through the breached reactor vessel, (2) suppression pool overflow as a result of water addition from water sources outside containment, and (3) ingress of suppression pool water after the core debris has penetrated the wetwell-drywell connecting vents (DCVs). PRA-based sensitivity studies indicate that the incremental risk reduction offered by the passive flooder system is minimal. This is because of credit taken in the ABWR for continued water addition using the ACIWA mode of RHR.

Reactor Pedestal and Drywell Floor — The effect of CCIs is minimized in the ABWR by the use of a robust reactor pedestal and the use of basaltic concrete in the floor of the lower drywell. The reactor pedestal is 1.7 m (5.6 ft) thick and capable of withstanding 1.55 m (5 ft) of core-concrete erosion without loss of structural integrity. The use of basaltic concrete in the floor minimizes the production of noncondensable gases, thereby delaying the time of COPS actuation.

Containment Ultimate Pressure Capacity — The ultimate pressure capacity of the ABWR containment is limited by the drywell head, the failure mode of which is plastic yield of the torispherical dome. After the original SSAR submittal, GE increased the pressure capability of the drywell head from 100 pounds per square inch (psig) to 134 psig, and increased the COPS setpoint from the original value of 80 psig to the final value of 90 psig. The strengthening of the drywell head increases the ability of the containment to withstand rapid pressurization events, such as DCH, without loss of structural integrity, and provides additional margin between the COPS setpoint and the drywell failure pressure, thereby reducing the potential for drywell failure before COPS actuation.

COPS — COPS is part of the atmospheric control system in the ABWR and consists of a pair of rupture disks installed in a 10-in. diameter line, which connects the wetwell airspace to the stack. COPS provides for a scrubbed release path in the event that containment pressure cannot be maintained below 90 psig. Without this system, late containment overpressure failures would be expected to occur in the drywell, resulting in unscrubbed releases. COPS provides a significant benefit by reducing the source terms for late releases, and minimizing the potential for containment-failure-induced loss of core cooling (e.g., in Class II sequences).

Containment Inerting — The ABWR containment will be made inert during power operation. As a result, the threat of containment failure as a result of hydrogen combustion is essentially eliminated for power operation.

Containment Sump Protection — To preclude significant debris from entering the containment sump following a severe accident, a protective barrier was added around the sumps to prevent the entrance of molten[1] debris, while allowing water to enter the sumps during normal operations.

19.1.3.5.4 Impact of Severe Accident Phenomena on Containment Performance

DCH occurs only in sequences with reactor vessel failure at high pressure. For the ABWR, reactor vessel failures at high pressure constitute about 30 percent of the reactor vessel failures (about 4.3E-8 per year). In view of the large uncertainties inherent in estimating the pressure loads associated with DCH, the staff conservatively based its findings on the results of a GE sensitivity analysis for DCH which reflected an increased contribution to pressurization due to higher baseline containment pressure, and combustion/recombination of hydrogen with residual oxygen. As such, DCH is the principal contributor to containment failure for the ABWR. The staff has estimated the containment failure probability for DCH to be about 5 percent, conditional upon reactor vessel failure at high pressure. This results in a very low frequency of containment failure from DCH (2.3E-9 per year). The low frequency of reactor vessel failure at high pressure is a result of the highly reliable depressurization system. There are no specific ABWR containment design features to deal with DCH loads other than the general arrangement of the drywell, wetwell, and connecting vents, which provides for a series of 90-degree bends that debris must traverse in order to reach the upper drywell.

Fuel-coolant interactions (FCI) or steam explosions are considered negligible in the ABWR design because of the very low probability that the lower drywell will be flooded at the time of reactor vessel failure (0.3 percent). In addition, the ABWR reactor pedestal is capable of withstanding the best-estimate loads associated with an ex-vessel steam explosion as predicted by GE and staff calculations. The design features that contribute to the low probability of a flooded lower drywell at the time of vessel failure were discussed above in the context of the lower drywell design. The structural capability of the reactor pedestal is discussed further in Section 19.2 of this report.

CCIs have a minimal impact on ABWR containment structural integrity because of the inclusion of: (1) a thick reactor pedestal, (2) the use of basaltic concrete in the floor of the lower drywell, and (3) a sump shield to prevent core debris from entering the lower drywell sump. These features provide significant confidence in reactor pedestal and containment integrity for well beyond 24 hours following reactor vessel failure and render CCI-induced containment failure a relatively insignificant contributor to risk.

Hydrogen combustion is not an important containment challenge in the ABWR since the atmosphere is made inert during normal operation.

19.1.3.6 Level 3 Analysis Insights

19.1.3.6.1 Risk to the Public

Based on the Level 3 PRA, the estimated total risk to the public from the ABWR is extremely small. GE's analysis indicates a total dose of about 0.2 person-rem over the 60-year period. The staff's estimate is about 1 person-rem. The difference is largely a result of the contribution from unisolated LOCAs outside containment, and an increased probability of early containment failure from DCH. It should be noted that while vented scenarios are the dominant contributor to the staff's estimate of containment failure using the dose definition of containment failure, these sequences do not contribute significantly to total risk when measured in terms of person-rem exposure. Rather, total risk is dominated by events that lead to early containment structural failure and containment bypass. This is consistent with results from PRAs for operating plants.

19.1.3.6.2 Leading Contributors to Risk

Despite their small contribution to total core damage frequency (0.8 percent based on the staff's assessment), unisolated LOCAs outside containment dominate the ABWR risk profile. The frequency of these sequences is extremely low in absolute terms (1.2E-9 per year). In the staff's assessment, these sequences dominate risk because of the severe releases associated with complete bypass of containment. These sequences also dominate risk because of the fact that the baseline core damage frequency and the contribution of more familiar sequences and containment challenges that dominate PRAs for operating plants have been reduced in the ABWR design. The fact that these sequences dominate risk is, in a way, a reflection of the low estimated risk of the ABWR.

Other early containment failures (DCH and ATWS) are the second most dominant contributor to risk for the ABWR. Although the frequency of these early failure mechanisms (2.5E-9) is higher than unisolated LOCAs outside containment, fission product releases are reduced somewhat because of holdup in containment.

It should be noted that the reported frequency of unisolated LOCAs reported in Amendment 32 of the SSAR is about a factor of 10 lower than the above value on which the staff based its finding. This would result in the total risk's contribution, as well as the contribution of bypass scenarios to risk, being lower than described above.

19.1.3.7 PRA-Based Input to the Certified Design

19.1.3.7.1 Reliability Assurance Program

The ABWR is the lead plant for development of a RAP for advanced reactors as required by 10 CFR Part 52. GE made a particularly strong effort in identifying important SSC for inclusion in RAP based on insights from the ABWR PRA. In Appendix 19K of the SSAR, GE has listed the reliability and maintenance actions that it believes should be considered throughout the life of the plant so that the PRA remains an adequate basis for quantifying plant safety and determining safety insights. It is anticipated that a COL applicant will make these insights (given in Table 19K.11-1 in the SSAR) the cornerstone of its programs (DRAP and ORAP).

Internal and external event sequences, both for Levels 1 and 2, were considered in drawing up the list of reliability and maintenance actions. In developing the list, GE performed a systematic search that involved both quantitative and qualitative considerations. For Level 1 internal events, the key considerations were the results of risk achievement worth and Fussell-Vesely importance measures. These results are reported in Tables 19K.3-1 and 19K.3-2 in the SSAR and Appendix K in this report. From risk achievement worth measures, one gets a list of SSCs where maintenance or testing resources should be focused to help assure high levels of availability. A Fussell-Vesely importance measure answers the question, "For which SSCs would improvement of current unavailabilities be most beneficial in order to best lower the estimated core damage frequency?"

19.1.3.7.2 Tiers 1 and 2 Information or Requirements

In its review, the staff believed that it was important to systematically search a PRA for safety insights. As part of this search for insights, GE identified important safety insights that need to be passed on to a COL applicant. Appendix K of this report provides a cross-reference between the PRA insights and the ITAAC, Tier 1 design descriptions, and Tier 2 material in the DCD. Appendix K explains the disposition of these ABWR PRA safety insights. The staff has reviewed the COL action items proposed by GE and finds them to be appropriate dispositions of these insights, and finds they are acceptable.

19.1.3.8 PRA Insight Conclusions and Safety Findings

This section documents the overall conclusions about the insights drawn from the ABWR PRA about the ABWR design.

19.1.3.8.1 Vulnerabilities

In its performance of the ABWR PRA, GE did not specifically identify any design features or procedures that constituted a vulnerability to severe accidents. GE did search for and identify many cost-effective design and procedure improvements that it included in the ABWR design. Details of the design improvements motivated by the ABWR PRA are discussed previously in Section 19.1.3 and listed in Table 19.1-4. The search by the staff and GE for design and procedure vulnerabilities (or areas of potential improvement) included internal and external events (seismic, fire, and internal floods) as well as events in Modes 3, 4, and 5.

19.1.3.8.2 Balance of Prevention and Mitigation

The staff examined the ABWR design to determine if the design has an inappropriate reliance on either prevention of severe accidents or mitigation of severe accidents to the exclusion or detriment of the other. The staff finds the balance to be acceptable for the following reasons:

In the area of prevention of core damage, GE has reported that the ABWR PRA estimates internal event core damage frequency to be in the range of 1E-7 per year. This estimate reflects the design modifications incorporated into the ABWR design that have eliminated or reduced most of the important sequences that have led to core damage in other BWR PRAs. The NRC does not endorse the absolute value of the core damage frequency reported by GE, but does endorse the insights drawn from the PRA by GE and the NRC, which are based on relative and other considerations. For the external events seismic, fire, and internal floods, the ABWR design is an improvement over those of existing operating plants. The seismic design is built to an SSE of 0.3 g while most sites in the United States east of the Rocky Mountains would have lower SSEs when determined on a site-specific basis. In addition, the ABWR PRA-based seismic margins analysis demonstrated that the design has significant margin to earthquakes well above the SSE. For fires and floods, the ABWR design has three physically separated safety divisions that limit the chances of a fire or flood affecting more than one division. This capability is a design enhancement for advanced reactors. For mitigation of a severe accident in which core damage has occurred, the ability of the containment design to withstand severe accidents is at least as robust as that of operating BWRs today and has some additional features not in current designs. Additional features include the ac-independent water addition system, an increased drywell head ultimate pressure capacity, and the passive flooder system.

19.1.3.8.3 Comparison With NRC and EPRI Safety Goals

In the DSER (SECY-91-309), the staff compared the integrated risk results for the ABWR with the Commission's quantitative health objectives and safety goals, NRC and EPRI requirements or goals for advanced light-water reactors (ALWRs), and GE's design goal. GE and staff estimates of various risk measures based on the original GE analysis (Amendment 8 to the SSAR), are reported in Figures 19.1-1 and 19.1-2 of the DSER. Based on these comparisons, the staff reached tentative conclusions on how the various objectives and goals were met. After 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 itself to reporting the results of its review of the integrated risk estimates in this report.

The staff's evaluation of GE's updated PRA analyses is presented in the preceding portions of Section 19.1 of this report. Based on a review of the updated PRA analysis, the staff performed a limited update of the comparisons with the safety goals. The updated comparisons with safety goals are not as exhaustive as the one provided in the DSER since: (1) the staff did not attempt to requantify uncertainties in core damage frequency and containment performance as part of the final evaluation, and (2) only doses at the site boundary were recalculated as part of the final evaluation, rather than the full set of offsite consequence measures. However, for certain performance measures, such as individual risk of early fatality, the original comparisons indicated significant margins between the PRA results and the health objectives or safety goals, even when the uncertainty associated with various phenomenological issues and other issues raised in the staff review were factored into the assessment. In those cases, the staff's tentative conclusions presented in the DSER are still considered valid for the updated ABWR PRA and have been used as the basis for the staff's final conclusions. Limited reliance on the original analyses is considered acceptable on the basis that the updated core damage frequency and containment release characteristics are not significantly different from those on which this original assessment was based. Furthermore, the analyses and plant modifications performed after Amendment 8 to the SSAR did not identify any plant vulnerabilities and are expected to further reduce risk.

The comparison of the ABWR PRA results with the various health objectives and safety goals follows. As discussed in Section 19.1.1 of this report, the staff does not believe that it is appropriate, or possible, to directly compare the results of the updated ABWR external events analyses with the safety goals because in the updated PRA these events were analyzed using either a margins approach (seismic) or bounding analyses (internal floods and fire). Accordingly, the comparisons with safety goals are limited to internal events.

<u>Individual risk of early fatality — Goal: $\leq$ 5E-7</u>

The staff concluded in the DSER that the ABWR meets the Commission's quantitative health objective for individual risk of early fatality by a wide margin. The calculated mean estimate, as well as the upper bound of the uncertainty distribution (95th percentile), were orders of magnitude below the goal. Because the updated core damage frequency and containment release characteristics are not significantly different from those on which this original assessment was based, the staff concludes that the final ABWR design would also meet this health objective by a wide margin, even if the effects of unisolated LOCAs outside containment are taken into account. The staff further believes that the remaining margin provides reasonable assurance that consideration of the impact of those portions of the analysis for which full PRA quantification was not performed (e.g., seismic events) will not result in the mean value's exceeding the goal.

Individual risk of cancer fatality — Goal: < 2E-6

The staff concluded in the DSER that the ABWR meets the Commission's quantitative health objective for individual risk of cancer fatality by a wide margin. The calculated mean estimate, as well as the upper bound of the uncertainty distribution, were several orders of magnitude below the goal. Because the updated ABWR core damage frequency and containment release characteristics are not significantly different from those on which this original assessment was based, even if the effects of unisolated LOCAs outside containment are taken into account, the staff concludes that the final ABWR design would also meet this health objective by a wide margin. The staff further believes that the significant margin between the calculated risk and this goal provides reasonable assurance that consideration of the impact of those portions of the analysis for which full PRA quantification was not performed will not result in the mean value's exceeding the goal.

Core damage frequency — Commission Goal: < 1E-4
EPRI Goal: < 1E-5

The staff concluded in the DSER that the ABWR meets both the Commission and the EPRI goals for internal event core damage frequency by a wide margin. The calculated mean value was below the goal, with the upper bound of the uncertainty distribution about 1 order of magnitude below the Commission goal. Based on the updated PRA and a supporting Level 1 uncertainty analysis, GE has estimated the ABWR mean core damage frequency at 1.6E-7 per year, with an upper bound (95th percentile) value of 4.5E-7 per year. Based on review of the ABWR PRA, the staff concludes that the final ABWR design meets the Commission's goal for internal events, even if uncertainties in the calculated core damage frequency are taken into account. The analyses performed by GE for ABWR external and shutdown events do not lend themselves to comparison with the Commission's core damage frequency goals, based on methods and assumptions used. The staff determined that this is acceptable, since the GE analyses of external events and shutdown events included an appropriate search for vulnerabilities.

Probability of large release (one or more early fatalities) — Goal: < 1E-6

The staff concluded in the DSER that the ABWR meets the Commission's goal for the probability of large release by a considerable margin, recognizing that the definition of "large" was still under Commission consideration. The calculated mean estimate was well below the goal, with the upper bound of the uncertainty distribution (95th percentile) about 2 orders of magnitude below the goal. Because the updated core damage frequency and containment release characteristics are not significantly different from those on which this original assessment was based, the staff concludes that the final ABWR design would also meet this objective by a wide margin. The staff notes that this goal can be met as a result of the very low calculated core damage frequency for internal events, even without taking credit for the containment. Compliance with the Commission's containment performance goal provides added assurance that the probability of a large release will remain below the goal.

Conditional containment failure probability (structural integrity definition) — Goal: < 0.1

The staff concluded in the DSER that the ABWR did not strictly meet the Commission's goal for CCFP using the structural integrity definition of containment failure. The staff's uncertainty distribution bridged the goal with a median value slightly above the goal, suggesting that the bulk of the distribution was slightly above the goal.

GE's updated point estimate of CCFP based on structural integrity definition is 0.005. The staff adjusted this value to account for: (1) uncertainties in DCH, and (2) treatment of unisolated LOCAs outside containment. These adjustments increased CCFP to about 0.026, which is still below the Commission goal. However, the staff did not attempt to requantify the uncertainties associated with this estimate. It believes that if these uncertainties were requantified reflecting the final ABWR design and additional severe accident analyses completed since the DSER (e.g., increased containment pressure capacity and additional information supporting a low probability of a flooded reactor cavity at the time of vessel failure), the resulting CCFP estimate and associated uncertainties would be somewhat lower than reported in the DSER, but would still be significant.

<u>Conditional containment failure probability (dose definition) — Goal: < 0.1</u>

The staff concluded in the DSER that the ABWR did not strictly meet the Commission's goal for CCFP using GE's definition of containment failure (dose). The staff's uncertainty distribution bridged the goal, with a mean estimate above the goal and median estimate slightly below the goal. This indicated that about half of the distribution is below the goal.

GE's updated point estimate for CCFP (based on the dose definition) is 0.002. The CCFP based on the dose definition is not significantly different than the CCFP based on the structural integrity definition (where containment venting is not considered as a containment failure). The reason for this is that in GE's analysis, doses in excess of 25 rem at 0.8 km (0.5 miles) primarily occur only when structural integrity is breached. In contrast, the staff's consequence calculations indicate a high probability (about 60 percent) of exceeding 25 rem at the boundary for vented scenarios. Thus, the staff's estimate of containment failure frequency using the dose definition of containment failure includes essentially all of the frequency of events leading to loss of structural integrity, plus about 60 percent of the frequency of sequences with COPS actuation. This results in a staff estimate of CCFP of about 0.10, which still meets the Commission's safety goal.

The staff did not attempt to requantify the uncertainties associated with this estimate. It expects that if these uncertainties were requantified reflecting the final ABWR design, additional severe accident analyses completed since the DSER, and more deterministic estimates of source terms for vented scenarios, the resulting CCFP estimate and associated uncertainties would be lower than reported in the DSER. Nevertheless, the uncertainties would still be significant and would extend above the goal. Although the uncertainty bands are different, the staff and GE both concludes that the mean values of their risk estimates meet the Commission's safety goal.

19.1.3.8.4 Comparison With Operating BWRs

The staff evaluated the ABWR design and its improved or unique features to prevent or mitigate severe accidents such as the three full (high and low pressure) ECCS divisions, the COPS, and the passive flooder. The staff evaluated the ABWR PRA and concluded that the absolute value of the estimated core damage frequency for the ABWR is lower than those of operating BWRs in the United States. Based on these reviews, the staff finds that the ABWR design represents an improvement in safety over operating BWRs in the United States.

19.1.3.8.5 ABWR Design Acceptability

Based on the staff's review of the ABWR PRA and ABWR design as set forth in this section (19.1) of this report, the staff finds that the ABWR design and the submittals made for the ABWR in the SSAR meet the intent of the Commission's Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants, dated August 8, 1985, the requirement of 10 CFR 50.34(f)(1)(i) to perform a plant-specific PRA that seeks improvement in the reliability of core and containment heat removal systems, the staff's proposed applicable regulation for analysis of external events for the ABWR PRA, and the requirement of 10 CFR 52.47(a)(1)(v) for an evolutionary plant design vendor to submit a PRA.

19.1.3.9 ACRS Concerns Related to the ABWR PRA

19.1.3.9.1 Adequacy of the ABWR PRA

The Advisory Committee on Reactor Safeguards (ACRS) asked the staff to explain how it intends to use the ABWR PRA in the design certification process and when PRA guidance will be provided.

In preparing for design certification, the staff and GE used the ABWR PRA in a number of traditional as well as new areas. The staff expects that the PRA will continue to prove to be useful throughout the entire process — from design certification through the end of the life of an operating ABWR. Examples of traditional areas in which the ABWR PRA was used include estimating core damage frequencies and risk significance and identifying design vulnerabilities. Innovative ways in which the PRA has been used include the following:

- helping to identify systems and components to be included in the RAP, increasing the safety of the design of the ABWR by examining design options
- helping to identify human errors that need to be considered when designing an advanced control room
- identifying areas that should receive special attention under ITAAC
- helping to determine the balance of prevention and mitigation capabilities
- providing a structure for helping to determine procedural and TS needs during modes other than full power.

19.1.3.9.2 Adequacy of GE's Treatment of the Reactor Water Cleanup System

The ACRS performed a review, independent from the staff, of the reactor water cleanup (RWCU) system. GE's treatment of this system in the PRA was not reviewed in detail as part of the staff's initial review. Several of the questions raised by the ACRS review applied to the ABWR PRA. First, the ACRS review concluded that the ABWR PRA did not evaluate RWCU system line breaks as initiators. Second, the ACRS review concluded that GE erroneously took credit for the RWCU system as a potential heat removal path at high pressure during non-LOCA transients. The ACRS found credit to be inappropriate, in part, because the RWCU system was designed to isolate on high temperature (to protect resin beds) and because even if the resin beds were bypassed, the outlet temperatures of the heat exchanger piping would have exceed the design limits of the piping.

The ACRS correctly noted that GE did not treat RWCU system line breaks, or other LOCAs outside the primary containment, as initiating events in the PRA. Rather, the approach taken by GE (SSAR Section 19E.2.3.3) was to show that the risks associated with such events constitute a small fraction of the risk from all non-bypass paths, and that these ex-containment LOCAs therefore need not be included in the PRA. The ACRS questioned the validity of this analysis, and the fact that it failed to consider that the RWCU system line break could be the initiator for the core damage event. Based on concerns identified by the ACRS, the staff performed a more detailed review of GE's treatment of the RWCU system in the PRA. After further review of GE's treatment of ex-containment LOCAs, the staff identified additional concerns related to flow area assumptions on which the split fractions were based and isolation valve failure probability assumptions in the analysis (i.e., whether the assigned values adequately account for environmental effects and common cause failure).

In response to ACRS and staff concerns, GE revised its suppression pool bypass analysis and provided additional clarification regarding the effects of a break in the RWCU suction line. GE indicated that the system arrangement and emergency procedure guidelines (EPGs) provide assurance that unisolated breaks in the RWCU suction line will not result in core uncovery and long-term releases. The staff required that GE develop a COL Action Item that would have a COL applicant develop postaccident recovery procedures for an unisolated CUW line break.

As part of its review, the staff required GE to provide an analysis of LOCAs outside containment that was based on event trees and fault trees rather than on split fraction arguments.

The ACRS correctly stated that GE erroneously took credit for the RWCU system at high pressure during transients. GE has corrected this design deficiency 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, the RWCU will only be put into operation by emergency procedure after the RHR fails. Thereafter, 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 calculates that this temperature increase is only a few degrees above the design temperature and argues that this is acceptable because the RWCU is a backup system that only will have to be used in this configuration for very low probability, beyond-design-basis events.

19.1.4 Evaluation of the Quality of the ABWR Probabilistic Risk Assessment

The staff has completed its review of the quality of the ABWR PRA. It finds that the ABWR PRA is of sufficient quality that, at a minimum, it can be used in the following ways:

(1) to assess (within the limits of PRA methods and uncertainties) the risks associated with the ABWR design

(2) to identify strengths and weaknesses of ABWR design features

(3) to evaluate ABWR containment failure probabilities for early and late failure modes

(4) to compare the ABWR risk results with the Commission's safety goal and the "safety margin basis design requirements" provided in the EPRI ALWR Requirements Document within the limitations of the latest technology in risk assessment

(5) to provide an integrated perspective of the overall risk estimates for the design

(6) to identify major contributors to uncertainty in estimated core damage frequency.

The staff concludes that the quality of the ABWR PRA is adequate for its intended functions such as supporting and improving the ABWR design process, providing relative importance of sequences (as well as identifying important SSC) leading to core damage or containment failure, and searching for design and procedure vulnerabilities that could be eliminated on a cost-effective basis.

19.1.5 Open Item Closure

All Open and "Outstanding" Items identified in Chapter 19 of the staff's ABWR DSER and DFSER are resolved satisfactorily. The staff is issuing a letter separate from this report documenting the resolution of these issues. All important issues related to containment performance are discussed in Section 19.2.

19.2 Severe Accident Performance

19.2.1 Introduction

The purpose of this section is (1) to describe the NRC's approach to resolution of severe accident issues for evolutionary light water reactors as specified in SECY-90-016, SECY-91-262, SECY-93-087, and the corresponding SRMs and (2) to evaluate the approach proposed by GE for resolution of severe accident issues for the ABWR design.

To provide adequate protection of the public health and safety, current NRC regulations require conservatism in design, construction, testing, operation, and maintenance of nuclear power plants. A defense-in-depth approach has been mandated in order to prevent accidents from happening and, if accidents should occur, to mitigate their consequences. Siting of nuclear power plants in less populated areas is emphasized. Furthermore, the NRC, State, and local governments mandate emergency response capabilities that provide additional defense-in-depth protection to the surrounding population.

The reactor and containment systems design provides a vital link in the defense-in-depth philosophy. Current reactors and containments are designed to withstand a LOCA and to meet the siting criteria of 10 CFR Part 100 and General Design Criteria (GDC) of 10 CFR Part 50, Appendix A. The large-break LOCA and other accidents analyzed in accordance with the NRC's Standard Review Plan (NUREG-0800) and documented in Chapter 15 of the SSAR are commonly referred to as "design-basis accidents" (DBAs) for nuclear power plants. This high level of confidence in a defense-in-depth approach results, in part, from stringent requirements for meeting single-failure criterion, redundancy, diversity, quality assurance, and utilization of conservative models.

The NRC also has requirements to mitigate adverse conditions associated with transients or events considered outside the design basis such as ATWS (10 CFR 50.62), station blackout (SBO) (10 CFR 50.63), and combustible gas control (10 CFR 50.44); however, a definitive set of regulatory requirements for addressing specific severe accident phenomenon does not exist. Existing regulations that require conservative analyses and inclusion of mitigative features for design-basis events, provide margin for severe accident challenges. In addition, the staff, in keeping with the Commission's Policy Statement on Severe Accidents that future designs for nuclear power plants achieve a higher standard of severe accident safety performance, concluded that severe accidents should be considered in the design of future nuclear power plants.

In an SRM, dated January 28, 1992, on SECY-91-262, the Commission approved the staff's recommendation to proceed with design-specific rulemakings through individual design certifications to resolve selected technical and severe accident issues. The effect of these actions on the ABWR is that the criteria specified for resolution of severe accident issues in SECY-90-016 and SECY-93-087 will be incorporated into the ABWR design certification rulemaking as applicable regulations. The following discussion describes the criteria that were used for the deterministic evaluation of severe accident issues.

19.2.2 Deterministic Assessment of Severe Accident Prevention

19.2.2.1 Severe Accident Preventative Features

Accident initiators can be separated into two general groups: transients and LOCAs. Transients include planned reactor shutdowns and transients that result in reactor scrams. Examples of transients include manual shutdown, main steam isolation valve closure, loss of condenser vacuum, loss of feedwater, nonisolation event (trip with bypass), inadvertent open relief valve, and loss of offsite power. LOCAs generally fall within three categories, small, medium, and large, based on the size of the line break.

Following the accident initiator, plant systems respond to control reactivity, reactor pressure, reactor water level, and containment parameters within the design basis spectrum. Ensuring sufficient heat removal from the core to prevent overheating and subsequent fuel damage is of paramount importance. Failure to provide this heat removal can result in fuel overheating and the potential for oxidation and melting of the reactor core.

Transient-induced accidents are usually accompanied by actuation of the safety or relief valves that transfer steam from the reactor directly to the suppression pool, thus limiting an increase in containment pressure. LOCAs blow down into the drywell atmosphere until the suppression pool horizontal vents [i] are uncovered, resulting in containment pressurization lower than the containment design pressure. In the event that the steam from the reactor bypasses the suppression pool and travels directly to the wetwell atmosphere, significant pressurization of the containment may result. The issue of suppression pool bypass is addressed in Sections 6.2.1.8 of this report for design-basis accident (DBA) bypass and 19.2.3.3.5 for severe accident bypass.

In response to accident initiators identified through operating reactor experience and analyses of the results of probabilistic safety assessment, the staff developed criteria for evolutionary light-water reactors (LWRs) to prevent the occurrence of such initiators from leading to a severe accident. These criteria were specified in SECY-90-016 and SECY-93-087 and include design provisions for anticipated transient without scram (ATWS), SBO, fires, and intersystem LOCAs.

19.2.2.1.1 Anticipated Transient Without Scram

An ATWS is an anticipated operational occurrence followed by the failure of the trip portion of the reactor protection system (RPS). Anticipated operational occurrences (transients) are those conditions of normal operation that are expected to occur one or more times during the life of the nuclear power plant and include, but are not limited to, loss of power to all recirculation pumps, tripping of the turbine generator set, isolation of the main condenser, and loss of all offsite power. Dependent upon the transient and its severity, the plant may recover and continue normal operation or the plant may automatically shut down (scram) via the RPS. The RPS is designed to safely shutdown the reactor to prevent core damage.

These transients when coupled with a failure of the RPS may lead to conditions beyond the design basis of the plant. In these cases, the reactor must be manually scrammed in order to avoid reactor fuel damage or coolant system damage. Subsequent failure of the manual scram system and inadequate core cooling may lead to core damage.

Transients with the greatest potential for significant damage to the reactor core and containment are those leading to an increase in reactor pressure and temperature, a loss of heat sink, or a failure of the RPS to scram the reactor. During an ATWS event, reactor power, pressure, and temperature must be controlled or the potential exists for a severe accident.

The ATWS rule (10 CFR 50.62) was promulgated to reduce the probability of an ATWS event and to enhance mitigation capability if such an event occurred. For BWRs, the ATWS rule specifies inclusion of an alternate rod insertion system, an SLCS, and equipment to trip the reactor recirculation pumps. In Section 15.5 of this report, the NRC concluded that the ABWR complies with the ATWS rule.

19.2.2.1.1.1 Preventive and/or Mitigative Features

In SECY-90-016, the staff recommended that the Commission approve its position that diverse scram systems should be provided for evolutionary LWRs. In its June 26, 1990 SRM, the Commission approved the staff's position, but directed that if the applicant can demonstrate that the consequences of an ATWS are acceptable, the staff should accept the demonstration as an alternative to the diverse scram system.

The ABWR has a number of design features that reduce the risk from an ATWS event including a diverse scram system with both hydraulic and electric run-in capabilities for the fine motion control rod drives (FMCRD), an automatic SLCS, and a reactor internal pump (RIP) trip capability. In addition, the scram discharge volume has been removed from the ABWR, eliminating some of the potential problems that could affect the scram function associated with older BWR designs.

The ABWR has an alternate rod insertion system that is diverse, when compared with that of the RPS, from sensor output to the final actuation device. The ARI system has redundant scram air header exhaust valves and is designed to perform its function in a reliable manner.

19.2.2.1.1.2 Basis for Acceptability

In SECY-90-016, the NRC concluded that evolutionary LWR designs should provide diverse methods of inserting control rods to mitigate a potential ATWS and to ensure a safe reactor shutdown. The ABWR incorporates a diverse method for inserting control rods. The ABWR complies with the ATWS rule, as concluded in Section 15.5 of this report, and the design is capable of satisfactorily mitigating the effects of an ATWS and preventing an ATWS event from evolving into a severe accident with core damage. The staff concludes that the ABWR meets the criteria specified in SECY-90-016 through incorporation of the features discussed above.

19.2.2.1.2 Station Blackout

An SBO involves the complete loss of ac electrical power to the essential and nonessential switchgear busses in a nuclear power plant (i.e., loss of offsite electric power system concurrent with turbine trip and unavailability of the on-site emergency ac power system). SBO does not include the loss of available ac power to buses fed by station batteries through inverters or by alternate ac sources, nor does it assume a concurrent single failure or DBA.

During normal plant operation, power is supplied to the Class 1E distribution system from the main generator. Following plant shutdown, the preferred power source is the offsite grid, which provides a continuous source of ac electric power to equipment required to maintain core coolability. If the power from the offsite grid is not available, the on-site distribution system will sense an undervoltage condition and initiate a transfer to the emergency diesel generators (EDGs) for continued power. In the event of the loss of both the offsite grid and EDGs, an SBO has occurred. As most DHR and containment heat removal systems are dependent upon ac power for operation, failure to provide core cooling during an SBO will likely result in core temperature and pressure increases and may lead to a severe accident.

The SBO rule (10 CFR 50.63) allows several design alternatives to ensure that a plant is able to withstand an SBO for a specified duration and recover. A complete evaluation of the ABWR relative to the SBO rule is provided in Section 8.3.9 of this report.

19.2.2.1.2.1 Preventive and/or Mitigative Features

In SECY-90-016, the staff stated that the preferred method of demonstrating compliance with 10 CFR 50.63 is through the installation of a spare (full- capacity) alternate ac power source of diverse design that is consistent with the guidance in Regulatory Guide (RG) 1.155 and is capable of powering at least one complete set of normal shutdown loads. The staff recommended that the Commission approve the requirement for an alternate ac source for evolutionary LWRs. In its June 26, 1990 SRM, the Commission approved the staff's position. Therefore, the staff's proposed applicable regulation for an alternate ac souce is as follows:

> The standard design must provide an alternate ac power souce for the purposes of dealing with station blackout.

The ABWR design includes three independent electrical divisions, each capable of providing power to a high-pressure and low-pressure water injection division, each powered by a full-capacity EDG, and each division capable of independently shutting down the reactor. In addition, the ABWR design includes an alternate ac combustion turbine to back up the diesel generators. The RCIC system (with its supporting systems) is designed to perform its function without ac power for at least 2 hours and also have an ultimate capability to function for 8 hours without ac power. Extended blackout capabilities are also provided by the ACIWA system. This system allows for makeup to the reactor vessel following depressurization or to the containment sprays from a direct-drive diesel fire pump or by connecting an external pumping source, such as a fire truck, to a yard standpipe into the RHR system.

19.2.2.1.2.2 Basis for Acceptability

In SECY-90-016, the NRC concluded that designers should meet the SBO rule by including an alternate ac power source (i.e., CTG) of diverse design capable of powering at least one complete set of normal shutdown loads. To cope with SBOs, the ABWR has included an alternate CTG. Based on the preventive and mitigative features described above, the staff concludes that the ABWR has met the criteria of SECY-90-016 and the staff's proposed applicable regulation for station blackout.

19.2.2.1.3 Fire Protection

The Commission concluded that fire protection issues that have been raised through operating experience and the external events program must be resolved for evolutionary light water reactors. In SECY-90-016, the staff recommended that current NRC guidance to resolve fire protection issues be enhanced to minimize fire as a significant contributor to the likelihood of severe accidents and DBAs. As indicated in SECY-90-016, the ABWR design must ensure that safe shutdown can be achieved, assuming that all equipment in any area will be rendered inoperable by fire and that reentry into the fire area for repairs and operator actions will be impossible. Because of its physical configuration, the control room is excluded from this approach, provided an independent alternative shutdown capability that is physically and electrically independent of the control room is included in the design. The ABWR design must also provide fire protection for redundant shutdown systems in the reactor containment building that will ensure, to the extent practical, that one shutdown division will be free of fire damage. Additionally, the ABWR design must ensure that smoke, hot gases, or fire suppressant will not migrate into other fire areas to the extent that they could adversely affect safe-shutdown capabilities, including operator actions. These fire protection measures are for both DBAs and severe accidents.

19.2.2.1.3.1 Preventive and/or Mitigative Features

Section 9.5.1 of this report provides a description and evaluation of the ABWR features provided to prevent and mitigate fires. In particular, this section addresses protection of safe-shutdown equipment, passive fire protection features, fire detection, fire protection water supply system, water fire suppression systems, gaseous fire suppression systems, fire extinguishers, emergency communication and lighting, emergency breathing air, curbs and drains, smoke control, access and routes, construction materials and combustible contents, and interaction with other systems.

19.2.2.1.3.2 Basis for Acceptability

Based on the evaluation in Section 9.5.1 of this report and on the discussion above, the staff concludes that the ABWR design meets the criteria identified in SECY-90-016 and is acceptable for preventing and mitigating threats from fires for DBAs and severe accidents.

19.2.2.1.4 Intersystem Loss-of-Coolant Accident

Intersystem loss-of-coolant accidents (ISLOCAs) are defined as a class of LOCAs in which the RCS pressure boundary is breached and coolant is lost through an interfacing system with a lower design pressure. The breach may occur in portions of piping located outside the primary containment, causing a direct and potentially unisolable discharge from the RCS to the environment. An ISLOCA is of concern because of potential direct releases to the environment, loss of core cooling, and loss of core makeup.

High or low pressure interfaces occur on many lines including low-pressure injection lines and the RHR heat exchangers. An ISLOCA occurs when high pressure is introduced in a low-pressure system because of valve failure or an inadvertent valve actuation. In either case, the overpressurization can cause the low-pressure system or components to fail. An ISLOCA concurrent with a loss of all core cooling may lead to core damage.

19.2.2.1.4.1 Preventive and/or Mitigative Features

In SECY-90-016, the staff recommended that evolutionary LWR designs reduce the possibility of a LOCA outside containment by designing (to the extent practicable) all systems and subsystems connected to the RCS to an ultimate rupture strength (URS) at least equal to the full RCS pressure. The "extent practicable" phrase shows a realization that all systems must eventually interface with atmospheric pressure and that for certain large tanks and heat exchangers, it would be difficult or prohibitively expensive to design such systems to a URS equal to full RCS pressure. The staff further recommended that systems that have not been designed to withstand full RCS pressure should include (1) the capability for leak testing of the pressure isolation valves, (2) indication in the control room of valve position when isolation valve operators are deenergized, and (3) high-pressure alarms to warn control room operators when rising RCS pressure approaches the design pressure of attached low-pressure systems and both isolation valves are not closed.

In its June 26, 1990 SRM, the Commission approved the staff's position on ISLOCA provided that all elements of the low-pressure system are considered (e.g., instrument lines, pump seals, heat exchanger tubes, and valve bonnets).

The structural capability of low-pressure piping systems interfacing with the reactor coolant pressure boundary to withstand the consequences of an ISLOCA is discussed in Section 3.9.3.1.1 of this report. In addition, GE performed a systematic evaluation of interfacing systems to ensure that the SECY-90-016 requirements were satisfied. The resolution of this issue is provided in Section 20.2.19 of this report.

19.2.2.1.4.2 Basis for Acceptability

As indicated in Chapter 20 and Section 3.9.3.1.1 of this report, the staff concludes that GE has met the criteria from SECY-90-016, as approved by the Commission, regarding ISLOCA prevention and mitigation for the ABWR.

19.2.3 Deterministic Assessment of Severe Accident Mitigation

19.2.3.1 Overview of the ABWR Containment Design

The ABWR containment maintains the pressure suppression design of other BWR containments. The containment is a reinforced concrete cylindrical structure with a steel upper drywell head and internal steel liner to reduce leakage. The ABWR containment atmosphere is made inert to preclude hydrogen combustion and a slightly positive pressure is maintained to prevent air from leaking in. The containment basemat is 5.5 m (18 ft) thick with several layers of reinforcements. The top slab is an integral part of the fuel pool. The containment wall is a right circular cylinder 2 m (6.56 ft) thick with an inside radius of 14.5 m (47.57 ft) and height of 29.5 m (96.78 ft). Its internal space is divided by the diaphragm floor and the reactor pedestal into an upper drywell chamber, a lower drywell chamber, and a wetwell. The upper drywell volume surrounds the RPV and houses the steam and feedwater lines and other connections of the reactor primary coolant system and safety/relief valves, and the lower drywell volume houses the reactor internal pumps and FMCRD.

The cylindrical RPV pedestal separates the lower drywell from the wetwell. Ten DCVs are built into the RPV pedestal and connect the upper drywell and lower drywell. The DCVs are extended downward by steel pipes, each of which has three horizontal vent outlets into the suppression pool. The wetwell consists of an air volume and suppression pool. Steam from a reactor vessel blowdown or from a break in a major pipe inside the drywell condenses in the suppression pool through the SRVs or the DCVs. A vacuum breaker system is provided between the drywell and wetwell to prevent excessive differential pressures.

The containment heat-removal system is an integral part of the RHR system. Suppression pool temperature is controlled in the suppression pool cooling mode of RHR. RHR also includes the containment spray feature to cool the containment air space. A capability also exists to directly connect the drywell spray header to the fire protection system pumps, one of which is driven by a diesel engine.

The ABWR design includes a passive flooder system. This system consists of a group of pipes that horizontally passes through the pedestal wall and connects the suppression pool to the lower drywell. These pipes terminate in the lower drywell with fusible plugs. These plugs are designed to melt and open a connection between the suppression pool and the lower drywell region when the drywell airspace temperature reaches 260 °C (500 °F).

A COPS consisting of two relief rupture disks in succession from the wetwell airspace is provided. The inner rupture disk, which controls the actual pressure, actuates at a pressure of 617.8 kPa gage (90 psig). This is above the containment design basis pressure of 309.9 kPa gage (45 psig), but below the ASME Service Level C limit of 666.9 kPa gage (97 psig).

19.2.3.2 Severe Accident Progression

This section provides a description of the processes, both physical and chemical, that may occur during the progression of a severe accident and how these phenomena affect containment performance. This description is intended to be generic in nature. However, many aspects of severe accident phenomena depend on the specific reactor type or on the containment design features. This information has been extracted from NUREG/CR-5132 Severe Accident Insights Report, NUREG/CR-5597 In-Vessel Zircaloy Oxidation/Hydrogen Generation Behavior During Severe Accidents, and NUREG/CR-5564 Core-Concrete Interactions Using Molten UO_2 With Zirconium on a Basaltic Basemat.

Severe accident progression can be divided into two phases: an in-vessel stage and an ex-vessel stage. The in-vessel stage generally begins with insufficient DHR and can lead to melt-through of the reactor vessel. The ex-vessel stage involves the release of the core debris from the reactor vessel into the containment and resulting phenomena such as CCI, FCI, and DCH.

19.2.3.2.1 In-Vessel Melt Progression

In severe accidents that proceed to vessel failure and release of molten core material into the containment, the in-vessel melt progression establishes the initial conditions for assessment of the thermal and mechanical loads that may ultimately threaten the integrity of the containment. In-vessel melt progression encompasses the phenomena and processes involved in a severe core damage accident. These phenomena and processes start with uncovering of the core and initial heat-up, and continue until either (1) the degraded core is stabilized and cooled within the reactor vessel, or (2) the reactor vessel is breached and molten core material is released into the containment. The phenomena and processes in the ABWR that can occur during in-vessel melt progression include the following:

- Core heat-up resulting from loss of adequate cooling
- Metal-water reaction and cladding oxidation
- Eutectic interactions between core materials, e.g., control blades and fuel assembly channel boxes, resulting in relocation of molten material. Eutectics are mixtures of materials with a melting point lower than that of any other combination of the same components.
- Melting and relocation of cladding, structural materials, and fuel
- Formation of blockages near the bottom of the core resulting from the solidification of relocating molten materials (a wet-core scenario)
- Drainage of molten materials to the vessel lower head region (a dry-core scenario)
- Formation of a melt pool, natural circulation heat transfer, crust formation, and crust failure (a wet-core scenario)
- Lower head breach resulting from failure of a penetration or from local or global creep-rupture.

Adequate core cooling can be defined as providing enough cooling water flow to the reactor core to remove the decay heat produced. Or, if the decay heat is transferred to the containment, providing enough cooling water to containment heat removal systems to remove the decay heat transferred from the core. The mechanisms by which sufficient cooling is provided to the reactor core or the containment are numerous, diverse, and redundant. These mechanisms include both safety and non-safety systems. Examples of safety systems include the RCIC system, high-pressure core flooder systems, RHR system, and ADS. Examples of non-safety systems include the condensate and feedwater systems, condenser, relief valves, and the ACIWA system.

In the event of failure of all safety and non-safety systems to remove the decay heat produced, the core will heat up to the point where damage to the fuel and fuel cladding may occur. Decay heat is transferred through radiative, conductive, and convective heat transfer to the steam, other core materials, and nonfuel materials within the reactor. The insufficient cooling supply results in coolant boiloff and a decreasing level within the reactor vessel as the decay heat generation exceeds the heat removal rate. The coolant level within the core further decreases so that the fuel rods above the coolant level are only cooled by rising steam. The fuel rods begin to overheat and cladding oxidation begins in the presence of steam at high temperatures. As the cladding oxidizes in the presence of steam, hydrogen and additional heat are generated. The fuel cladding is made of a zirconium alloy called zircaloy.

The initial zircaloy oxidation involves oxygen diffusion through a ZrO_2 surface layer. As the fuel rods continue to heat up from decay heat and the exothermic zirconium oxidation reaction, the materials within the reactor with low melting points are expected to melt first and may form eutectics.

Zircaloy with a melting point of 1,757 °C (3,194 °F) begins to melt breaking down the protective ZrO_2 layer, exposing unoxidized zircaloy. Following this, local melting of the fuel rods may cause changes in the core geometry resulting in different steam flow paths. On the one hand, this can lead to an increase in the oxidation process as access to the unoxidized zircaloy is available. On the other hand, the melt formation or changes in the steam flow path could reduce the zircaloy surface available for oxidation and thereby decrease the overall reaction process. In some accident scenarios in which residual amounts of water remain in the bottom of the core and lower plenum, substantial steaming and oxidation can take place.

In addition to oxidation, the potential exists for the zircaloy to interact with the UO_2 fuel, forming eutectics. Formation of eutectics may decrease the effective surface area for oxidation and the overall oxidation rate. The melting point of zircaloy is dependent upon its oxidation state and lattice structure. It has three melting points which include 1,877 °C (3,410 °F) (beta-Zr), 1,977 °C (3,590 °F) (alpha-Zr(O)), and 2,677 °C (4,850 °F) (ZrO_2). When partially oxidized zircaloy is in contact with UO_2, an alpha-Zr(0)/UO_2 based eutectic will form with a liquefaction temperature of approximately 1,897 °C (3,446 °F). Therefore, in the presence of good fuel or cladding contact, fuel liquefaction and melt relocation will commence around this temperature. This has the potential to affect the oxidation behavior of zircaloy-based melt.

Various severe-fuel damage (SFD) test programs discussed in the NUREG's listed above sponsored by the NRC indicate that oxidation of the zircaloy is largely controlled by the availability of a steam supply and that high rates of hydrogen generation can continue after melt formation and relocation. Some of these experiments indicate that the majority of the hydrogen generated occurs after onset of zircaloy melting and fuel dissolution. In steam-rich experiments, oxidation took place over most of the fuel bundle length, and most of the hydrogen is generated early. For steam-starved experiments, oxidation was limited to local regions of the fuel bundle and the majority of the hydrogen is generated after the onset of Zr/UO_2 liquefaction and relocation.

The ABWR contains more than 72,000 kg (158,700 lbm) of zirconium in the active fuel region that has the potential to generate more than 3,100 kg (6,834 lbm) of hydrogen. Hydrogen production and accumulation may represent challenges to the containment in numerous ways including deflagration, detonation, and pressurization. The ABWR containment will be made inert with nitrogen to prevent the occurrence of any deflagration or detonation. Pressurization of the containment from the generation of hydrogen gases will not exceed ASME Service Level C limits.

The SFD tests indicated the potential for incoherent melt-relocation due to noncoherent temperatures within the test bundles. This is because of the different core materials present with a wide range of melting points and eutectic temperatures. Formation of eutectics would result in a nonuniform melting and relocation process. Further differences in the melt-relocation process can be attributed to asymmetric bundle heating that can increase because of zircaloy oxidation. This process begins when one area of the fuel bundle is initially at a temperature higher than the other areas. The higher-temperature zircaloy will consume the available steam through oxidation at a quicker rate.

The oxidation reaction makes the hotter areas hotter still, which further increases the oxidation rate and the local temperatures. This autocatalytic nature of zircaloy oxidation appears to contribute to asymmetric bundle heatup and the potential for incoherent melt-relocation.

As the temperature of the core increases, fission products in vapor form are released. The mixing and transport of these fission products within the primary system depend upon flow paths set up by any existing steam or hydrogen and any interactions with surfaces within the reactor. Such surfaces as the upper internal structures of the reactor vessel may act as a filter where microscopic fission product aerosols suspended in the gas can settle on comparatively cool surfaces by thermophorisis and diffusiophoresis. In addition, retention mechanisms such as turbulent deposition and gravitational settling occur.

The core melt progression, including relocation and fission product release, becomes increasingly difficult to predict as it continues to degrade. The core melt could relocate into the lower reactor vessel plenum. If water is present in the lower plenum, the potential exists for in-vessel steam explosions, where molten fuel rapidly fragments and transfers its energy causing rapid steam generation and shock waves. Another possibility is that the core debris within the lower plenum may quickly melt through the reactor vessel or interact with available water before melting through and entering the lower drywell.

The in-vessel core melt progression, including core degradation, relocation, and failure of the reactor vessel, is rather uncertain. This uncertainty includes the potential for in-vessel steam explosion, the interaction of core debris with internal vessel structures, the time and mode of vessel failure, the composition of the core debris released at vessel failure, the amount of in-vessel hydrogen generation, the in-vessel fission product release, and the transport and retention of fission products and other core materials in the RCS.

19.2.3.2.2 Ex-Vessel Melt Progression

Ex-vessel severe accident progression is affected by the mode and timing of the reactor vessel failure; the primary system pressure at reactor vessel failure; the composition, amount, and character of the molten core debris expelled; the type of concrete used in containment construction; and the availability of water to the lower drywell. The initial response of the containment to ex-vessel severe accident progression is largely a function of the pressure of the RCS at reactor vessel failure and the existence of water within the reactor cavity. If not prevented through design features, risk consequences are usually dominated by early containment failure mechanisms that could result from energetic severe accident phenomena such as HPME with DCH and ex-vessel steam explosions. The long-term response of the containment from ex-vessel severe accident progression is largely a function of the containment pressure and temperature resulting from core-concrete interaction and the availability of containment heat removal mechanisms.

At high RCS pressures, the molten core debris could be ejected from the reactor vessel in jet form causing it to fragment into small particles. The potential exists for the core debris ejected from the vessel to be swept out of the lower drywell and into the upper drywell. Finely fragmented and dispersed core debris could heat the containment atmosphere and lead to large pressure spikes. In addition, chemical reactions of the core debris particulate with oxygen and steam could add to the pressurization loads. This severe accident phenomenon is known as HPME with DCH.

To prevent this phenomenon, the ABWR has incorporated a reliable depressurization system to provide assurance, that in the event of a core melt scenario, that failure of the RPV would occur at a low pressure. Should the RPV fail at a high pressure, the design of the ABWR containment would provide an indirect pathway from the lower to the upper drywell in an effort to decrease the amount of core debris that could contribute to DCH.

The equipment tunnels are located on the periphery of the lower drywell at a midlevel elevation. Core debris exiting the reactor vessel or entrained from the lower drywell during HPME has the potential to reach the tunnels. An accumulation of core debris within the tunnels could lead to melt-through and development of a suppression pool bypass mechanism. GE had not addressed this issue in the SSAR. The staff indicated that it believed that an acceptable resolution to this issue would be for GE to provide reasonable assurance that an appreciable amount of core debris would not enter the tunnels. This could be done by showing that the existing equipment within the lower drywell provides a tortuous pathway to the lower drywell periphery or providing an additional shield structure over the tunnels. This was Open Item F19.2.3.2.2-1.

GE addressed this issue in a letter dated February 7, 1994, which proposed a new SSAR Section 19.E.2.3.6. GE indicated that the equipment tunnels will be partially covered with 1.2 meters of suppression pool water at the low water level allowed by technical specifications. In the event core debris melts through the equipment tunnel, the debris will enter the suppression pool and any additional gases from the lower drywell will pass through the indicated suppression pool level. Also, a HPME results in core debris and elevated temperatures within the upper drywell. In order for the containment penetrations to withstand the elevated temperatures, the operator must actuate the containment spray system. The water from the containment spray system eventually accumulates in the suppression pool, raising the water level to provide additional water coverage of the equipment tunnels. Therefore, the suppression pool water level covering the lower section of the equipment tunnels is sufficient to preclude this potential suppression pool bypass pathway. The staff finds this acceptable. Therefore, Open Item F19.2.3.2.2-1 is resolved.

RPV failure at high or low pressure coincident with water present within the lower drywell could lead to FCI with the potential for rapid steam generation or steam explosions. Rapid steam generation involves the pressurization of containment compartments from nonexplosive steam generation beyond the capability of the compartment to relieve the pressure so that local overpressurization failure of the compartment occurs. Steam explosions involve the rapid mixing of finely fragmented core debris with surrounding water resulting in rapid vaporization and acceleration of surrounding water creating substantial pressure and impact loads. The ABWR is designed so that there is a very low likelihood of water within the lower drywell at the time of reactor vessel failure.

The eventual contact of molten core debris with concrete in the lower drywell will lead to core-concrete interaction (CCI). CCI involves the decomposition of concrete from core debris and can challenge the containment in various mechanisms, including (1) pressurization resulting from the production of steam and noncondensible gases to the point of containment rupture, (2) the transport of high temperature gases and aerosols into the upper drywell leading to high-temperature failure of the containment seals and penetrations, (3) liner melt-through, (4) reactor pedestal melt-through leading to relocation of the reactor vessel and tearing of containment penetrations, and (5) the production of combustible gases such as hydrogen and carbon monoxide. CCI is affected by many factors including the availability of water to the lower drywell, the containment geometry, the composition and amount of core melt, the core melt superheat, and the type of concrete involved.

The ABWR has incorporated several design features to mitigate the effects of CCI. These include an LDF system, an ACIWA system, basaltic concrete for the lower drywell floor, and the COPS. The LDF system provides suppression pool water to assist in cooling core debris once it has entered the lower drywell. The ACIWA system provides for both reactor vessel injection and drywell spray capability to cool core debris or control containment pressurization. Basaltic concrete protects the containment liner from melt-through and decreases the amount of non-condensible gases generated during CCI when compared with limestone-based concretes. The COPS is designed to passively relieve containment pressure to prevent gross containment failure during severe accidents when the containment pressure approaches ASME Service Level C limits. This relief pathway takes advantage of the scrubbing capability of the suppression pool to limit any offsite releases.

19.2.3.3 Severe Accident Mitigative Features

19.2.3.3.1 Hydrogen Generation and Control

Generation and combustion of large quantities of hydrogen is a severe accident phenomenon that can threaten containment integrity. The major source of hydrogen generated is from the oxidation of zirconium with steam when the zirconium reaches temperatures well above normal operating levels. This reaction is commonly referred to as the metal-water reaction.

Research indicates that in-vessel hydrogen generation associated with core-damage can vary over a wide range. The specific amount of oxidation is dependent on a variety of parameters related to sequence progression. These include the RCS pressure, the timing and flow rate of reflooding if it occurs, and the temperature profile of the reactor core during the course of the accident sequence. In addition, ex-vessel hydrogen generation must be considered. Hydrogen is produced as a result of ex-vessel core debris reacting with steam or concrete.

19.2.3.3.1.1 Preventive and/or Mitigative Features

10 CFR 52.47(a)(1)(ii) requires applicants for a standard design certification to provide demonstration of compliance with any technically relevant portions of the Three Mile Island Requirements set forth in 10 CFR 50.34(f). 10 CFR 50.34(f)(2)(ix) requires a system for hydrogen control that can provide with reasonable assurance that uniformly distributed hydrogen concentrations in the containment do not exceed 10 percent during and following an accident that releases an equivalent amount of hydrogen as would be generated from a 100 percent fuel-clad metal-water reaction, or that the postaccident atmosphere will not support hydrogen combustion.

In SECY-90-016, the staff recommended that the Commission approve the staff's position that the requirements of 10 CFR 50.34(f)(2)(ix) remain unchanged for evolutionary LWRs. In its June 26, 1990 SRM, the Commission approved the staff's position.

To comply with 10 CFR 50.34(f)(2)(ix), the ABWR will have an inert atmosphere during normal operation. The inert containment prevents hydrogen combustion and/or detonations from occurring.

10 CFR 50.34(f)(3)(v) requires containment integrity to be maintained below ASME Service Level C limits for steel containments and the factored load category for concrete containments during an accident that releases hydrogen generated from 100 percent fuel-clad metal-water reaction. GE performed an analysis in Section 19E.2.3.2 of the ABWR SSAR of the capability of the containment to withstand pressurization from a 100-percent, fuel-clad, metal-water reaction coupled with a large-break LOCA. The analysis indicated a peak containment pressure of about 618 kPa absolute (75 psig).

The ABWR has a concrete containment with a steel upper drywell head. The steel upper drywell head, based on GE structural analyses, has been shown to be the most limiting structural component of the containment, with a Service Level C limit of 666.9 kPa gage (97 psig). Therefore, the containment pressurization from a 100-percent, fuel-clad, metal-water reaction coupled with a large-break LOCA is below the Service Level C limit.

19.2.3.3.1.2 Basis for Acceptability

The ABWR design meets the requirements of SECY-90-016 and 10 CFR 50.34(f)(2)(ix) by utilizing a nitrogen-inerted atmosphere within its containment. The ABWR design is capable of withstanding the pressurization loadings resulting from a large-break LOCA and hydrogen generation equivalent to a 100-percent fuel-clad metal-water reaction as required by 10 CFR 50.34(f)(3)(v).

19.2.3.3.2 Core Debris Coolability

Coolability and quenchability have been the subject of extensive research over the past decade. However, much uncertainty still exists about these phenomena, which will most likely not be resolved in the near future. Because of this uncertainty, the NRC decided not to address the question of whether coolability or quenchability has been achieved or can be achieved, but rather, what the impact on the containment design is if they are not achieved.

CCI is a severe accident phenomenon that involves the melting and decomposition of concrete in contact with molten corium. This phenomenon may occur following accident sequences that result in molten corium's breaching the reactor vessel and spreading onto the lower drywell floor. The thickness of the corium layer within the lower drywell depends upon the amount of core debris, its spreadability, and the lower drywell floor area. Once on the drywell floor, the molten corium may react with the concrete and any available water, producing noncondensible gases, water vapor, and heat from exothermic reactions.

CCI can challenge the containment by various mechanisms including pressurization from noncondensible gas and steam generated, destruction of structural support members, and melt-through of the containment liner. Noncondensible gases, primarily carbon dioxide, carbon monoxide, and hydrogen, are released from the concrete as it decomposes and are formed from reactions between water and metals within the molten corium. The corium and concrete are heated from the combined effects of decay heat and exothermic chemical reactions.

19.2.3.3.2.1 Preventive and/or Mitigative Features

In SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs," the staff recommended that the Commission approve the position that both the evolutionary and passive LWR designs meet the following criteria: (1) provide reactor- cavity floor space to enhance debris spreading, (2) provide a means to flood the reactor cavity to assist in the cooling process, (3) protect the containment liner and other structural members with concrete, if necessary, and (4) ensure that the best-estimate environmental conditions (pressure and temperature) resulting from core-concrete interactions do not exceed Service Level C for steel containments or factored load category for concrete containments, for approximately 24 hours. In addition, they must ensure that the containment capability has margin to accommodate uncertainties in the environmental conditions from core-concrete interactions. In its July 21, 1993 SRM, the Commission approved the staff's position.

Therefore, the staff's proposed applicable regulation for core debris coolability is as follows:

> The standard design must include features that reduce the potential for and effect of interactions with molten core debris by:
>
> (1) providing reactor cavity floor space to enhance debris spreading;
>
> (2) providing a means to flood the reactor cavity to assist in the cooling process;
>
> (3) protecting the containment liner and other structural members with concrete, if necessary; and

(4) providing design features that ensure that the best-estimate environmental conditions (pressure and temperature) resulting from core-concrete interactions do not exceed service level C for steel containments or factored load category for concrete containments, for approximately 24 hours.

GE incorporated numerous features in the ABWR to help mitigate the effects of core-concrete interaction. The following features were judged by the staff as being most important: a large lower drywell floor area with minimal obstructions to the spreading of core debris, an LDF system, an ac-independent water addition system, use of sacrificial basaltic concrete for the lower drywell floor, a thick reactor pedestal wall, and a COPS.

19.2.3.3.2.1.1 Lower Drywell Floor Area

The lower drywell is 10.6 meters (34.8 ft) in diameter, which provides a floor area of 88 m^2 (947.2 ft^2). The lower drywell contains embedded sump pits that could lead to the accumulation of core debris and accelerated CCI. To prevent this, the sumps are provided with protection to prevent the entrance of core debris. The sump protection is described in 19.2.3.3.8 below.

Even with the presence of the sumps, the lower drywell will have an unobstructed floor area greater than 79 m^2 (850.4 ft^2). This is sufficient floor area to satisfy the EPRI design criterion of 0.02 m^2/MWt for debris coolability. This value represents the EPRI Requirements Document estimate of what is required to adequately cool corium debris. The staff does not support or dispute the EPRI floor sizing criterion. Instead, the staff concludes that an unobstructed floor area, along with the design features mentioned above, provides measures to promote the potential for core debris coolability, but does not necessarily ensure it.

To determine whether the lower drywell meets the criteria within SECY-93-087 relative to providing reactor cavity floor space to enhance debris spreading, the staff evaluated the total size of the lower drywell, the number of obstructions present to prevent the spreading of molten core debris, and the impact on the containment design of requiring further modifications. Based on minimal obstructions on the floor area described above, the staff concludes that the design is acceptable.

19.2.3.3.2.1.2 Lower Drywell Flooder System

An LDF was incorporated into the ABWR design to supply water from the suppression pool to the lower drywell to assist in the cooling process of the corium. The water also cools and condenses gases that evolved during CCI, thereby limiting containment temperature and pressure increases. The LDF is discussed in Sections 9.5.12 and 19E.2.8.2 of the SSAR.

The LDF consists of ten 100 mm (4 in.) stainless steel piping lines from the suppression pool to the lower drywell with thermally activated flooder valves attached to them. The thermally activated flooder valves open when the lower drywell air temperature reaches 260 °C (500 °F). Each flooder valve has a minimum flow rate of 10.8 Kg (2.77 gallons/sec) and contains four components: a stainless steel disk, a teflon disk, a fusible metal plug, and a plastic cap. The stainless steel disk prevents suppression pool water from corroding the teflon disk and fusible metal plug. The teflon disk provides an insulating barrier to prevent the suppression pool water from contacting the fusible metal plug. This insulating barrier is needed to assure that the fusible metal plug is not cooled by the suppression pool water and prevented from melting. The teflon disk will not melt or stick in the valve because its softening temperature is approximately 400 °C (769 °F) and its chemical resistance is higher. It therefore will not adhere to the stainless steel plug or to the fusible plug. The fusible metal plug has a small, raised, annular ring around its circumference approximately 2.0 mm (0.08 in.) high. It is this annular ring that has to melt in order for the LDF to actuate.

The fusible metal plug is made of an alloy mixture of two or more metals so that the plug melts when its temperature reaches 260 °C (500 °F). The end of the flooder valve line is covered with a plastic cover with a low melting point below 130 °C (266 °F). This plastic cover prevents corrosion of the fusible metal material from intrusion of moisture. The flooder valves are mounted in the vertical position so that the fusible metal faces downward to facilitate opening of the valve when the melting temperature of the fusible metal is reached. Heat transfer resulting in melting of the fusible plug occurs through a combination of conduction, convection, and radiation. Heat is conducted from the stainless steel pipe to the fusible plug causing it to melt. This heat is received from the atmosphere within the lower drywell through convection. In addition, the stainless steel pipe also receives radiative heat from the corium on the lower drywell floor. The LDF is safety-related and seismic Category I.

During each refueling outage, the 10 fusible plug flanges and outlets will be inspected to ensure there is no leakage. Once every two refueling outages, 2 of the 10 fusible plugs will be removed, inspected, and tested to confirm their function and verify the temperature setpoint.

In Section 19E.2.8.2.2, GE calculated the minimum acceptable flow rate to remove decay heat and heat generated during CCI to be a total of 0.018 m^3/sec (5 gallons/sec) compared to the system flow rate of .099 m^3/sec (26 gallons/sec) assuming the failure of one of the ten flooder lines to open. This indicates that only two of the flooder lines are needed to remove decay heat and exothermic heat from zirconium oxidation. Opening of additional flooder lines contributes to the flooding within the lower drywell.

In Section 19E.2.8.2.4, GE calculated approximate values for the minimum time (21 minutes) and maximum time (1.3 hours) to fill the lower drywell. These times are very sensitive to the assumptions used and are strongly dependent upon the accident sequence selected. For instance, the calculations assume a 100-percent core debris at 1-percent-rated thermal power, no heat from exothermic reactions, all heat rejection to the water, and failure of one passive flooder valve to operate. The staff concludes that this is acceptable as it provides a relative time frame to judge the adequacy of the LDF design, but the actual flooding rate and time to complete flooding are accident-sequence specific.

Based on the above discussions, the staff concludes that the LDF meets the criteria of SECY-93-087 for providing a means to flood the reactor cavity to assist in the cooling process of core debris.

19.2.3.3.2.1.3 AC-Independent Water Addition System

In addition to the three electrically and mechanically independent divisions of the RHR system of the ABWR that provide reactor vessel injection and containment spray, an ACIWA system has been incorporated. The ACIWA system consists of piping and manual valves connecting the fire protection system to the loop C RHR pump discharge line downstream of the pump's discharge check valve. The C loop is capable of providing low-pressure injection to the reactor vessel or containment spray to the upper drywell. Within the fire protection system, an independent diesel-driven pump exists that would provide the pumping capability. Additionally, an external hookup outside the reactor building for connection of a fire truck pump to an alternate water source is provided. An ac-driven fire pump is included within the fire protection system. However, its contribution to severe accident prevention and mitigation has been excluded because of its dependence on electrical power. The ACIWA system is discussed in Sections 5.4.7 and 19K.11.5 of the SSAR.

Injection to the reactor vessel using the ACIWA system is intended to prevent core damage. In the event that it is not initiated in time to prevent core damage and reactor vessel melt-through, the ACIWA, when operated in the reactor vessel injection mode, would provide water to the lower drywell through the breech in the reactor vessel to assist in cooling ex-vessel core debris. This flooding of the lower drywell could be in addition to or in-place of the flooding provided by the LDF. The actual circumstances are accident-sequence specific. For example, if the ACIWA provides flooding to the lower drywell immediately following vessel breach, then the temperature within the lower drywell may never reach the initiation temperature for the LDF or if initiation of the ACIWA is delayed, the LDF would open followed by the ACIWA.

Operation of the ACIWA in the containment spray mode controls atmospheric temperatures in the upper drywell and provides fission product scrubbing. This system is very beneficial in delaying the time to or preventing the opening of the COPS as is indicated in Appendix 19E to Section 19 of the SSAR.

In both the vessel injection mode or containment spray mode, the ACIWA supplies water to the containment thus increasing the thermal mass, which in turn slows the overall pressure rise. Operation of the ACIWA is manual. The diesel-driven fire pump can be operated from the control room; the injection valves, which must be opened or closed, are located in the same loop C ECCS valve room. The ACIWA system can supply, from either the diesel-driven fire pump or fire truck pump, between 0.04 m^3/sec (630 gpm) and 0.06 m^3/sec (950 gpm) for conditions between runout and back pressure equal to the COPS initiation setpoint. Inspection and testing of the ACIWA system are discussed in SSAR Section 19K.11.5.

Based on the above discussions, the staff concludes that the ACIWA provides another means of flooding the lower drywell to assist in the cooling process of core debris, as specified in SECY-93-087.

19.2.3.3.2.1.4 Sacrificial Basaltic Concrete

Basaltic concrete is a type of siliceous concrete used in the construction of nuclear power plants and is found throughout the United States. This concrete melts over a range of 1,077 °C - 1,376 °C (1970 - 2,510 °F) and typically liberates 1.5 weight-percent carbon dioxide gas and 5 weight-percent water vapor when heated to melting (NUREG/CR-5564).

In Section 6.2.1.1.10.3 of the SSAR, GE stated that the ABWR will use a 1.5 m layer of basaltic concrete above the containment liner with a low gas content. The basaltic concrete selected will have less than 4 weight-percent of calcium carbonate, which results in low gas generation rates when subjected to CCI. This concrete is designed to protect the containment liner in the event that CCI occurs. Basaltic concrete was chosen as it reduces the production of noncondensible gases when compared with other forms of concrete, such as limestone and limestone-common sand concrete. Reduced noncondensible gas and water vapor production from basaltic concrete has been observed during testing (NUREG/CR-5564 and NUREG/CR-5423). However, basaltic concrete does have a lower melting point and quicker ablation rate when compared with limestone and limestone-common sand concrete.

Using the observations discussed above, based on engineering judgment, the staff concludes that the 1.5 m layer of basaltic concrete meets the criteria specified in SECY-93-087 relating to protecting the containment liner and provides sufficient protection for the containment liner.

19.2.3.3.2.1.5 Reactor Pressure Vessel Pedestal

The basaltic concrete discussed above protects the containment liner from core-concrete attack in the axial direction. Core-concrete attack in the radial direction could affect the RPV pedestal. The cylindrical RPV pedestal is formed from two concentric steel rings interspaced with internal stiffeners and filled with concrete. The RPV pedestal is rigidly connected to the diaphragm floor and separates the lower drywell from the wetwell while supporting the loads from the RPV and the diaphragm floor. Built into the RPV pedestal is the drywell-to-wetwell connecting vent system that directs steam from the lower drywell to the suppression pool and upper drywell.

The inner diameter of the RPV pedestal is the outer boundary of the lower drywell. As such, the pedestal is the radial barrier to the horizontal flow of corium. If corium contacts the RPV pedestal, the inner steel cylinder would be attacked and the concrete fill would be subject to ablation. Unabated ablation could lead to failure of the pedestal and subsequent collapse of the RPV and diaphragm floor leading to gross containment failure.

The width of the RPV pedestal is 1.7 m (5.6 ft). The steel rings and internal stiffeners provide the design strength for the RPV pedestal, while the concrete strength is not considered. In Section 19EC of the ABWR SSAR, GE presents the results of an analysis that indicate that only the steel outer shell and 15 cm (6 in.) of internal stiffeners are required to maintain RPV pedestal loads below 90 percent of yield strength.

The staff performed an estimate of the stresses in the RPV pedestal based on the methodology in "Formulas for Stress and Strain," by R. J. Roark and W. Young, McGraw Hill, 1982. Based on these approximate calculations, the staff concludes that adequate margin exists to the yield strength of the RPV pedestal following 1.5 m (5 ft) of radial ablation and that the RPV pedestal is thick enough to withstand the effects of radial ablation resulting from CCI. These attributes meet the criteria specified in SECY-93-087 for protecting structural members with concrete.

19.2.3.3.2.1.6 Containment Overpressure Protection System

The COPS passively relieves containment pressurization before containment pressure reaches ASME Service Level C limits. This system provides for a controlled release through a containment vent pathway with fission product scrubbing provided by the suppression pool. With respect to CCI, the COPS prevents catastrophic overpressurization failure of the containment for severe accident sequences involving prolonged periods of CCI. The COPS ensures that containment pressurization resulting from CCI does not exceed the ASME Service Level C limit of 666.9 kPa gage (97 psig), as the actuation setpoint is 617.8 kPa gage (90 psig).

19.2.3.3.2.2 Analyses

In SECY-93-087, the staff concluded that the evolutionary light water reactors should ensure that the best estimate environmental conditions (pressure and temperature) resulting from core-concrete interactions do not exceed service Level C for steel containments or factored load category for concrete containments, for approximately 24 hours. In addition, designers should ensure that the containment capability has a margin to accommodate uncertainties in the environmental conditions from core-concrete interactions.

The staff concluded that twenty-four hours was an appropriate time period based on sufficient time to allow for decay of fission products, operator intervention, utilization of accident management strategies, fission product deposition in the containment through natural mechanisms, and offsite protective measures. It was developed as a guideline and not a strict criterion in recognition of the uncertainties in severe accident progression and phenomenology.

19.2.3.3.2.2.1 GE Analyses

In Section 19E.2 of the ABWR SSAR, GE provided the results of its deterministic evaluation for several specific accident challenges to evaluate the containments performance. To perform this evaluation, GE used the MAAP3.0B code modified to model the configuration of the ABWR. The new version of the code is referred to as MAAP-ABWR.

Using the ABWR probabilistic safety assessment, GE considered accident classes representing the largest frequencies in selecting the accident sequences to be studied. Eight accident sequences were selected for analysis using MAAP-ABWR. These accident sequences include loss of core cooling with the reactor vessel failing at low and high pressure, SBO, loss of containment heat removal, large break loss-of-coolant accident, and ATWS at low and high pressure and ATWS concurrent with an SBO. For each accident sequence, several mitigating systems could be used to prevent or reduce the release of fission products into the environment. These mitigating systems include in-vessel recovery, passive flooder system, ACIWA, containment heat removal, and containment sprays.

The results of the analyses for each accident sequence are presented in summary form in Table 19E.2-16 of the ABWR SSAR. These analyses generally indicate core debris coolability and little, if any, CCI. The time-to-release of fission products ranges from 8.6 to 50 hours from the start of the transient with the most likely fission product release location through the COPS. The COPS prevents the containment pressure from reaching the ASME Service Level C limit. However, for some sequences, the time to COPS actuation is less than 24 hours. For example, the accident sequence resulting in a release time of 8.6 hours is of extremely low probability involving an SBO with failure of the combustible gas turbine concurrent with an ATWS in which all reactivity control fails. However, if credit is given to operation of the ACIWA in the containment spray mode, the time-to-release of fission products increases to 26.4 hours.

A benchmark of the containment's passive pressure capability is its ability to accommodate the loss of containment heat removal sequence analyzed by GE in section 19E.2.2.4 of the ABWR SSAR. This analysis assumes that reactor vessel injection is maintained with all the decay heat being transferred to the suppression pool. Core damage does not occur. This analysis indicates that the COPS would actuate in approximately 21.7 hours. COPS actuation results from saturation of the suppression pool pressurizing the containment. This sequence indicates that the time to COPS actuation, even without the added pressurization and energy sources from severe accidents, cannot be extended much beyond 20 hours in the absence of active decay heat removal.

With the addition of noncondensible gases from CCI and heat from the exothermic metal-water reactions during a severe accident, the time to COPS actuation will be less. This is an important point in that COPS actuation before 24 hours cannot be prevented unless additional heat capacity is added to the containment or a containment heat removal system is recovered. The ACIWA system, as discussed above in Section 19.2.3.3.2.1.3, can provide additional heat capacity to prolong the time to COPS actuation. Based on GE's analysis provided in Table 19E.2-16 of the ABWR SSAR, the time to COPS actuation is delayed by at least 10 hours for cases in which additional water is added to the containment by the ACIWA, when compared with the same sequence in which only the LDF system actuates to cool the core debris. The ACIWA is crucial to delaying the time to COPS actuation.

Section 19EC presents the results of an uncertainty analyses performed by GE using MAAP-ABWR to investigate the uncertainties associated with debris coolability. These analyses evaluated the impact of parameters such as the amount of core debris, debris-to-water heat transfer, amount of steel in the debris, delayed flooding of the lower drywell, and use of the ACIWA system on CCI, containment pressurization, COPS actuation, and fission product release.

As discussed in Section 19.2.3.3.2.1.4 above, the ABWR will have a 1.5 m (4.9 ft) layer of basaltic concrete above the containment liner. This concrete layer is designed to protect the containment liner from being breached in the event that significant CCI occurs. In Section 19EC using the MAAP-ABWR code, GE provided the results of an uncertainty analyses that calculated the extent of axial ablation. The results, provided in Table 19ED.5-2 of the ABWR SSAR, indicate that axial ablation will not exceed 1 m (3.3 ft) in a 24-hour period.

As discussed in Section 19.2.3.3.2.1.5 above, GE indicated that the distance the molten corium must ablate in the radial direction is 1.55 m (5.1 ft) before the minimum wall thickness of the pedestal is reached. In Section 19EC using the MAAP-ABWR code, GE provided the results of an uncertainty analyses that calculated the extent of radial ablation by multiplying the axial ablation depth by 1/5. GE selected the 1/5 value based on the results of previous CCI experiments. This multiplying factor was necessary, as MAAP assumes that radial and axial penetration are identical. The results, provided in Table 19ED.5-2 of the ABWR SSAR, indicate that radial ablation does not represent a significant threat to the containment.

19.2.3.3.2.2.2 Staff Analyses

The staff analyzed in-house the response of the ABWR using the MELCOR code. In addition, the staff's contractor Sandia National Laboratories (SNL) performed additional analyses using the MELCOR code. The results of the SNL evaluation were sent to GE and placed on the docket. The MELCOR results generally reproduced the event sequences predicted by MAAP, albeit usually with timing shifts. These timing shifts did not affect the safety insights for the containment analyses.

19.2.3.3.2.2.3 Conclusions

The staff did not rely on any one specific sequence or scenario performed by GE using the MAAP-ABWR code nor by the staff's contractor (Sandia National Laboratories) in determining whether the ABWR met the criterion in SECY-93-087 for ensuring that containment conditions do not exceed Service Level C for approximately 24 hours from CCI. Rather, the staff evaluated the range of results provided by these codes, with due consideration of the uncertainties inherent within them, and the capability of the design to extend the time period to COPS actuation through intervention. The ACIWA is fundamental to prolonging the period to COPS actuation. Once COPS is actuated, containment pressurization is relieved through a controlled pathway that takes advantage of scrubbing by the suppression pool. The staff recognizes that there are sequences in which COPS actuation in under 24 hours is required to maintain containment stresses below ASME Service Level C limits.

The staff concludes that the ABWR design meets the criterion when use of the mitigation systems incorporated into the design is factored in, such as the LDF and ACIWA system.

19.2.3.3.2.3 Basis for Acceptability

The ABWR meets the criteria of SECY-93-087 and the staff's proposed applicable regulation for core debris coolability through (1) providing a lower drywell unobstructed floor area greater than 79 m^2 (850 ft^2) to enhance debris spreading, (2) providing an LDF system and ACIWA system to flood the lower drywell, (3) providing a 1.5 m (4.92 ft) layer of basaltic concrete to protect the containment liner, (4) providing a thick reactor vessel pedestal, and (5) providing a COPS. Containment conditions resulting from CCI can be maintained below Service Level C for approximately 24 hours, through incorporation of the above-listed design features.

19.2.3.3.3 High-Pressure Core Melt Ejection

High-pressure core melt ejection (HPME) and subsequent DCH are severe accident phenomena that could lead to early containment failure resulting in large radioactive releases into the environment. HPME is the ejection of core debris from the reactor vessel at a high pressure. DCH is the sudden heatup and pressurization of the containment resulting from the fragmentation and dispersal of core debris within the containment atmosphere.

19.2.3.3.3.1 Preventive and/or Mitigative Features

In SECY-90-016, Evolutionary Light-Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements, the staff concluded that evolutionary LWR designs should include a depressurization system and cavity design features to contain ejected core debris. In its June 26, 1990, SRM, the Commission approved the staff's position that evolutionary LWR designs include a depressurization system and cavity design to contain core debris. In addition, the Commission stated that the cavity design, as a mitigating feature, should not unduly interfere with operations including refueling, maintenance, or surveillance activities.

In SECY-93-087, Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs, the staff recommended that the Commission approve the general criteria that the evolutionary LWR designs provide a reliable depressurization system and cavity design features to decrease the amount of ejected core debris that reaches the upper containment. In its July 21, 1993, SRM, the Commission approved the staff's position.

Based on engineering judgment, the staff believes that examples of cavity design features that will decrease the amount of ejected core debris that reaches the upper containment include ledges or walls that would deflect core debris and an indirect path from the lower drywell to the upper containment. The staff position within SECY-93-087 evolved from the staff position in SECY-90-016 and forms the basis for the staff's review and evaluation.

Therefore, the staff's proposed applicable regulation for high-pressure core melt ejection is as follows:

> The standard design must provide a reliable means to depressurize the reactor coolant system and cavity design features to reduce the amount of ejected core debris that may reach the upper containment so that the potential for and effects of interactions with molten core ejected under high pressure are reduced.

The ABWR has an ADS that is discussed in Sections 5.2.2, 6.3, 7.3, and 19D.6.2.5 of the SSAR. The staff's evaluation of the ADS is provided in Sections 6.3 and 7.3.1.2 of this report. The ADS is a safety grade system that can be used to depressurize the reactor when it is shut down and isolated. The ADS consists of 8 SRVs, which are a subset of a total of 18 SRVs. The SRVs provide three main functions: overpressure relief operation using pneumatic actuators, safety operation using steam overpressure, and depressurization operation using the ADS valves. The eight ADS SRVs can function in either the ADS or SRV mode. All of the SRVs are located on the main steamlines and discharge to the suppression pool. The ADS is automatically initiated or can be manually initiated. ADS requires dc power for the solenoid valves and a nitrogen gas supply for the servo valves and pneumatic actuators. Nitrogen gas is supplied from either the high-pressure nitrogen gas supply system or two backup safety-grade nitrogen gas supplies through an accumulator. The SRVs of the ADS each have two accumulators: one for the ADS function and one for the relief function. The accumulator capacity is sufficient for one actuation at drywell design pressure or five actuations at normal drywell pressure.

The ADS SRVs must remain open during the in-vessel phase of a severe accident to ensure that any potential vessel failure occurs at low pressure. Once the reactor vessel has failed, the ADS system is no longer needed. GE indicated that the capability of the depressurization system will not be degraded as a result of the radiation exposure or thermal loads. The DBA radiation environment (TID-14484) is more limiting than that predicted through best-estimate analysis for a severe accident. The thermal loads on the valve actuators are expected to be similar to those used for equipment qualification, and therefore the ADS valves will not be subject to degradation. Nitrogen, which is used to hold the ADS valves open, is supplied from outside of containment and therefore will not be exposed to the harsh severe accident environment. GE indicated that the nitrogen supply will be adequate to assure SRV operability over a full range of hypothetical accidents.

The design of the lower drywell of the ABWR is expected to decrease the amount of ejected core debris that reaches the upper drywell. This decrease is anticipated through the following: (1) capture and trapping of some debris in the lower drywell, (2) impaction and removal of core debris as it is transported between the lower and upper drywell, and (3) division of exiting core debris and gas from the lower drywell into both the upper drywell and wetwell. The lower drywell is a cylindrical cavity with horizontal vent openings to the downcomers at two-thirds of the cavity height. The upper portion of the lower drywell contains the CRD mechanisms. Debris circulating within the lower drywell may be trapped on the CRD mechanisms and other stagnant areas.

For debris to travel from the lower cavity floor to the upper drywell, it must travel vertically from the lower drywell, horizontally to the downcomer vent, and then vertically through the downcomer to the upper containment. This tortuous path, which contains two 90-degree turns, provides an indirect path from the lower drywell to the upper drywell and is expected to enhance removal of core debris from the gas jet stream through impaction. As pressurization of the lower and upper drywell increases, the suppression pool level within the lower downcomers will be forced down to expose the horizontal vents to the suppression pool. Once the horizontal vents have been cleared, the gas and debris leaving the lower drywell will split into two paths: one to the upper drywell and the other to the suppression pool.

The pathway alongside of the reactor vessel is closed off by the reactor vessel skirt. This prevents core debris from the lower drywell from being ejected alongside of the vessel into the upper drywell.

The ABWR containment is inert. A postulated loading from an HPME/DCH event results from hydrogen generation and combustion generated from the oxidation of metallic debris ejected with the core melt. As the ABWR is inerted, any combustion of hydrogen and resulting pressurization loadings is limited to the amount of residual oxygen present within the containment atmosphere.

19.2.3.3.3.2 Basis for Acceptability

In SECY-93-087, Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs, the staff recommended that the Commission approve the general criteria that the evolutionary LWR designs provide a reliable depressurization system and cavity design features to decrease the amount of ejected core debris that reaches the upper containment. In its July 21, 1993 SRM, the Commission approved the staff's position.

The ADS of the ABWR is provided with a reliable nitrogen supply and dc power to ensure its operability. The containment design of the ABWR is expected to decrease the amount of ejected core debris that reaches the upper drywell. This decrease is anticipated through the following: (1) capture and trapping of debris in the lower drywell, (2) impaction and removal of core debris as it is transported between the lower and upper drywell, and (3) division of exiting core debris and gas from the lower drywell into both the upper drywell and wetwell. Based on the above, the staff concludes that the criteria of SECY-93-087 and the staff's proposed applicable regulation for high-pressure core melt ejection have been met.

19.2.3.3.4 ABWR Containment Vent Design

In SECY-90-016, the staff discussed the incorporation of a containment vent system in the ABWR. The containment vent system is identified as the COPS. The design basis of the COPS is discussed in Sections 6.2.5.2.6.2, 19.E.2.8.1, and 19K.11.6 of the ABWR SSAR.

The desirability of venting a BWR containment to mitigate multiple-failure accidents beyond the design basis has been accepted for some time. Since 1981, the BWR EPGs, developed by the BWR Owners' Group and approved by the NRC for existing BWRs, have called for venting the containment wetwell airspace. The ABWR has a COPS that is designed to avoid gross containment failure resulting from postulated slow-rising overpressure scenarios. The COPS vents the containment from the wetwell airspace, thereby taking advantage of the scrubbing capability of the suppression pool. In transient events, fission products will be directed to the suppression pool through the SRVs. For LOCAs and severe accident scenarios in which the reactor vessel fails, fission products will be directed through the DVCs to the suppression pool.

Without incorporation of the COPS, overpressurization of the containment could lead to failure of the drywell head and fission product releases that have not been scrubbed. The COPS provides a controlled scrubbed vent path from the wetwell with provisions that allow for reisolation of the containment.

19.2.3.3.4.1 System Description

The COPS consists of two containment isolation valves (CIVs) (F007 and F010) and two 200 mm (8 in.) diameter overpressure relief rupture disks (D001 and D002) mounted in succession in a 250 mm (10 in.) diameter line that connects the wetwell airspace to the plant stack. The CIVs are located in the reactor building as close as practical to the containment. Downstream of the CIVs is the first rupture disk (D001) with a pressure setpoint of 617.8 kPa (90 psig) at 93 °C (200 °F). This rupture disk is expected to have a mean opening tolerance of ±5 percent pressure. Further downstream, before the entrance to the plant stack, is the second rupture disk (D002) with a pressure setpoint of approximately 0.03 MPa (4.35 psig).

The area between the rupture disks is made inert with nitrogen to eliminate the potential for combustion within the portion of the vent path within the reactor building. The rationale for setting the second rupture disk setpoint much lower than that of the first rupture disk is to eliminate the possibility of pressurization between the rupture disks adversely affecting the actuation pressure. For example, if some type of in-leakage occurred between two high- pressure rupture disks, the containment pressure required to burst the first rupture disk could exceed the design bursting pressure of the disk. By utilizing a high-pressure rupture disk in succession with a low-pressure rupture disk, it is expected that significant in-leakage would pressurize the airspace and burst the low-pressure rupture disk and thereby not affect the required containment pressure for bursting the high-pressure rupture disk.

The CIVs in the COPS pathway are subject to the leak-testing requirements associated with 10 CFR Part 50, Appendix J as specified in the SSAR Section 6.2 and the Inservice Testing Requirements as specified in the SSAR Section 3.9. These CIVs are normally open so that the containment atmospheric pressure is exposed to the first rupture disk during all modes of operation. The CIVs are intentionally not provided with an automatic isolation signal to ensure that they remain open in the event that the COPS is needed to mitigate the consequences of a severe accident. With the CIVs open, the first rupture disk provides the barrier to releases during DBA scenarios. In addition, the CIVs are designed to fail-open upon loss of actuating power. This failure position ensures the availability of the COPS during severe accident scenarios involving multiple failures. If under design-basis conditions, leakage past the rupture disks occurs, the CIVs can be remotely isolated from the control room. In-line radiation monitoring of the vent pathway could be used to detect leakage. However, no leakage is expected within the containment design-basis spectrum 309.9 kPa gage (45 psig). This is based on the design pressure of the rupture disk being substantially above the containment design- basis pressure and pressures associated with a DBA.

The rupture disks will be tested and replaced every 5 years providing additional confidence of actuation pressure. When the rupture disks are procured, a number will be procured at the same time to provide uniformity in the relief pressure.

The purpose of the CIVs is to contribute to the control of the venting process. Following rupture disk actuation, plant operators may decide to reclose the vent path based upon accident management guidance or procedures to be developed by the COL applicant. The CIVs are designed to be capable of fully opening and closing against the pressures associated with venting. GE indicated that the sizing of the COPS vent path is sufficient to allow 35 kg/sec (77.2 lbm/sec) of steam flow at the opening pressure of 617.8 kPa gage (90 psig), which corresponds to an energy flow of about 2.4 percent rated power. At the time of rupture disk actuation, the decay power is expected to be well below 1 percent rated power. The staff concludes that a sufficient margin exists to ensure pressure relief once the rupture disk has been actuated. The sizing of the COPS also prevents suppression-pool-level swell that could force water into the COPS pathway.

The rupture disk setpoint was selected to assure an adequate margin prior to drywell head failure while maximizing the time before fission product releases through COPS. The rupture pressure of the COPS is slightly below the containment's Service Level C capability of 666.9 kPa gage (97 psig), which provides confidence that the integrity of the containment will be maintained before rupture disk actuation.

19.2.3.3.4.2 Basis for Acceptability

In SECY-90-016, the staff recommended that the Commission approve the use of an overpressure protection system that used a dedicated containment vent for the ABWR. In its June 26, 1990, SRM, the Commission approved the staff's recommended use of the COPS on the ABWR, subject to the results of a comprehensive regulatory review, which should fully weigh the potential "downside" risks with the mitigation benefits of the system. In addition, the Commission directed the staff to ensure that the design should provide full capability to maintain control over the venting process.

19.2.3.3.4.2.1 Regulatory Review of the Downside Risks and Mitigation Benefits

The COPS is intended to protect the containment against sequences in which containment integrity is challenged by overpressurization. Without the COPS, the containment failure location is expected to be the upper drywell head with a mean ultimate pressure capability of 1,025 kPa absolute (134 psig). The staff concludes that COPS actuation is preferable to failure of the upper drywell head for the following reasons: (1) releases through the COPS have the advantage of suppression pool scrubbing of fission products, (2) following COPS actuation, the vent pathway can be reisolated, and (3) the actuation setpoint and relief capacity of COPS are selected for optimal containment performance.

The staff evaluated the potential downside risks of COPS on the containment failure frequency and source term.

The pressure setpoint for COPS actuation has an impact on the containment failure frequency. If set too high, the potential exists for failure of the upper drywell head before COPS actuation. GE evaluated the variability in the pressure setpoint at which COPS is actuated, as well as the uncertainty of the drywell head failure pressure. Based on this evaluation, GE concluded that there is between a 2 percent and 5 percent probability of drywell head failure before COPS actuation, dependent upon the specific accident scenario.

The temperature of the rupture disks has an impact on its bursting pressure. GE evaluated a range of temperatures from 38 to 149 °C (100 to 300 °F) for the time various temperatures would take to the rupture disk's opening. Higher temperatures cause a decrease in the bursting pressure, whereas lower temperatures cause an increase. GE concluded that for the range of the temperatures evaluated, the time to rupture disk's opening was within 0.8 hours of the base case and the probability of drywell head failure before rupture disk's opening varied slightly. These results indicate that temperature variations have only a minor effect on the COPS.

A potential adverse impact of the COPS involves its actuation when it might have been possible to recover a containment heat removal system in the time period after COPS initiation and before failure of the upper drywell head. The recovery of the containment heat removal system would therefore prevent the fission product release through the COPS. GE estimated the probability of this to be about 4 to 11 percent depending upon the accident scenarios.

In Section 19E.2.8.1.4 of the ABWR SSAR, GE provided a comparison of ABWR performance with and without the use of the COPS. This included 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. These results indicate that for the dominant sequence, COPS reduces the time of fission product release from 27 to 20 hours if drywell sprays are not available and from 35 to 31 hours if drywell sprays are available. The cesium iodide release fractions are reduced from about 4 percent without COPS to less than 1E-7 with COPS as a result of suppression pool scrubbing.

Table 19E.2-27 of the ABWR SSAR provides the results of the probability of release modes with and without COPS. The probability of drywell head failure increases by about a factor of 40 for accident classes involving transients and loss-of-coolant accidents without COPS. For accident classes that include loss of containment heat removal but successful core cooling before containment failure, the frequency of both drywell head failure and resulting core damage would increase by 2 orders of magnitude without COPS.

In summary, the major benefit of COPS is that it provides assurance that releases from the containment are scrubbed. This fission product scrubbing is provided by the suppression pool. Operators can control releases via COPS through manual isolation valves. While actuation of COPS results in an earlier release of fission products when compared with failure of the upper drywell head, the magnitude of the release is significantly less. Additionally, COPS is expected to reduce the frequency of core damage resulting from accident sequences involving loss of core cooling induced by containment failure.

A major concern related to COPS is that it could lead to earlier, and perhaps unnecessary, releases. However, this does not seem to be a significant factor for the ABWR. In particular, the selection of the system setpoint appears to provide a reasonable balance between the competing goals of minimizing the probability of drywell head failure and maximizing the time before fission product release into the environment. The setpoint is sufficiently high that the time of release will generally still be on the order of 15 to 20 hours, yet low enough that the probability of containment failure before COPS actuation is very small (about 5 percent). Also, the probability of unnecessary system actuation (when it might have been possible to prevent a release by recovering containment heat removal systems before containment failure) is reasonably low (about 10 percent) and COPS is capable of being manually isolated from the control room.

The staff concludes that the COPS has a significant net benefit that outweighs the potential negative aspects of the system.

19.2.3.3.4.2.2 Control Over the Venting Process

The staff reviewed the provisions provided by GE for ensuring full capability to maintain control over the venting process. These provisions include:

(a) selection and justification of the containment pressure when the COPS will actuate

(b) provisions for reclosure of the vent pathway using CIVs that are designed to be capable of fully opening and closing at pressures up to the COPS actuation pressure

(c) providing radiation monitoring within the vent pathway to monitor the potential for an early release or provide guidance for accident management strategies for reclosure of the vent pathway

(d) sizing of the vent pathway to prevent further containment pressurization, once the venting process is actuated, and prevent suppression-pool-level swell into the system piping.

The use of a passive rupture disk prevents the opportunity for venting at pressures below the actuation pressure of the rupture disks. This was a deliberate decision made by GE to prevent early venting and maintain the integrity of the containment as long as possible. To allow for early venting, GE could have installed motor-operated valves instead of a rupture disk.

The staff believes that good engineering rationale can be developed to support either position — to allow for early venting or to maintain containment integrity as long as possible. The decision to provide rupture disks and therefore prevent early venting is philosophical in nature, with the underlying theme of maintaining an intact containment and venting only as a last resort. The containment is the final barrier preventing release of radioactivity into the environment. As such, it should not be unnecessarily or prematurely breached by plant staff.

GE evaluated severe accidents in Chapter 19 of the ABWR SSAR. This evaluation included use of the COPS, where needed, with the rupture disks installed. The evaluation did not include early venting. In this Section (19.2) and Section 19.1 of this report, the staff concludes that GE's analysis in Chapter 19 is acceptable. This acceptability includes the COPS as an integral part of the containment.

19.2.3.3.4.3 Conclusions

The ABWR dedicated containment vent is the COPS, which consists of a vent path from the wetwell airspace to the plant stack. The COPS actuates at a pressure less than ASME Service Level C and has provisions for isolation following actuation. Control over the venting process is assured through selection of the actuation pressure and the capability for vent path reclosure. The results of the regulatory review indicate that the net benefits of the system outweigh the potential negative aspects. Based on the above, the staff concludes that the GE ABWR design meets the Commission-approved staff's position in SECY-90-016 for inclusion of a dedicated containment vent path.

19.2.3.3.5 Suppression Pool Bypass

Although suppression pool bypass is not a severe accident phenomenon, it can become a significant contributor to plant risk. Suppression pool bypass is associated with the failure of the containment system to channel steam and fission product releases through the suppression pool.

The fundamental characteristic of a BWR pressure-suppression containment is that steam released from the RCS will be condensed and scrubbed of radionuclides in the suppression pool and that the pressure rise in the containment will therefore be limited. This is accomplished by directing the steam from the RCS to the suppression pool through a drywell to wetwell connecting vent system. However, leakage paths could exist in the pathway between the drywell and wetwell that could allow steam to bypass the suppression pool, pressurize the containment, and lead to a release. Potential sources of steam bypass include leakage through the vacuum relief valves, cracking of the drywell structure, and penetrations through the drywell structure.

19.2.3.3.5.1 Preventive and/or Mitigative Features

In SECY-90-016, the staff concluded that a special effort should be made to eliminate or further reduce the likelihood of a sequence that could bypass the containment. In SECY-93-087, the staff stated that vendors should make reasonable efforts to minimize the possibility of bypass leakage and should account, in their containment designs, for a certain amount of bypass leakage.

The bypass scenario with the greatest threat to containment integrity in the ABWR is suppression pool bypass through the wetwell-to-drywell vacuum breakers. The vacuum breakers prevent the passage of steam from the drywell to the wetwell by use of a single check valve. However, pressure transients in the wetwell and drywell or suppression pool swell could force the vacuum breaker check valve to open. If the vacuum breaker disk fails to reseat, a bypass path would be created and allow drywell steam passage directly to the wetwell airspace. Such a scenario allows steam to bypass the condensation function of to the suppression pool, to pressurize the wetwell, and challenge containment integrity. If sufficient steam bypass occurs, containment pressurization rates would increase, thereby decreasing the time to opening of the COPS.

GE analyzed the maximum allowable leakage path area of the ABWR design for DBA-type scenarios. The DBA results are in Section 6.2.1.1.5 of the SSAR. The staff's evaluation of the DBA results are provided in Section 6.2.1.8 of this report.

To mitigate the consequences of suppression pool bypass for both DBAs and severe accidents, the ABWR design includes a containment spray system in the wetwell and upper drywell. The B or C train of the RHR system supplies the water flow to the safety-grade sprays. The spray flow is split with about 800 m^3/hr (211,000 gal/hr) going to the drywell spray header and 114 m^3/hr (30,100 gal/hr) going to the wetwell spray header. Although RHR pumps are automatically aligned for their ECCS function, the flow can be manually diverted to the containment sprays to mitigate the pressurization of the wetwell airspace during suppression pool bypass.

In the event of failure of the RHR system, the ac-independent water addition (ACIWA) system can be interconnected with the C loop of the RHR system to provide a flow of between 2385 liters/m (630 gpm) and 3596 liters/m (950 gpm) to the upper drywell containment spray header. The COPS prevents gross containment failure for bypass scenarios in which the containment pressure would exceed the rupture pressure of the rupture disk. COPS allows for a controlled release through reclosure of isolation valves in the vent pathway.

19.2.3.3.5.2 GE Analyses

In Section 19EE of the SSAR, GE performed an analysis to determine the impact on the containment from varying amounts of suppression pool bypass. The study examined the effect of varying a vacuum breaker's bypass leakage area (from zero leakage to leakage from one full open vacuum breaker) on the time to fission product release and the Cesium Iodine (CsI) release fraction at 72 hours for five different scenarios. These scenarios are as follows:

(1) Bypass leakage begins after passive flooder activation; aerosol plugging is neglected.

(2) Bypass leakage is present from the beginning of the accident; aerosol plugging is neglected.

(3) Bypass leakage begins after passive flooder activation; aerosol plugging of the vacuum breaker opening is considered.

(4) Bypass leakage is present from the beginning of the accident; aerosol plugging of the vacuum breaker opening is considered.

(5) Bypass leakage is present from the beginning of the accident and the operator initiates the firewater spray system.

The effective vacuum breaker area was varied from 0 to 2030 cm^2, the latter figure corresponding to one fully open vacuum breaker. The time to fission product release and the CsI release fractions were determined from MAAP-ABWR runs. The dominant severe accident sequence, LCLP (loss of all core cooling with vessel failure occurring at low pressure), was chosen to evaluate plant performance.

A reference point for comparison of the results of GE's analysis is the case with zero bypass leakage and without the firewater spray system. For this case, the elapsed time before rupture disk opening is about 20 hours and the CsI release fraction is less than 10E-07.

CsI release fractions are orders of magnitude larger for cases with bypass leakage than for the case without bypass leakage. For cases with effective suppression pool bypass areas greater than 400 cm^2 (62 in^2), the 72-hour CsI release fractions are approximately 17 percent.

Scenario 5 examined the effects of drywell spray from the ACIWA on cases with bypass leakage present from the beginning of the accident. Assuming the operator initiates the firewater spray within 2 hours of the start of the accident, the elapsed time to rupture disk opening can be delayed to nearly 30 hours for bypass pathways experiencing up to one fully open vacuum breaker.

19.2.3.3.5.3 Basis for Acceptability

The ABWR has the containment spray system and COPS to mitigate the effects of containment bypass and prevent a bypass scenario from progressing to containment failure. The containment spray system can be supplied from either the RHR or ACIWA system. In SECY-90-016, the staff stated that venting should be delayed for approximately 24 hours following the onset of core damage. For the cases in which one vacuum breaker is fully open, the ABWR meets the intent of this criterion when the initiating of containment sprays through the ACIWA system within 2 hours (as described in scenario 5 above) is factored in. The staff concludes that GE has performed a relatively complete analysis to allow an understanding of the capability of the ABWR containment to accommodate a range of bypass conditions through the vacuum breakers. This analysis highlights the importance of the containment spray system to mitigating the consequences of suppression pool bypass.

19.2.3.3.6 Fuel-Coolant Interaction

The containment function can be challenged by energetic or rapid energy releases. One such energetic or rapid energy release is an FCI that results in a steam explosion. The term "steam explosion" refers to a phenomenon in which molten fuel rapidly fragments and transfers its energy to the coolant resulting in rapid steam generation, shock waves, and possible mechanical damage. To be a significant safety concern, the interaction must be very rapid and must involve a large fraction of the core mass. Steam explosions can occur either in-vessel or ex-vessel.

19.2.3.3.6.1 In-Vessel Steam Explosion

NUREG-1116, A Review of the Current Understanding of the Potential for Containment Failure From In-Vessel Steam Explosions, summarized the deliberations of the Steam Explosion Review Group's (SERG's) understanding of the potential for containment failure arising from in-vessel steam explosions during core melt accidents. The consensus reached by the SERG was that the occurrence of an in-vessel steam explosion of sufficient energetics to lead to containment failure was sufficiently low in probability to allow elimination as a credible threat.

This conclusion was reached despite the expression of differing opinions on the modeling of basic steam explosion sequence phenomenology. An opinion supported by most members of the group is that the probability of containment failure is reduced because of the expectation of limited melt mass involvement in the explosion and/or low thermal-to-mechanical energy conversion.

This conclusion was reaffirmed at the meeting of the Committee on the Safety of Nuclear Installations (CSNI), "Specialist Meeting on Fuel-Coolant," in January 1993. The conclusion of the meeting was that alpha-mode failure was highly unlikely because of the structures in the lower reactor vessel head. These structures, such as the CRD guide tubes, would limit the melt mass involvement by causing incoherent relocation of the molten corium.

19.2.3.3.6.2 Ex-Vessel Steam Explosion

In SECY-93-087, the staff stated that any dynamic forces resulting from ex-vessel FCI on the integrity of the containment should be evaluated. One of the conditions necessary for an ex-vessel FCI in the ABWR is for the molten corium to be discharged from the reactor vessel into a body of water in the lower drywell. The design of the ABWR containment substantially reduces the probability of a preexisting body of water in the lower drywell at the time of reactor vessel failure.

The reactor vessel skirt is solid, preventing water transfer from the upper drywell to the lower drywell. In addition, there are no active injection systems in the lower drywell, and the passive flooder system does not actuate until the atmosphere within the lower drywell is approximately 260 °C (500 °F). The connection between the upper drywell and lower drywell is through the vertical connecting vent system, which contains a horizontal 90-degree bend preventing water from reaching the lower drywell.

In Section 19EB.1.1 of the ABWR SSAR, GE stated that only 0.3 percent (5.1E-10) of all core damage sequences would result in water in the lower drywell at the time of vessel failure. In Section 19.1.3.5.4 of this report, the staff evaluated GE's estimate and concluded that the probability of a flooded lower drywell at the time of reactor vessel failure is extremely small.

The staff believes that the low likelihood (5.1E-10) of a flooded lower drywell at the time of reactor vessel failure provides a sufficient basis to conclude that the probability of an ex-vessel steam explosion has been reduced to an acceptably low value and is therefore acceptable. Nevertheless, GE and the staff performed analyses to determine the capability of the ABWR containment to withstand ex-vessel steam explosions.

19.2.3.3.6.2.1 GE Analysis

GE provided an uncertainty and sensitivity analysis of the FCI phenomenon in Section 19EB of the ABWR SSAR. This analysis estimated the ability of the ABWR containment, specifically the lower drywell, to withstand a large energetic FCI (steam explosion). The analysis determined the peak pressure the ABWR pedestal is capable of withstanding and the amount of molten corium interacting with water that would be necessary to produce this peak pressure.

GE calculated the peak pressure the pedestal was capable of withstanding during a steam explosion by determining the average pressure of an impulse the amplitude of which can be estimated by the maximum pressure rise expected during an FCI. The ratio of resistance to deformation to the average pressure of an impulse is given by a series of curves. Using this approach, GE calculated the pedestal's resistance to deformation to be 1.7 MPa (246.5 psia).

When an impulse duration of 5 msec is used, which appears to be reasonable based on pulse widths observed during FCI experiments involving corium simulates, the ratio of resistance to deformation to the average pressure of an impulse is approximately 1.0. This implies that the pedestal can withstand a peak pressure of 1.7 MPa (246.5 psia). For additional conservatism, GE eliminated the need for a specific pulse duration by using the curve with the largest ratio of resistance to deformation to the average pressure of an impulse, which is 2.0. This resulted in a peak pressure capability of 0.85 MPa (123 psia).

To determine the amount of corium necessary to cause this peak pressure, GE calculated the steam formation rate assuming a mass of corium fragments into droplets of an identical radius 2.5 mm (.1 in.) and interacts with water. The mass of corium necessary to produce a peak pressure of 0.85 MPa (123 psia) was 22,400 kg (49,383 lbm), which is approximately 9.5 percent of the entire corium inventory. The peak pressure calculated by GE is at the location of the FCI and does not account for decay of the shock wave as it propagates towards the pedestal wall. GE concluded that the reactor pedestal wall could withstand an FCI involving 9.5 percent of the corium inventory.

19.2.3.3.6.2.2 Staff Analysis

The staff performed an independent assessment of the ABWR containment to withstand a steam explosion using the TEXAS-II computer code. The results of the assessment are documented in report EPRI/NRC 93-203, An Assessment of Ex-Vessel Fuel-Coolant-Interaction Energetics for the General Electric Advanced Boiling Water Reactor (Letter dated March 12, 1993, Richard Borchardt, NRC, to Patrick Marriott, GE).

This assessment evaluated two different possible accident progression sequences, one based on the MAAP code and the other based on the BWRSAR code. The MAAP code scenario used a relatively large (540 kg/sec) (1,190 lbm/sec) corium release composed of a lot of oxides; whereas, the BWRSAR code scenario is more gradual (16.7 kg/sec) (36.8 lbm/sec) and composed of mostly metallics. The results from the MAAP code scenario were identified as being conservative because of the large release rate, whereas the BWRSAR code scenario was identified as being best estimate. The reactor pedestal pressure loads were determined to be approximately 1.1 MPa (160 psia) for the best estimate case and 1.6 MPa (232 psia) for the conservative case. These estimates correspond to local pressure impulse loads of 2.6 kPa-sec (0.38 psia/sec) and 3.7 kPa-sec (0.54 psia/sec).

In a separate analysis, the staff concluded that the pedestal wall could withstand an FCI-generated pressure impulse of 3.7 kPa-sec (0.54 psia/sec) using a ductility ratio of 1.6. It also concluded that the associated radial deflection at this ductility ratio would not compromise the integrity of the reactor vessel and other safety-related piping and equipment.

19.2.3.3.6.3 Basis for Acceptability

Based on the conclusions reached in NUREG-1116 and reaffirmed in the recent CSNI meeting, the staff concludes that in-vessel steam explosions are not a threat to the ABWR containment.

As discussed above in Section 19.2.3.3.6.2, the ABWR containment substantially reduces the probability of a preexisting body of water in the lower drywell at the time of reactor vessel failure. The staff believes that the low likelihood of a flooded lower drywell at the time of reactor vessel failure provides a sufficient basis to conclude that the probability of an ex-vessel steam explosion has been reduced to an acceptably low value and is therefore acceptable.

19.2.3.3.7 Equipment Survivability

The purpose of this section is to discuss the survivability of equipment, both electrical and mechanical, that is needed to prevent and mitigate the consequences of severe accidents. GE addressed equipment survivability in Section 19E.2.1.2.3 of the ABWR SSAR.

Design bases events are defined as conditions of normal operation, including anticipated operational occurrences, design basis accidents, external events, and natural phenomena for which the plant must be designed. Safety-related equipment, both electrical and mechanical, must perform its safety function during design bases events. Section 3.11 of the ABWR SSAR defines the environmental conditions with respect to limiting design conditions for all safety-related mechanical and electrical equipment. The common terminology used for the level of assurance provided for equipment necessary for design bases events is "environmental qualification" or "equipment qualification."

Beyond design basis events can generally be categorized into in-vessel and ex-vessel severe accidents. The environmental conditions resulting from these events are generally more limiting than those from design bases events. The NRC established a criterion to provide a reasonable level of confidence that the necessary equipment will function in the severe accident environment for the time span for which it is needed. This criterion is commonly referred to as "equipment survivability" and is fundamentally different from equipment qualification.

In its SRM of June 16, 1990, relating to SECY-90-016, the Commission approved the staff position that features provided only (not required for design basis accidents) for severe-accident protection (prevention and mitigation) need not be subject to the 10 CFR 50.49 environmental qualification requirements; 10 CFR Part 50, Appendix B quality assurance requirements; and 10 CFR Part 50, Appendix A redundancy/diversity requirements. The reason for this judgement is that the staff does not believe that severe core damage accidents should be design basis accidents in the traditional sense that DBAs have been treated in the past. Therefore, the staff's proposed applicable regulation for equipment survivability is as follows:

> The standard design must include analyses, based on best-available methods, to demonstrate that:
>
> Equipment, both electrical and mechanical, needed to prevent and mitigate the consequences of severe accidents is capable of performing its function for the time period needed in the best-estimate environmental conditions of the severe accident (e.g., pressure temperature, radiation) in which the equipment is relied upon to function.
>
> Instrumentation needed to monitor plant conditions during a severe accident is capable of performing its function for the time period needed in the best-estimate environmental conditions of the severe accident (e.g., pressure, temperature, radiation) in which the instrumentation is relied upon to function.

19.2.3.3.7.1 In-Vessel Severe Accidents

The applicable criterion for equipment, both mechanical and electrical, required for recovery from in-vessel severe accidents is provided in 10 CFR 50.34(f).

Part 50.34(f)(2)(ix)(C) states that equipment necessary for achieving and maintaining safe shutdown of the plant and maintaining containment integrity will perform its safety function during and after being exposed to the environmental conditions attendant with the release of hydrogen generated by the equivalent of a 100 percent fuel-clad metal-water reaction including the environmental conditions created by activation of the hydrogen control system.

Part 50.34(f)(3)(v) states that systems necessary to ensure containment integrity shall be demonstrated to perform their function under conditions associated with an accident that releases hydrogen generated from 100 percent fuel-clad metal-water reaction.

Part 50.34(f)(2)(xvii) requires instrumentation to measure containment pressure, containment water level, containment hydrogen concentration, containment radiation intensity, and noble gas effluents at all potential accident release points.

Part 50.34(f)(2)(xix) requires instrumentation adequate for monitoring plant conditions following an accident that includes core damage.

These regulations collectively indicate the need to perform a systematic evaluation of all equipment, both electrical and mechanical, and instrumentation to ensure its survivability for intervention into an in-vessel severe accident. GE did not perform this systematic evaluation as of SSAR Amendment 32.

The staff stated in the advance SER that it believed that an acceptable resolution of this issue would entail the following:

(1) GE should perform an evaluation using best-estimate means of a degraded in-vessel core damage accident that results in a 100 percent metal-water reaction. The basis for the evaluation should be included. The evaluation should identify the most likely sequences resulting in substantial oxidation of the fuel cladding as a result of the probabilistic safety assessment (PSA). An example of an acceptable sequence would involve accident conditions in which ECCS performance is degraded for a sufficient time to cause cladding oxidation but is later recovered to ensure a safe shutdown. If the analysis assumes an intact primary loop, the basis for this should be supported by the results of the PSA (i.e., LOCA does not contribute significantly to core melt). The impact on the reactor system and containment system from the pressure, temperature, and radiation released should be evaluated. As an example, the safe shutdown and containment equipment identified below should be evaluated. Plots showing pressure and temperature as a function of time should be provided.

If the in-vessel severe accident environment has no effect on the equipment performance, this should be clearly indicated along with the supporting rationale. Examples of such instances include cases in which the equipment has already performed its function before the onset of the accident conditions or the equipment is located in an area not exposed to the environmental conditions, such as being located outside the primary containment. For equipment in which environmental conditions as a result of the in-vessel severe accident are in excess of the equipment qualification range, an engineering rationale must be developed as to why the equipment would survive the environment for the needed time span. This rationale could include such factors as limited time period in the environment; the use of similar equipment in commercial industry exposed to the same environment; the use of analytical extrapolations; or the results of tests performed in the nuclear industry or at national laboratories.

An acceptable example using this rationale is the work that GE performed for electrical penetration assemblies in Section 19F.3.2.2 of the SSAR. In particular, GE referred to experimental tests performed at Sandia National Laboratories on actual electrical penetration assemblies (EPAs) used in operating plants. The tests were performed representative under severe accident conditions with temperatures up to 371 °C (700 °F) and pressures up to 965 kPa (140 psig). Using the results of this work, GE committed to providing EPAs that will maintain leak tightness up to containment pressure of 924 kPa (134 psig) and a temperature of 371 °C (700 °F). The end result of this is that the assumptions used for equipment performance in GE's severe accident evaluation are consistent with the as-built plant.

Safe shutdown equipment that should be addressed include scram equipment, HPCF motor and pump, HPCF isolation valves, HPCF controls, RCIC turbine and pump, RCIC steam valves and cables, RCIC controls, RHR, ADS, shutdown cooling, and others.

Equipment for containment integrity should include containment structure, CIVs - inboard and outboard, electrical penetrations, mechanical penetrations, hatches, sealing mechanisms (welds, bellows, O-ring), as well as others.

(2) With respect to instrumentation requirements, the staff believes that sufficient instrumentation should exist to inform operators of the status of the reactor and the containment at all times as the in-vessel severe accident is intended to be recoverable from and lead to safe shutdown with containment integrity maintained. The emergency operating procedures (EOPs) direct specific manual operator actions based on instrumentation readings and as such all instrumentation should exist where manual operator actions are specified within the EOPs. As a minimum, the instrumentation identified below should be evaluated.

The instrumentation is designed to survive the environment as specified in RG 1.97. However, RG 1.97 only ensures that the instrumentation will survive in the worst environment resulting from a design bases event and not a severe accident. Therefore, engineering rationale must be developed as to why the instrumentation would survive the environment. This rationale could include such factors as limited time period in the environment; the use of similar equipment in commercial industry exposed to the same environment; the use of analytical extrapolations; or the results of tests performed in the nuclear industry or at national laboratories.

Instrumentation should include neutron flux, RPV water level, RPV pressure, sup pool temperature, sup pool level, drywell/wetwell (CW/WW) H_2 concentration, DW/WW O_2 concentration, DW temperature, DW pressure, WW pressure, WW temperature, DW water level, among others.

In response to the open item, GE provided the environmental profiles, a table of the necessary equipment, and the accompanying rationale for in-vessel severe accidents in Section 19E.2.1.2.3 of SSAR Amendment 34. The staff finds this information acceptable. Therefore, Open Item F19.2.3.3.7.1-1 is resolved.

19.2.3.3.7.2 Ex-Vessel Severe Accidents

The applicable criteria for equipment, both electrical and mechanical, required to mitigate the consequences of ex-vessel severe accidents is discussed in the Equipment Survivability section of SECY-90-016. This section indicates that features provided only (not required for design basis accidents) for severe-accident protection (prevention and mitigation) need not be subject to the 10 CFR 50.49 environmental qualification requirements; 10 CFR Part 50, Appendix B quality assurance requirements; and 10 CFR Part 50, Appendix A redundancy/diversity requirements. The reason for this judgement is that the staff does not believe that severe core damage accidents should be design basis accidents in the traditional sense that DBAs have been treated in the past.

However, mitigation features must be designed to provide reasonable assurance that they will operate in the severe-accident environment for which they are intended and over the time span for which they are needed. In cases where safety-related equipment (equipment provided for DBAs) is relied upon to cope with severe accident situations, there should be reasonable assurance that this equipment will survive accident conditions for the period that is needed to perform its intended function.

According to SECY-90-016, GE was to review the various severe accident scenarios analyzed and identify the equipment needed to perform various functions during a severe accident and the environmental conditions under which the equipment must function. Equipment survivability expectations under severe accident conditions should include consideration of the circumstances of applicable initiating events (e.g., SBO and earthquakes) and the environment (e.g., pressure, temperature and radiation) in which the equipment is relied upon to function. The staff concluded that GE had not performed the evaluation as outlined by SECY-90-016 as of Amendment 32 to the SSAR. This was identified as Open Item F19.2.3.3.7.2-1.

As stated in the Advance SER, the staff believed that an acceptable resolution of this issue would entail the following:

(1) GE should provide an evaluation of the dominant accident sequences identified in Section 19E.2.2 of the SSAR. For each accident sequence, GE should identify the mitigation features. Mitigation features should include ADS, ACIWA, and RCIC as appropriate.

In addition, the specific environment profile (pressure, temperature, radiation fields) should be specified. For each mitigation feature, an assessment of survivability should be done using ground rules similar to those specified above for in-vessel accidents. At least the following mitigation features should be evaluated SRVs, containment structure, vacuum breakers, inboard and outboard CIVs, electrical penetrations, mechanical penetrations, hatches, sealing mechanisms (welds, bellows, 0-rings), passive flooders, COPS, COPS CIVs, and others.

(2) With respect to instrumentation requirements, the staff believes that sufficient instrumentation should exist to inform operators of the status of the containment at all times. This instrumentation should also inform the status of the reactor during the early stages of the accident to ensure reactor failure at low pressure or to allow for low-pressure injection from the ac-independent water addition system.

As a minimum, the list of instrumentation identified below should be evaluated. Where extended ranges of operation of the instrumentation is needed, it should be identified along with the environment to which the instrumentation will be exposed.

The instrumentation is designed to survive the environment as specified in RG 1.97. However, RG 1.97 only ensures that the instrumentation will survive in the worst environment resulting from a design bases event and not from a severe accident. Therefore, engineering rationale must be developed as to why the instrumentation would survive the environment. This rationale could include such factors as limited time period in the environment; the use of similar equipment in commercial industry exposed to the same environment; the use of analytical extrapolations; or the results of tests performed in the nuclear industry or at national laboratories.

At least the following instrumentation should be evaluated: RPV water level, RPV pressure, sup pool temperature, sup pool level, DW/WW H_2 concentration, DW/WW O_2 concentration, DW temperature, WW pressure, WW temperature, and others.

In response to the open item, GE provided the environmental profiles, a table of the necessary equipment, and the accompanying rationale for ex-vessel severe accidents in Section 19E.2.1.2.3 of SSAR Amendment 34. The staff finds this information acceptable. Therefore, Open Item F19.2.3.3.7.2-1 is resolved.

19.2.3.3.7.3 Basis for Acceptability

GE developed a set of curves representing the bounding environmental conditions for both in-vessel and ex-vessel severe accidents. The environmental conditions were then compared to the equipment capabilities to provide a measure of confidence that the necessary equipment would survive the expected conditions. The staff concludes that the systematic process used by GE for assessing equipment survivability is acceptable and consistent with the assumptions used in GE's deterministic severe accident assessment. Further, the staff concludes that this meets the requirements of 10 CFR 50.34 discussed in Section 19.2.3.3.7.1, and the staff's proposed applicable regulation for equipment survivability.

19.2.3.3.8 Protection of Containment Sumps

The lower drywell contains two sumps: an equipment drain sump (EDS) and a floor drain sump (FDS). Figures 1.2-3b and 1.2-13e of the ABWR SSAR indicate that the sumps are embedded in the lower drywell floor with dimensions of approximately 1 m (3 ft) wide, 2 m (7 ft) long, and 1.25 m (4 ft) deep. Figure 1.2-3b of the ABWR SSAR indicates that the lower drywell has approximately 1.6 m (5.75 ft) of concrete protecting the liner, while SSAR Section 6.2.1.1.10.3 indicates that there are 1.5 m (5 ft) of concrete protecting the liner. Therefore, in the sump region, there is approximately 0.25 to 0.35 m (1 ft) of concrete protecting the containment liner. An accumulation of core debris within the sumps could lead to accelerated core-concrete interactions and, given the decreased thickness of concrete protecting the containment liner, the time to liner melt-through in the sump region from core-concrete interactions could be adversely affected.

To prevent liner melt-through in the sump region, the ABWR will have a protective layer of refractory bricks (corium shield) built around each sump to prevent corium ingression. The corium shield design is discussed in Sections 6.2.1.1.10.4 and 19ED of the ABWR SSAR.

19.2.3.3.8.1 Sump Design Criteria

The following general criteria included in Amendment 32, were developed by GE for designing the sumps:

- Corium shield height greater than maximum height of core debris bed
- Melting point of corium shield material above initial contact temperature
- Corium shield material having good chemical resistance to siliceous slags and reducing environments
- Seismic adequacy determined during the detailed design phase
- Shield roofs have provisions to allow water to flow into the sumps when the lower drywell is flooded.
- Shields extend to the floor of the sumps to prevent debris tunneling.

The EDS and FDS have different functions and therefore specific design criteria in addition to the above General Design Criteria (GDC) were developed. The specific design criteria are discussed below.

19.2.3.3.8.1.1 Equipment Drain Sump

The purpose of the EDS is to collect water leaking from valves and piping within the containment. The water enters and exits through piping from above the sump. As such, the following additional design criteria were specified in Amendment 32:

- Solid corium shield, except for the inlet and outlet piping through the roof
- Corium shield walls thick enough to withstand ablation

19.2.3.3.8.1.2 Floor Drain Sump

The purpose of the FDS is to collect water that falls onto the lower drywell floor. The water flows across the drywell floor and runs into the FDS at a height equal to the lower drywell elevation. As such, the following additional design criteria were specified in Amendment 32:

- The corium shield will have channels, which do not face the center of the lower drywell, at the lower drywell elevation to allow for water collection during normal operation.

- The channel length must ensure that debris will freeze before reaching the sump.

- The width and number of channels must ensure the required water flow rate during normal operation.

- Corium shield walls must be thick enough to allow residence time for debris solidification within the channels.

- The corium shield will extend beneath the lower drywell floor.

- The corium shield must be high enough to ensure long-term debris solidification.

19.2.3.3.8.2 Corium Shield Design

In Section 6.2.1.1.10.4.2, GE indicated in Amendment 32 that the corium shields are made of alumina with a height of 0.4 m (1 ft). The shield walls extend to the floor of the sumps.

For the FDS corium shield, GE analyzed in Amendment 32 the ability of the corium shield to initially freeze molten debris as it enters the channels and to transfer sufficient heat so that the debris remains solid in the long term. This analysis was in Sections 19ED.4 and 19ED.5 of the ABWR SSAR. It was used along with the GDC and the specific design criteria specified above to determine the actual corium shield design. The channels in the FDS corium shield are 1 cm (.4 in.) high and at least 0.5 m (1.64 ft) long; however, the width has not been specified.

19.2.3.3.8.3 Discussion

The staff stated in the advance SER that it believes that the sump shield designs proposed by GE have considerable merit and that some conservatism exists in the specified design criteria. For example, the design criteria are intended to ensure that no core debris enters the sumps. However, in actuality, the sumps could withstand limited amounts of core debris. In addition, GE in Amendment 32 did not take into account or factor in flooding the lower drywell with the LDF system or ac-independent water addition system.

Based on engineering judgment, the staff believes that the sump shields would prevent a substantial accumulation of core debris and that the channels within the FDS would lead to freezing of debris within them. However, the analysis provided to support the proposed shield designs in Amendment 32 was not sufficient to reach this conclusion. In particular, GE did not make use of existing experimental data and analytical tools in justifying its design in SSAR Amendment 33. This was Open Item F19.2.3.3.8.2-1.

In the advance SER, the staff stated that an acceptable resolution to this issue would entail the following:

(1) GE should evaluate related experimental and analytical work performed in this area to lend additional credibility to its design. In particular, GE should address how the results of its previous work supports its design. This would include a discussion of the prototypicality of the core debris, important parameters, and results. The staff has performed a quick review of related work and tools in this area and believes that they are relevant and readily available.

 (a) Experiments performed at (1) Kernforschungszentrum Karlsruhe (KfK) on ingression of molten debris into small cracks and openings, (2) Winfrith in the United Kingdom and (3) Grenoble in France.

 (b) Analytical tools such as PLUGM computer code (NUREG/CR-3190) and BUCOGEL computer code developed in France.

 (c) Work performed in the forging and casting industries.

(2) The analysis performed by GE in Amendment 32 for sizing the FDS evaluated an oxidic melt of around 2,227 °C (4,040 °F) and a eutectic melt of around 1,427 °C (2,600 °F). However, GE used the same correlations and key parameters for both, such as thermal conductivity and latent heat of fusion. To account for uncertainty in the progression of a severe accident and a range of material properties (such as density, melting point, and thermal conductivity), GE should perform separate analyses for oxidic, metallic, and eutectic melts clearly identifying the material properties and providing suitable references. In addition, GE should identify the parameters that the shields are most sensitive to (e.g., freezing point, heat of fusion, velocity of debris in channel, atmosphere temperature, and melt superheat). GE can use the results of its MAAP runs to identify the core debris composition at the time it enters the lower drywell.

In addition, GE could use the results of other code predictions (BWRSAR, MELCOR) as documented in NUREGs for similar BWRs.

(3) GE should address why the velocity of the debris in the FDS channels is not affected by the initial velocity of debris falling from the reactor pressure vessel (RPV)

(4) GE should modify the design criteria to do the following:

(a) Specify that the EDS extend below the lower drywell floor and that both shields prevent tunneling of core debris under them.

(b) Specify sloping of the shield roof to prevent accumulation of core debris or show that the long-term debris solidification in the channels is not affected by minor amounts of debris on the roof.

(5) GE should provide the thickness of the EDS corium shield necessary to withstand ablation.

GE has addressed these 5 issues in an SSAR markup of Sections 6.2.1.1.10.4 and Appendix 19ED dated February 7, 1994. Specifically, GE provided a markup of SSAR Section 19ED.6 providing an overview of related experimental and analytical work concerning the freezing of molten fuel in narrow channels to address Item 1. For Item 2, GE modified the analysis of channel length in Subsection 19ED.4 to account for three debris scenarios covering the expected range of melt phenomena. For Item 3, GE clarified the FDS shield design to clarify the location of channels. For Item 4, GE modified Subsections 19ED.3 and 19ED.5.2 and added 19ED.5.3 to establish that the EDS and FDS shields extend to the sump floor to prevent debris tunneling. Further, GE modified its long-term analysis in Subsection 19ED.5.1 to credit flooding of the lower drywell. Lastly, for Item 5, GE added Subsection 19ED.5.3 to address the thickness of the EDS shield walls and the shield wall of the FDS without channels. In addition, GE modified its general and specific design criteria for the EDS and FDS. This resulted in a revision to the design dimensions. Further, GE takes credit for the lower drywell flooder in determining the sump shield design. The staff concludes that the sump shield design proposed by GE is acceptable. This is based on GE's development of design criteria, proposed analytical solution, evaluation of the shield design to variations in key parameters, and review of existing related experimental and analytical work. GE has included the above information in SSAR Amendment 34 and the staff finds it to be acceptable. Therefore, Open Item F19.2.3.3.8.3-1 is resolved.

19.2.4 Containment Performance

The NRC approach for ensuring containment survivability from severe accident challenges consists of requiring inclusion of accident prevention and consequence mitigation features and the containment performance goal (CPG). The CPG ensures that the containment would perform its function in the face of most severe accident challenges and that the design (including its mitigation features) would be adequate if called upon to mitigate a severe accident.

Two alternative CPGs were identified in SECY-90-016: a conditional containment failure probability (CCFP) of 0.1 or a deterministic containment performance goal that offers comparable protection. In its June 26, 1990, SRM, the Commission approved the use of a 0.1 CCFP as a basis for establishing regulatory guidance for evolutionary LWRs. Two definitions of containment failure were discussed in SECY-91-309, "Draft Safety Evaluation Report on the General Electric Boiling Water Reactor Design Covering Chapter 19 of the Standard Safety Analysis Report, Response to Severe Accident Policy Statement." These include a CCFP based on a structural integrity definition and on a dose definition. For internal events, Section 19.1.3.5.1 of this report provides the results of the CCFP analyses and concludes that the ABWR design limits the CCFP to approximately 0.1. The treatment of external events for the ABWR is discussed in Section 19.1.2 of this report.

The Commission directed that the use of a 0.1 CCFP should not be imposed as a requirement, and that the use of the CCFP should not discourage accident prevention. Therefore, the staff's proposed applicable regulation for containment performance is as follows:

> The standard design must include design features to limit the conditional containment failure probability for the more likely severe accident challenges.

Section 19.1.3.5 of this report provides the staff's analysis of the design features that contribute to limiting the CCFP. The severe accident phenomena that are mitigated by these design features are evaluated in Sections 19.2.3.3 and 19.2.6 of this report. Based on the evaluations in these sections, the staff concludes that the acceptance criteria in SECY-90-016, SECY-93-087, and the staff's proposed applicable regulation for containment performance have been met.

19.2.5 Accident Management

The staff concluded, based on PRAs and severe accident analyses, that the risk associated with severe accidents could be further reduced through effective accident management (AM). AM encompasses those actions taken during the course of an accident by the plant operating and technical staff to (1) prevent core damage, (2) terminate the progress of core damage if it begins and retain the core within the reactor vessel, (3) maintain containment integrity as long as possible, and (4) minimize offsite releases. AM, in effect, extends the defense-in-depth principle to plant operating staff by extending the operating procedures well beyond the plant design basis into severe fuel damage regimes, using existing plant equipment and operator skills and creativity to terminate severe accidents and limit offsite releases.

In SECY-88-147 and Generic Letter 88-20, the staff identified the development of an "accident management plan" by each operating reactor licensee for severe accident "closure." A comprehensive description of the major goals, framework, and elements of an AM plan was subsequently provided in SECY-89-012, Staff Plans for Accident Management Regulatory and Research Programs. The AM plan provides a framework for evaluating information on severe accidents, for preparing and implementing severe accident operating procedures, and for training operators and managers in these procedures. An effective AM plan could reduce the risk associated with severe accidents by incorporating improvements in five general areas:

- Accident Management Strategies and Implementing Procedures
- Training in Severe Accidents
- Guidance and Computational Aids for Technical Support
- Instrumentation
- Delineation of Decision Making Responsibilities

In response, the nuclear industry has initiated an AM program, as described in SECY-90-313, Status of Accident Management Program and Plans for Implementation. Key issues to be resolved in establishing the AM program include industry completion and NRC review of (1) the industry-proposed process for evaluating AM capabilities, and (2) vendor-specific AM guidance. The industry-proposed AM program was scheduled for completion in 1993.

The overall responsibility for AM, including development, implementation, and maintenance of the AM plan, lies with the COL applicant, since the applicant is ultimately responsible for the safety of the plant and for establishing and maintaining an emergency response organization capable of effectively responding to potential accident situations.

The COL applicant should submit the AM plan as part of the COL application. The plan should provide the applicant's commitment to perform a systematic evaluation of the plant's ability to deal with potential severe accidents and to implement the necessary enhancements within the detailed plant design and organization. The staff will review the AM plan at the COL stage to assure that the evaluation process and commitments proposed by the COL applicant provide an acceptable means of systematically assessing, enhancing, and maintaining AM capabilities, consistent with staff expectations. The COL applicant would later implement the plan and submit the results for staff review before plant operation.

19.2.6 Capacity of the ABWR Primary Containment Vessel

19.2.6.1 Introduction

In Appendix 19F to Chapter 19 of the ABWR SSAR, the applicant (GE) discussed the ultimate capacity of its primary containment. The pressure boundary of the containment consists of the reinforced concrete containment vessel (RCCV) and steel torispherical upper drywell head (STUDH). The staff's evaluation of the adequacy of the containment to withstand the postulated design basis loads is provided in Sections 3.8.1 and 3.8.2 of this report. The purpose of this evaluation was to assess the containment's capability beyond the design basis.

19.2.6.2 Evaluation

The ABWR's containment consists of an RCCV, a right cylindrical structure built of steel-lined reinforced concrete, and a STUDH. In order to establish the ultimate capacity of the containment, GE discussed in Appendix 19F of its SSAR (Reference (Ref.) 1) the analyses performed for the RCCV. It also provided the calculated capacity of the STUDH and related the potential leak path under pressures and temperatures that could represent the environment inside the containment during severe accident conditions. This evaluation was based on a review of the information provided in Reference 1.

GE arrived at the following conclusions regarding the structural capability and functionality of the containment structure:

(1) The STUDH of the containment will have an internal pressure capacity of 770.1 kPa (97 psig) at

260.0 °C (500 °F) when the allowable stresses in the steel drywell head are held to the Level C Service Limit of the ASME Code Section III, Subarticle NE-3220 (Ref. 1, Section 19F.3.1.2).

(2) The pressure capacity of the STUDH at the median fragility (as defined by the conditional failure probability at 50 percent confidence level) is determined to be 1025.3 kPa (134 psig) at 260.0 °C (500 °F), and 928.7 kPa (120 psig) at 371.1 °C (700 °F) (Ref. 1, Section 19F.3.1.2).

(3) The equivalent ASME Level C Service Limit of the RCCV is 1232.1 kPa (164 psig); the ultimate capacity of the RCCV, being higher, is not estimated by the applicant (Ref. 1, Sections 19F.1, 19F.2, and 19F3.1.1).

(4) Liner plate and its anchorages will maintain their integrity when subjected to a severe accident pressure of 1025.3 kPa (134 psig) and a temperature of 260.0 °C (500 °F) (Ref. 1, Section 19F.3.2.1).

(5) The total leak area through various penetrations at 928.7 kPa (120 psig) is estimated to be 67.55 cm^2 (10.47 in^2) (Ref. 1, Section 19F.3.2.2).

The acceptability of GE's five conclusions is discussed and evaluated by the staff in the following sections.

19.2.6.2.1 Level C Service Limit

19.2.6.2.1.1 Steel Torispherical Upper Drywell Head

Membrane Stress Intensity

On the basis of NASTRAN analysis and the stress intensity criterion provided in the ASME Code, Section III, Paragraph NE-3221, GE stated that the general primary membrane stress controls the design at the Level C Service Limit for an internal pressure of 832.2 kPa (106 psig) at 171.1 °C (340 °F). At 260.0 °C (500 °F), the allowable internal pressure at the Level C Service Limit for membrane stress intensity was found to be 770.1 kPa (97 psig). To verify GE's results concerning the stresses using NASTRAN computer code, the staff independently calculated the membrane stresses using ALGOR computer code. The staff's comparison (Table 19.2-1) confirmed the applicant's conclusion.

Buckling

Galletly (Ref. 2) developed, on the basis of the actual test data, a simple parametric equation to calculate limiting internal pressure that would prevent buckling of steel torispherical heads. Based on the equation and the analysis of the data in Reference 2, GE calculated a best-estimate internal pressure value of 1838.8 kPa (252 psig) and a lower bound value of 1245.9 kPa (166 psig) as values corresponding to buckling failure values of torispherical heads tested.

When test data are used to establish the ASME Code Section III, NE-3222, Allowable Buckling Stress, it is appropriate to use the best estimate test data. Accordingly, when the best-estimate test data values are used in NE-3222, the allowable internal pressure at the Level C Service Limit for buckling stress is 797.7 kPa (101 psig). An alternate method of computing the allowable buckling stress is provided in ASME Code, Section III, Code Case N-284. The factor of safety at the Level C Service Limit in Code Case in N-284 (1.67) is less than that in NE-3222 (2.5). However, the factor of safety has to be applied to the lower-bound test data. When the lower-bound test data are used in Code Case N-284, the allowable internal pressure at the Level C Service Limit is 783.9 kPa (99 psig). This evaluation is based on the staff's position on the shell buckling as a result of internal pressure (See Appendix A to Section 3.8.1 of this report).

From the above three limiting pressure values of 770.1 kPa (97 psig) based on membrane stress intensity criterion, 797.7 kPa (101 psig) from buckling criterion of NE-3222, and 783.9 kPa (99 psig) from buckling criterion of N-284, the staff finds that because the 770.1 kPa (97 psig) is the least of the pressure values, it controls the internal pressure capacity of STUDH corresponding to the ASME Level C Service Limit stress criterion. Thus, the staff finds the proposed internal pressure of 770.1 kPa (97 psig) as an acceptable value for the STUDH.

19.2.6.2.1.2 Concrete

For the concrete portion of the RCCV, the staff considers the factored load acceptance standards of ASME Code Section III, Division II, Article CC-3000 as an appropriate acceptable criterion. It should be noted that in applying the criterion, a factor of 1.0 (instead of 1.5) should be Table CC-3230-1 for the internal pressure generated by severe accidents. The pressure capability estimate for the RCCV using this criterion is discussed in the following paragraphs.

Table 19.2-1 GE ABWR drywell head comparison of membrane stresses

75 psig internal pressure, 0.03175 m (1.25 in.) thickness, 171 °C (340 °F)

	Hoop MPa (psi)	Longitudinal MPa (psi)	Intensity MPa (psi)	Remarks
NASTRAN	101.5 (-14,725)	59.57 (8,640)	161.1 (23,365)	5° wedge
ALGOR[1]	101.0 (-14,650)	58.05 (8,420)	159.1 (23,070)	5° wedge 1st quadrant
ALGOR	92.91 (-13,475)	59.50 (8,630)	152.4 (22,105)	10° wedge Full model

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From NASTRAN,

Stress intensity, P_m: 161.1 MPa (23,365 psi) @ 0.618 MPa (75 psig) and 171 °C (340 °F)

Yield strength (S_y) (SA-516, Gr. 70): 229.7 MPa (33,300 psi) @ 171 °C (340 °F)

211.7 MPa (30,700 psi) @ 260 °C (500 °F)

Allowable stress intensity (S_{mc}) (SA-516, Gr. 70): 133.1 MPa (19,300 psi)

Level C Service Limit Criteria (NE-3221): $P_m \leq$ max. (1.0 S_y, 1.2 S_{mc}),

$P_m \leq$ 229.7 MPa (33,300 psi) @ 171 °C (340 °F)

$P_m \leq$ 211.7 MPa (30,700 psi) @ 260 °C (500 °F)

Allowable pressure = (75)*(33,300)/(23,365)

= 0.832 MPa (106 psig) @ 171 °C (340 °F)

= (106)*(30,700)/(33,300)

= 0.770 MPa (97 psig) @ 260 °C (500 °F)

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[1] ALGOR is a three-dimensional finite element program for structural analysis and design and has been used by the staff for its independent evaluation.

Initially, in Amendment 21 of Ref. 1, the applicant provided the information related to the RCCV ultimate capacity based on the scaled model test of the RCCV. Later, in Amendment 30 of Ref. 1, to avoid inclusion of proprietary information in the SSAR and recognizing the differences in the model and the proposed RCCV (having a significantly higher percentage of reinforcement) GE decided to perform analyses of the RCCV for severe accident conditions. The FINEL computer program developed by Bechtel Corporation is used for the nonlinear finite element analysis of the RCCV. In the analysis, the RCCV and the internal structures are considered as axisymmetric, and the reinforced concrete girders integral with the top slab of the RCCV are represented by solid elements with appropriate stiffness. The pressure capacity of the top slab is calculated based on the extrapolation of the elastic 3-D STARDYNE analysis results.

Within the elastic range of response, both codes have been well recognized and benchmarked against the structural integrity tests of containments. Though the FINEL computer program permits the specification of bilinear, brittle, and ductile material properties, it is not benchmarked against results of any containment test with nonlinear responses. However, within the pressure range of interest, the RCCV responses are in the linear range and hence the use of these codes to compute pressure capacities of the RCCV is reasonable.

The results of the combined analyses indicate that the weakest link in the RCCV is the top slab where the equivalent ASME Level C Service Limit acceptance criterion is reached at a pressure of 1232.1 kPa (164 psig).

Thus, the staff finds the containment performance under an internal pressure of 770.1 kPa (97 psig) to be acceptable and is aware that it is limited by the capacity of the STUDH. The RCCV has higher capacity and does not limit the containment performance.

19.2.6.2.2 Median Fragility Level for the Containment

19.2.6.2.2.1 Steel Torispherical Upper Drywell Head

GE stated that the limit pressure for plastic deformation was found to be 940.4 kPa (121.7 psig) at 260.0 °C (500 °F) by Shield and Drucker's proposed equation (Ref. 3). The minimum yield strength of material SA-516, Gr. 70, as specified in Appendix I of ASME Section III was increased by 10 percent for the realistic estimate of the structural strength. Thus, the limiting internal pressure is determined to be 1025.3 kPa (134 psig) at 260.0 °C (500 °F) and 928.7 kPa (120 psig) at 371.1 °C (700 °F) to account for the lower yield strength of the material.

From the STUDH buckling capability, the applicant predicted a best-estimate internal pressure value of 1838.8 kPa (252 psig) and a lower bound value of 1245.9 kPa (166 psig) as values corresponding to buckling failure of the STUDH. These values are based on the results of the tests (Ref. 2) performed on stainless steel and carbon steel torispherical heads fabricated using the pressed and spun (PS) technique as well as the crown and segment technique (CS). Out of 43 tests, six data points were from the actual failure resulting from internal pressure on the carbon steel CS heads. These are the most relevant data for the drywell head. The lower-bound value is estimated to be 1245.9 kPa (166 psig). As these values are based on a limited data base and as they correspond to actual buckling failure of the heads, use of them to arrive at the actual median fragility internal pressure for the STUDH should be made with some margin on the lower bounds. Providing an arbitrary knockdown factor of 1.2, the internal buckling pressure of the median fragility level can be derived as 1052.8 kPa (138 psig). However, GE adopts 1025.3 kPa (134 psig) as the capability pressure and uses it as the median fragility value at 260.0 °C (500 °F). Based on this median fragility value of 1025.3 kPa (134 psig), a containment pressure capacity fragility curve was provided with the following uncertainty parameters in Appendix 19FA to Reference 1.

In Appendix 19FA to Reference 1, the applicant estimated the uncertainties associated with the median fragility value using engineering judgment and the results from prior analysis. The uncertainties in the prediction of the failure pressure generally result from uncertainties in modeling and material strength. The lognormal distribution is selected to characterize the fragility curve and defined in terms of the median pressure capacity and the combined logarithmic standard deviation. The logarithmic standard deviations from uncertainties for modeling and material properties of steel structures (β_m and β_s) are estimated as 0.14 and 0.08, respectively, and the combined logarithmic standard deviation (β_c) is estimated to be 0.16.

The use of 0.14 for β_m is acceptable because it is consistent with Reference 4 in which the coefficient of variation (COV) [defined as $(\exp(\beta^2)-1)^{1/2}$] associated with the modeling error by the use of approximate methods including torispherical heads is 0.12.

The use of 0.08 for β_s is acceptable because the variability associated with material strength is expected to be the same regardless of temperature (Ref. 5). The statistical data for SA-516, Gr. 70, show that the average yield strength of the material is 48.62 ksi and the standard deviation is 3.525 ksi (Ref. 6). The COV is 0.073, which is less than 0.08 used for β_s.

Therefore, the combined logarithmic standard deviation (β_c) of 0.16 ($(\beta_m^2 + \beta_s^2)^{1/2}$) is acceptable.

Based on the staff's evaluation and by taking the above factors into account, the staff considers the median fragility value for the STUDH of 1025.3 kPa (134 psig) at 260.0 °C (500 °F) with the combined logarithmic standard deviation of 0.16 to be reasonable. If the temperature reaches 371.1 °C (700 °F), the median fragility internal pressure value is reduced to 928.7 kPa (120 psig).

19.2.6.2.2.2 Concrete

As discussed in A.2 above, the equivalent Level C Service Limit internal pressure capacity of the RCCV is estimated as 1130 kPa (164 psig). As this capacity is higher than the median fragility internal pressure capacity of the STUDH, GE decided not to spend resources in estimating the ultimate (median fragility) pressure capacity value for the RCCV. The staff considered this approach acceptable provided GE demonstrated the integrity of the RCCV liner plate and its anchorages under 1025.3 kPa (134 psig) and 260.0 °C (500 °F) temperature.

In Section 19F.3.2.1 (Ref. 1), GE provided an evaluation of the liner plate and its anchorages under the specified conditions. The applicant determined the maximum hoop strain in the liner plate at 1025.3 kPa (134 psig) to be 0.13 percent. With a strain concentration factor of 33 to account for a failure mode similar to that in the Sandia test (Ref. 7), GE uses the argument that the resulting strain in the ABWR RCCV is 4.3 percent, which is significantly lower than the ultimate tensile strain (i.e., 21 percent) of the liner plate material. The staff recognizes that the geometry and type of anchors to be used for the ABWR RCCV are quite different and the strain concentration factors may be different between the Sandia test and the ABWR. Nevertheless, a factor of 33 is conservative and the staff finds it acceptable.

Additionally, GE evaluated the effects of thermal loading on the liner and the anchorages using Bechtel's topical report, BC-TOP-1, Rev. 1, Containment Building Liner Plate Design Report. For this evaluation, the temperature inside the RCCV is considered as 260.0 °C (500 °F), and that outside the RCCV is assumed as 37.8 °C (100 °F). The liner is shown to be buckled (because of temperature-induced compressive stresses) between the continuous anchorages, but the force on the anchorages is shown to be less than its yield capacity.

Although the staff finds this approach reasonable, it should be recognized that there is a difference in the duration and magnitude of the temperature loading that would be applied during a design-basis accident (DBA) (which is of a smaller and magnitude shorter duration) compared with that postulated to occur during a controlling severe accident (SA) scenario. Also, the condition of the RCCV structure would be different in the two cases. The pressure loading during an SA would reduce the stiffness of the structure considerably. Thus, the assumptions used in the BC-TOP-1 report are not necessarily applicable. However, they provided a basis for calculating an upper-bound value for compressive buckling of the plate and for the forces on the liner anchorages. Alternatively, under an SA, the liner together with the extensively cracked concrete can be considered to grow because of high (260.0 °C (500 °F)) temperature. Considering the differential temperature of 204.4 °C (400 °F), the staff calculated the tensile strain in the liner to be 0.24 percent. The combined tensile strain in the general shell liner could be as high as 0.37 percent. When the same strain concentration factor of 33 is used again, the maximum tensile strain in the liner at a discontinuity could be as high as 7.9 percent. This strain is 40 percent of the ultimate tensile strain of the liner material. Under this condition, the continuous anchorages could detach themselves from the cracked concrete, but, the liner would still provide the required leaktightness.

Thus, the staff finds the median fragility internal value of 1025.3 kPa (134 psig) at 260.0 °C (500 °F) and 928.7 kPa (120 psig) at 371.1 °C (700 °F) acceptable as limiting values.

19.2.6.2.3 Interface Between STUDH and RCCV

GE performed the pressure capacity calculations for the interface between the STUDH flanges and the top slab of the RCCV and arrived at the following pressure capacities: (a) 1176.9 kPa (156 psig) for the maximum allowable concrete peripheral shear stress of the top slab using the acceptance criteria of ASME Section III, Division II, CC-3421.6, (b) 1687.1 kPa (230 psig) for anchorage steel ring plate using the Level C acceptance criteria of ASME Section III, Division I, NE-3221, and (c) 1528.5 kPa (207 psig) for the anchorage steel gusset plates using the Level C acceptance criteria of ASME Section III, Division I, NE-3227.2. The staff finds the pressures to be above the median fragility value of 1025.3 kPa (134 psig), with the stresses in the interface area well within the ASME Section III Code allowables, and thus acceptable. Therefore, the staff agrees with GE's conclusion that the probable leakage path would be through the STUDH flanges. The leakage through the STUDH flanges is discussed in Section 2D of this report.

19.2.6.2.4 Leakage Through Penetrations

In Section 19F.3.2.2 (Ref. 1), GE discusses the leakages through various RCCV penetrations and from the seal area between the flanges in the STUDH under severe accident temperatures and pressures. The assumptions and rationale used in the discussions by the applicant are based upon the experimental work performed at SNL and Argonne National Laboratory (ANL). GE determined that under the postulated SA scenarios, the leakage through the fixed electrical and mechanical penetrations was negligible.

In a facsimile dated September 7, 1993, GE revised Subsection 19F.3.2.2 of the SSAR to discuss the leakage performance of the containment EPAs under conditions. GE stated that the EPAs to be used in the ABWR containment would be capable of maintaining leaktightness up to the containment pressure of 1025.3 kPa (134 psig) at 371.1 °C (700 °F). The staff finds GE's assessment regarding the fixed electrical and mechanical penetrations acceptable. GE has included this information in Subsection 19F.3.2.2 of the SSAR.

The containment has five operable penetrations: two pressure-seating airlocks, two pressure unseating equipment hatches, and the pressure-unseating STUDH. In Section 19F.3.2.2 of Ref. 1, GE estimated the leakage areas from the three pressure-unseating penetrations with conservative assumptions; such as, (1) the seal is assumed lost at 260.0 °C (500 °F), and (2) the springback capability of degraded seals was not factored in. The total leakage area is estimated as 110 percent of the leakage areas of the three pressure-unseating penetrations to account for the potential for small leakages through the airlocks. The leakage areas are provided for the pressures above the structural integrity test pressure at an interval of 170.3 kPa (10 psig). The total leakage area at 928.7 kPa (120 psig) is estimated as 67.5 cm^2 (10.47 in^2).

The staff considers the estimate of these leakage areas to be reasonable.

19.2.6.3 Summary and Conclusion

Based on the review of the information GE provided and the staff's evaluation as discussed above, the staff concludes the following:

(1) The staff evaluation indicates that the ABWR containment structure has an internal pressure capacity of 770.1 kPa (97 psig) and 260.0 °C (500 °F) when the allowable stresses in the steel drywell head are held to the Level C Service Limit.

(2) As discussed in 2B of the evaluation, the staff considers the median fragility of the containment structure of 1025.3 kPa (134 psig) at 260.0 °C (500 °F) to be a conservative value.

<u>References</u>:

1. Appendix 19F to Chapter 19 of the General Electric Nuclear Energy, Advance Boiling Water Reactor Standard Safety Analysis Report (up to and including Amendment 34).

2. Galletly, G. D. "A Simple Design Equation for Preventing Buckling in Fabricated Torispherical Shells Under Internal Pressure," ASME Journal of Pressure Vessel Technology, Vol. 108, November 1986.

3. Shield, R. D., Drucker, D. C. "Design of Thin-Walled Torispherical and Toriconical Pressure-Vessel Heads," Transactions of ASME, June 1961.

4. NUREG/CR-2442, "Reliability Analysis of Steel-Containment Strength," June 1982.

5. NUREG/CR-5405, "Analysis of Shell-Rupture Failure Due to Hypothetical Elevated-Temperature Pressurization of the Sequoyah Unit 1 Steel Containment Building," February 1990.

6. NUREG/CR-2137, "Realistic Design Margins of Pumps, Valves, and Piping," June, 1981.

7. NUREG/CR-5341, "Round-Robin Analysis of the Behavior of a 1:6-Scale Reinforced Concrete Containment Model Pressurized to Failure: Posttest Evaluations," Sandia National Laboratory, October 1989.

19.3 Shutdown Risk

19.3.1 Introduction

Various incidents occurring at nuclear power plants during low power and shutdown operation modes over the past several years have raised NRC staff concerns regarding plant vulnerability during these operating modes. The shutdown events have caused plants to lose their ability to maintain core cooling, provide make-up coolant to the reactor, and to maintain electrical power to essential equipment. The April 10, 1987, event at the Diablo Canyon Nuclear Power Plant emphasized the sensitivity of operating a pressurized-water reactor (PWR) with a reduced inventory in the RCS. Following an evaluation, the staff issued Generic Letter 88-17, "Loss of Heat Decay Removal," on October 14, 1988, requesting PWR licensees to address numerous generic deficiencies to improve safety during operation at reduced inventory.

On March 20, 1990, Vogtle Unit 1 experienced a complete loss of decay heat removal (DHR) capability during shutdown and refueling operations from a loss of the required offsite ac source and failure of the available onsite diesel generator to provide and maintain power to safety-related buses. In June 1990, the staff issued NUREG-1410, Loss of Vital AC Power and the Residual Heat Removal System During Mid-Loop Operations at Vogtle Unit 1 on March 20, 1990. In the NUREG, the staff discussed the loss of vital ac power and DHR capability during midloop operations and the need to manage risk for shutdown operations.

These events, as well as others, prompted the staff to begin a comprehensive review of low-power and shutdown operations including hot shutdown, cold shutdown, and refueling at all nuclear plants and other shutdown-related issues identified by foreign regulatory organizations and the NRC. The objective of the review was to assess risk during shutdown, refueling and low-power operation. In February 1992, the staff issued Draft NUREG-1449, Shutdown and Low Power Operation at Commercial Nuclear Power Plants in the United States, for public comment to document technical findings associated with shutdown conditions. In September 1993, the staff issued NUREG-1449 final report, which incorporated and responded to comments received on draft NUREG-1449. The staff is using these technical findings to prepare appropriate regulatory actions to address shutdown risk.

The fundamental conclusion of NUREG-1449 is that public health and safety have been adequately protected while plants were in shutdown conditions. However, numerous and significant events have indicated that substantial safety improvements should be made in the areas of:

- outage planning and control
- fire protection
- TS
- instrumentation.

The staff utilized the technical findings and insights from NUREG-1449 in its safety evaluation of the ABWR.

19.3.2 Evaluation Scope

On January 12, 1990, the Office of the Secretary of the Commission issued SECY-90-016, Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements. In this Commission paper, the staff identified reduced inventory operation and a number of other issues significant to reactor safety that were considered fundamental to the agency's decisions on the acceptability of evolutionary ALWR designs. On July 12, 1993, the staff submitted SECY-93-190, Regulatory Approach To Shutdown and Low-Power Operations, in which it discussed the advantages and disadvantages of a proposed rulemaking to establish new regulatory requirements for shutdown and low-power operations in the following areas: outage planning and control, TS, fire protection, and instrumentation. The proposed new regulatory requirements, which will be developed further as part of the shutdown risk rulemaking process, were a result of technical findings and insights from NUREG-1449.

Based on the above, the NRC requested GE, as part of the design review process for the ABWR, to perform a systematic examination of shutdown risk, including evaluation of specific ABWR design features that minimize shutdown risk, quantification of the reliability of the DHR systems, identification of any vulnerabilities introduced by new design features and consideration of fires and floods with the plant in modes other than full power.

GE submitted its Shutdown Risk Evaluation on July 2, 1992, for staff review. In this report, GE stated that a total of 70 loss-of-power and over 100 loss- of-DHR events that occurred at operating BWRs were reviewed and evaluated from NUREG-1410, NSAC-88, Residual Heat Removal Experience Review and Safety Analysis — Boiling Water Reactor, Institute of Nuclear Power Operation (INPO) reports, and NRC information notices. GE stated that the ABWR design features would prevent or mitigate the most safety significant of these events. GE also addressed what it believed to be all shutdown risk issues, concerns, and design features to minimize risk associated with low-power and shutdown operations.

GE evaluated the ABWR design for risks associated with plant conditions in Modes 3 (hot shutdown), 4 (cold shutdown), and 5 (refueling), as well as areas that are considered operational improvements for shutdown risk operation which will be further addressed by COL holders implementing guidance provided by the reactor vendor (e.g., outage planning and control and operator training and procedures). Other shutdown risk concerns, such as DHR capability during low-power and shutdown operation, are addressed by specific design features and TS.

The staff evaluated this submittal based on technical findings and insights from NUREG-1449, a number of studies from the international community, and PRA of shutdown and low-power operating modes for a BWR to screen for important accident sequences. The staff also considered reports on operating plant events associated with shutdown operation and staff-sponsored research issues associated with low-power and shutdown operation.

The purpose of the staff review is to ensure that the ABWR design has appropriately addressed the shutdown risk concerns based on experience with operating plants, including appropriate vendor guidance for COL applicants in areas of outage planning and control, fire protection, and instrumentation. Design improvements and/or design modifications GE identified were reviewed to ensure insights from shutdown operation experiences were addressed and that the design improvements reduce the likelihood of core damage and enhance public health and safety. Also, the staff evaluated vulnerabilities that may result from new design features; DHR capability; treatment of fires and floods with plant in modes other than full power; and related technical findings discussed in NUREG-1449.

Therefore, the staff's proposed applicable regulation for shutdown risk is as follows:

> The application for design certification must include a systematic examination of shutdown risk including an assessment of:
>
> (1) specific design features that minimize shutdown risk;
> (2) the reliability of decay heat removal systems;
> (3) vulnerabilities introduced by new design features; and
> (4) fires and floods with the plant in modes other than full power.

These items are discussed in the sections below.

19.3.3 ABWR Design Features Minimizing Shutdown Risk

GE stated that the risks associated with shutdown operations have been reduced in the ABWR design by such features as the enhanced DHR system, improved electrical systems, instruments that give important safety parameters during shutdown, and alternate features that maintain core cooling in case of a loss of DHR. The applicant concluded that ABWR TS and utility operating and maintenance procedures will ensure that, during shutdown conditions, the ABWR is adequately protected against accidents.

19.3.3.1 Decay Heat-Removal Capability

In NUREG-1449, the staff stated that DHR can be lost at shutdown because of a loss of flow in the RHR system or a loss of an intermediate or ultimate heat sink caused by the loss of electric power or by valve failures. Events during shutdown can lead to fuel's being uncovered and damaged. Past events have led to interrupted shutdown cooling (SDC) at BWR operating plants because of a loss of power to the RPS logic that caused the RHR system isolation valves to fail closed.

The applicant described the ABWR DHR capability in SSAR Section 19.Q.4.1 of the ABWR PRA Shutdown Risk Final Report. The DHR capability consists of several features that minimize the loss of DHR. One of these features consists of the three independent divisions of RHR systems and is the first line of defense in maintaining DHR. Three suction lines for shutdown cooling will be connected directly to the reactor pressure vessel (RPV). This is an improvement over the current BWR designs in which one common shutdown cooling line draws suction from the external reactor recirculation loops. This single suction line in current BWRs is more vulnerable to a loss of shutdown cooling by single failure of valves in the suction line. The RHR shutdown cooling return for RHR subsystem A will be routed to the RPV vessel through the feedwater system. RHR shutdown cooling return for RHR subsystems B and C will be routed directly to the RPV. If a single failure occurred in the operating RHR loop while a second loop was in maintenance, the third loop could be placed in service. Under certain conditions, all three RHR loops or any two RHR loops would be run in parallel. In these cases, failure of one loop would not result in a loss of RHR. The ABWR DHR systems include a mode selector control for choosing among five modes of operation: low pressure flooding, suppression pool cooling, shutdown cooling, wetwell spray, and drywell spray. If any one of these modes is selected, the DHR systems will automatically align the valves as required for the mode selected. This feature will reduce the chance of operator error from incorrectly aligning the required valves, thus increasing the availability of the DHR capability during shutdown operations.

The applicant indicated that a loss of power to the RPS will not result in isolation of the shutdown cooling (SDC) system. A loss of power to the multiplexed safety system logic would cause each SDC isolation valve to fail in its current position (fail as-is), thus preventing closure of the CIVs between the RHR system and the RCS. Shutdown cooling is thereby maintained.

Upon receiving a low RPV water-level signal, RHR shutdown cooling isolation valves will close to stop all flow out of the RPV. The RPV low-level set point is 3.18 m (10.43 ft) above the top of the active fuel. If the RPV low-level isolation feature fails, inventory loss resulting from flow out of the RPV will stop when the RHR suction cooling nozzle is uncovered. At this point, 1.7 m (5.58 ft) of water will remain above the top of the active fuel. The applicant will design two-out-of-four logic to control the initiation of ABWR RPV water-level signals. This will reduce the likelihood of a false loss-of-coolant accident (LOCA) signal.

The staff finds these provisions acceptable and concludes that GE has appropriately addressed concerns in NUREG-1449 related to inadvertent loss of RHR.

19.3.3.1.1 ABWR Alternate Decay Heat Removal Features

In NUREG-1449, the staff stated that in case of a loss of RHR, BWR operators can significantly extend the time available for recovering the system by adding water to the core from several sources, including the condensate system and the CRD system, to raise water to a level that can support natural circulation and continue to remove decay heat, thus reducing the probability of damage to the core.

The applicant discussed the alternate DHR capability for the ABWR in SSAR Table 19.Q-1 of the ABWR PRA Shutdown Risk Final Report. In the event that RHR is not available, operators can use several non-safety-related systems as alternate methods to remove decay heat:

- Main steam SRVs can be used to vent steam to the suppression pool thus depressurizing the RPV and allowing the use of other low-pressure systems.
- During shutdown, the reactor water cleanup (CUW) system can be used under certain conditions to remove decay heat. Water is moved through a line attached to the RPV bottom head, through a series of heat exchangers and filter demineralizers and then returned to the RPV through an attachment to the upper head or the feedwater lines.
- The fuel pool cooling (FPC) system can be used for DHR during Mode 5 (refueling).
- RPV water boiling with the vessel head off is an effective way to remove heat but is not a preferred method because of the potential for offsite releases. However, it can be used as long as the RPV level can be maintained by available make-up sources.

The staff asked GE to discuss the impact of direct RPV water boiling to the containment as an acceptable alternate DHR method. In a letter dated January 13, 1993, GE stated that analysis results indicate that offsite doses from direct boiling in the RPV during Mode 5 will be much lower than required by regulatory limits of 10 CFR Part 100. Component operability is ensured because the DHR components will be qualified for a harsh environment. Reliable components such as the CRD pumps are expected to survive for a significant period of time in a low-pressure steam environment to support alternate make-up capability. Minimum operator actions required to initiate direct boiling of RPV water with the head removed include opening three manual valves inside secondary containment and actuating the ACIWA system. Since the time to reach steam boiling is several hours after DHR cooling is lost, the operator actions to manually open the valves are reasonably assured. The staff concludes that direct boiling in the containment can be used as an alternate DHR method, but should be used only after attempts to restore other DHR methods, such as use of the FPC system, have been unsuccessful.

The ABWR design will enable operators to cool the core using alternate DHR methods as described above. The loss of RHR during shutdown can be responded to as long as non-safety-related equipment used for the alternate methods is made available and clear procedures have been prepared for applying the methods. Maintenance of the decay heat capability and procedures are discussed in Section 19.3.7.2 of this report. The staff concludes that the applicant has sufficiently addressed concerns in NUREG-1449 regarding the capability to provide alternate core cooling in case of a loss of RHR.

19.3.3.2 Inventory Control

In NUREG-1449, the staff stated that loss of inventory is more likely to occur during shutdown or refueling than during normal operating conditions because of system repairs, maintenance, and component replacement. Activities such as test and maintenance during shutdown that require seldom-used valve line-ups and plant configurations increase the probability of operator error associated with inventory control. Loss of inventory can lead to fuel damage by overheating if no make-up water is added to the core.

GE discussed the ABWR inventory control in SSAR Section 19.Q.4.2 of the ABWR PRA Shutdown Risk Final Report. In the event that the RPV level decreases during shutdown conditions, Modes 4 and 5, the ABWR design will automatically initiate, on low reactor water level, the high-pressure core flooder (HPCF), and the low-pressure flooder (LPFL) systems to inject water from the suppression pool. These features are part of the ECCS, and the TS require automatic actuation of the ECCS to be operable in these modes. Other manually initiated systems could also be used to inject water from the main condenser hotwell, CRD system, and ACIWA system. These features will be part of the ABWR alternate DHR capability and the COL applicant will develop administrative control procedures, with appropriate guidance from the reactor vendor, to ensure that alternate DHR capability is available. All isolation valves used in the shutdown cooling mode of RHR and CUW, except those in the injection lines, will automatically close on a low RPV water level signal to isolate the inventory losses. If the HPCF and LPFL systems are in the test mode and an RPV low-level signal is received, the systems will automatically switch to the vessel injection mode. Additionally, the ABWR design uses reactor internal pumps that eliminate recirculation piping external to the RPV, hence the probability of LOCAs during normal and shutdown operations will likely be reduced.

To further reduce the likelihood of a loss of inventory, RHR system valves will be interlocked with the reactor system pressure to ensure that low-pressure RHR piping will not be exposed to full system pressure. The low-pressure portions of RHR piping are designed to withstand the full reactor pressure without rupture in the event that the interlocks fail or are bypassed.

In NUREG-1449, it was noted that draindown of the RPV water to the suppression pool could occur if the motor-operated valves in the RHR system inadvertently opened or by operator errors. The ABWR design includes provisions for preventing inadvertent draining of the RPV to the suppression pool. Interlocks require that the RPV shutdown cooling suction valve be fully closed before the suppression pool return or suction valves can be opened. In the reverse situation, interlocks require that suppression pool return or suction valves must be fully closed before SDC suction valve can be opened. The permanently installed RPV level indication system will give level indications and alarms to operators in the control room during shutdown operations.

19.3.3.2.1 Temporary Reactor Coolant System Boundaries

19.3.3.2.1.1 Use of Freeze Seals in ABWR

The RCS in the ABWR design includes significantly less piping than does the RCS in currently operating BWR designs. For example, the design includes no external recirculation loops, and the RHR piping connected to the RPV will enter at a higher level than the top of the active fuel. Therefore, inadvertent draining from these lines will stop without exposing the fuel. The ABWR will contain no pipes larger than 5.08 cm (2 in.) in diameter below the core, thus allowing operators more time to recover coolant inventory if coolant is lost as a result of maintenance, valve failures, or pipe breaks. However, freeze seals are sometimes used and are discussed in SSAR Section 19.Q.8 of the ABWR PRA Shutdown Risk Final Report. Freeze seals are often used to isolate system piping for the repair and replacement of components such as valves, pipe connections, pipe fittings, and pipe stops, where it is impossible to perform the task without isolating the piping system in such a way. The use of freeze seals to perform such activity in the RCS boundary is of concern because a failed seal would compromise the structural integrity of the boundary, thus creating a loss of inventory control. In NUREG-1449, the staff discussed events in which a freeze seal used in secondary system equipment failed, resulting in a loss of inventory, causing flooding, and rendering equipment inoperable. These incidents occurred because plant personnel failed to follow the procedures and properly maintain the freezing media.

GE stated that the COL administrative procedures will ensure the integrity of the temporary boundary when freeze seals are used. Mitigative measures will be identified in advance, and appropriate back-up systems will be made available to minimize the effects of a loss of coolant inventory.

The staff considers this a COL action item and will ensure that COL applicants provide guidance on controlling and maintaining the integrity of freeze seals. The guidance should address the use of an engineering safety analysis on a case-by-case basis to ensure that the use of freeze seals, where a failure could result in loss of inventory, will not result in any unresolved safety review questions. GE has included this COL action item in SSAR Amendment 34 and the staff finds it to be acceptable.

19.3.3.2.1.2 Reactor Internal Pump Motor and Impeller Replacement

The ABWR reactor internal pumps (RIPs) are used to supply coolant circulation and to replace the external coolant recirculation system used in the BWR designs. This is a design improvement over the BWR designs in which an unisolable pipe break or component repair in the external recirculation system could result in a major loss of inventory control. The RIP concept was adopted from European BWRs that have been operating for more than 15 years and have had no indications of difficulty in maintenance or in operation that resulted in a loss of inventory. GE discussed the procedures to maintain and to replace the RIPs in SSAR Section 19.Q.4.2 of the ABWR PRA Shutdown Risk Final Report.

Removal of the RIP motors for maintenance is accomplished by using integral inflatable seals that act as backup sealing devices to assure that no RPV water leakage occur. Following each motor removal, a temporary cover plate is bolted to the bottom of the motor's housing, which forms part of the reactor vessel. The impeller is then removed from the top. Upon the removal of the impeller, the bolted cover plate acts as the RPV boundary and prevents leakage of reactor water. A plug is then installed on the RPV bottom head at the impeller nozzle to provide additional protection against draining the RPV.

The staff asked GE to discuss the effect the increased temperature from a loss of DHR cooling during RIP replacement or maintenance could have on inflatable seals. In a letter dated January 13, 1993, GE stated that the inflatable seals on the RIP shaft will be permanently installed and will be designed to handle normal operating temperatures. Increased temperatures from a loss of DHR cooling will not affect the performance of the seals since the coolant temperature during shutdown conditions will be less than the design temperature of the seal.

The staff reviewed the RIP maintenance and replacement sequences and found that if leakage occurred during the removal of the RIP motor, the temporary cover plate could be installed to eliminate RPV water leakage. The pump internal primary and inflatable secondary seals would minimize the potential RPV water drainage, thus making the RIP motor removal process acceptable. However, the staff noted that during the RIP impeller and shaft removal, possible unisolable LOCA with an opening of about 20.32 cm (8 in.) exists in the event that operators failure to follow the maintenance procedure or possibly as a result of miscommunication. In addition, during pump impeller and shaft replacement, the containment would be opened, thus allowing a direct release path to the environment. The staff, therefore, requires that RIP impeller and shaft replacement be conducted only after fuel has been removed from vessel. This was Open Item F19.3.3.2.1-1.

In a letter dated January 14, 1994, and subsequent responses to staff questions, GE provided additional information with regard to RIP impeller-shaft removal and CRD replacement. During the RIP shaft and impeller removal, the following replacement sequence, maintenance requirements, and pump design features together with the refueling platform auxiliary hoist design are intended to minimize the likelihood of an unisolable LOCA.

- Upon completion of the RIP motor removal, a maintenance cover is bolted to the bottom of the motor housing, which forms a temporary RPV boundary. the motor housing is then pressurized to verify that the maintenance cover is providing a seal. The secondary inflatable seal is then depressurized. At this point, two seals (internal primary metal-to-metal, and maintenance cover) are still provided. Upon removal of the pump impeller-shaft, only the maintenance cover seal remains. To protect against removal of the impeller-shaft in the event that maintenance seal is not in place, an auxiliary hoist interlock is provided. The refueling platform auxiliary hoist interlocks will interrupt the hoisting power if the load exceeds a specified setpoint. The hoist load setpoint is less than the sum of the impeller-shaft weight and the hydrostatic head on the impeller. To overcome the static head, the motor housing must be pressurized, which requires the maintenance cover plate to be secured in place, and thus sealing is assuring.

- When the pump impeller-shaft has been removed, a maintenance diffuser plug is then installed over the shaft opening. The diffuser plug provides sealing and is the only means to prevent possible unisolable LOCA when the motor housing is drained and the maintenance cover plate is removed for secondary inflatable seal and stretch tube inspection or replacement. To prevent this potential unisolable LOCA, the diffuser plug is designed with a break-away lifting lug. If the maintenance cover is not secured in place and pressurized, the lifting lug will break during the attempted removal due to the static head pressure exceeding the lug's design force, thus ensuring that the diffuser plug seal is maintained. In the event that the operator inadvertently removed the plug, abnormal or excessive drainage will be discovered when the motor housing is partially drained through the drain line. At this point, RIP sealing is still provided by the maintenance cover plate. Discontinued drainage of the motor housing will eliminate the loss of reactor coolant and allow corrective actions.

To further ensure that there is no leakage with the motor bottom cover installed, GE specifically states that COL applicant develops procedure to visually monitor for potential leakage from the motor housing during pump shaft lifting and maintenance plug removal. The staff finds these provisions acceptable and that the potential unisolated LOCA is minimized during RIP maintenance and replacement. Therefore, Open Item F19.3.3.2.1-1 is resolved. However, the staff considers this a COL action item and will review the COL procedures to ensure appropriate installation and verification of motor bottom cover, as well as visual monitoring of the potential leakage during impeller-shaft and maintenance plug removal have been considered. Also, the staff will ensure that COL applicant develops a contingency plan (e.g., close personnel access hatch, safety injection), which assures that core and spent fuel cooling can be provided in the event that a loss-of-coolant occurs during RIP maintenance. GE has included this COL action item in SSAR Amendment 34 and the staff finds it to be acceptable.

19.3.3.2.1.3 Control Rod Drive Replacement

CRD replacement for the ABWR is similar to that of current BWRs, and will use the same maintenance procedures. The CRD is withdrawn to the point where its blade fits onto the CRD guide tube. This provides a metal-to-metal seal that minimizes the RPV water drainage when the CRD is removed. The staff reviewed the replacement process and found that unisolated LOCA with an opening of about 5.08 cm (2 in.) exists at the bottom of the vessel head if the CRD blade and drive are simultaneously removed because of operator failure to follow the procedures. In the DFSER, the staff stated that its position was that TS should be included to prohibit the removal of the blade and drive of the same assembly. This was Open Item F19.3.3.2.1-2.

The procedure for removal of the line motion control rod drive (FMCRD) for maintenance or replacement is similar to current BWRs. The control rod is first withdrawn until the CRD blade is backseated onto the control rod guide tube. This provides a metal-to-metal seal that minimizes the RPV drainage when the FMCRD is subsequently lowered and removed. The CRD blade normally remains in this backseated condition at all times with the FMCRD out. In the event that the CRD blade is required to be removed for replacement, a temporary blind flange will be first installed on the end of the CRD housing to prevent draining of the reactor water.

During the FMCRD removal, personnel are required to monitor under the RPV for water leakage out of the CRD housing. If abnormal or excessive leakage occurs after only a partial lowering of the FMCRD, which is the indicative of a metal-to-metal seal that has not yet been established, the FMCRD can then be raised back into its installed position to eliminate the leak and allow corrective action. In the event that the CRD blade and drive of the same assembly were inadvertently removed due to operator failures to follow procedures during refueling operations with water level greater than 23 ft above the vessel flange, the analysis results indicated that it would take approximately 36 minutes for the water to fill the lower drywell sump, reach the tunnel entrances, and begin flowing into the access tunnels. With the expected flow rate of 174 cubic meters per hour (6, 145 ft^3/hr) from the CRD opening, the water in the spent fuel would drop approximately .3 m/hr (.98 ft/hr). The high drywell sump level and the low spent fuel level would alarm in the main control room approximately 2 minutes and 28 minutes, respectively, into the transient. The normally operating non-safety-related makeup water condensate system (MUWC) will automatically start upon receiving a low level alarm in the spent fuel pool and transfer water to the spent fuel pool cooling and cleanup (FPCCU) system. The RHR spent fuel cooling mode can be manually initiated to provide makeup injection and the suppression pool cleanup system also can provide backup if the MUWC is not available. In the event of loss-of-offsite power, backup water also can be provided by RHR ac independent water addition system.

Upon identified leakages from the bottom of the RPV, it is expected that personnel door and equipment hatch in the lower drywell areas will be closed within 30 minutes before the water level would reach the tunnel entrances and begin flowing into the access tunnel. Appropriate actions will then be taken to reinsert the CRD blade and to mitigate the event using various water sources and injection systems and mentioned.

Additionally, the FMCRD design also allows partial removal of certain mechanical assemblies without the need to withdraw the associated CRD. These mechanical components include the stepping motor, position indication probe (PIP), and spool piece. While these components are removed for maintenance, the associated CRD will be maintained in the fully inserted position by one of two mechanical anti-rotation locking devices, which are part of the FMCRD design. Details of the anti-rotation locking devices and verification process are discussed in SSAR Section 4.6.2.3.4

The staff reviewed the FMCRD and its associated anti-rotation locking devices design and concluded that adequate locking mechanisms are provided to assure that control rods remain fully inserted with the FMCRD subassemblies removed. In addition to the locking devices, TS prohibits removal of any two adjacent CRD subassemblies to prevent a potential inadvertent criticality event during refueling operation.

The staff also notes that only two or three complete FMCRDs are required to be removed for inspection each refueling outage. This is an improvement relative to the CRD system design at current BWRs which have piston seal replacement needs such that 20 to 30 drives are typically removed each refueling outage.

The staff finds that the FMCRD design improvements, provisions to control potential loss of reactor water, and ample time available for operators to initiate corrective actions in the event of coincident removal of the CRD blade and drive of the same assembly during refueling outage acceptable, and therefore, Open Item F19.3.3.2.1-2 is resolved. However, the staff considers this issue a COL action item and will ensure that maintenance procedures have provisions prohibit coincident removal of the CRD blade and drive of the same assembly. The staff also will ensure that COL applicant develops contingency procedures to provide core and spent fuel cooling capability and mitigative actions during CRD replacement with fuel in the vessel. GE has included this COL action item in SSAR Amendment 34 and the staff finds it to be acceptable.

19.3.3.2.2 Alternate Reactor Inventory Control Feature

The ABWR design includes a non-safety-related feedwater and condensate system, consisting of three electric pumps and associated piping, that can be used as an alternate means for make-up during shutdown operation. The CRD pump can also be used to provide inventory control during shutdown by injecting water from the condensate storage tank to the RPV through the FMCRD system. An ACIWA system is also available to supply make-up water to the RPV if no ECCS make-up water is available.

The staff finds these provisions acceptable and concludes that GE has sufficiently addressed the concerns in NUREG-1449 related to alternate make-up capability to provide core decay heat removal. The alternate inventory control features using the feedwater, the condensate system, and the CRD pump will provide alternate core cooling upon loss of normal RHR capability. The staff also finds that an ACIWA system will further enhance the capability of the ABWR to maintain core cooling in the event that no ECCS make-up is available.

19.3.3.3 Containment Integrity

During refueling of the ABWR, the primary containment head is removed and cannot be readily repositioned to restore containment integrity. This is also the case for operating BWR plants with Mark I and II containments. NUREG-1449 stated that BWR secondary containments were judged unlikely to prevent an early release following initiation of boiling with an open RCS or during potential severe-core-damage scenarios. This is also the case for the ABWR.

In NUREG-1449, the staff evaluated the need to reestablish containment integrity for all operating plants under shutdown conditions. Based on operating experience, thermal-hydraulic analyses, and PRA assessments, it was concluded that containment integrity under some shutdown conditions may be necessary for pressurized water reactor (PWR) plants. However, this conclusion was not reached for BWR plants. This is a result in part, of the decreased frequency and significance of precursor events involving reduction in reactor vessel level or loss of RHR (or both) in BWRs as compared to PWRs. In addition, BWRs do not enter a midloop operating condition as do PWRs.

In NUREG-1449, the staff stated that operating BWR alternate DHR methods provide significant depth and diversity. For these reasons, the staff concluded that loss of RHR in BWRs during shutdown is not a significant safety issue as long as the equipment (pumps, valves, and instrumentation) needed for these methods is operable and clear procedures exist for applying the methods. As discussed in Sections 19.3.3.1, 19.3.3.1.1, and 19.3.3.2 of this report, GE provided design features to minimize the risk from shutdown events and ensure the availability of DHR and reactor inventory. GE stated that ABWR TS required that secondary containment automatically be isolated on high radiation from a radiological boundary breach or fuel handling accident. Also, procedures should be developed by the COL applicant to ensure that (1) the primary containment is available during Modes 3 and 4 (if appropriate), and (2) the secondary containment can be maintained functional as required, especially during higher-risk evolutions.

Based on the conclusions reached in NUREG-1449 and the improvements, beyond that of operating BWRs, provided in the ABWR design, the staff concludes that additional requirements are not necessary for the primary containment. The requirements to isolate secondary containment on high radiation and fuel- handling accidents, and procedures to ensure the availability of secondary containment during high-risk evolutions could contribute to the mitigation of a low-power and shutdown event.

19.3.3.4 Electrical Power

In NUREG-1449, the staff concluded that the availability of electrical power is vital to maintaining shutdown cooling. A loss of power could range from the complete loss of ac power to the loss of a dc bus or an instrument bus. Loss of electrical power generally leads to other events, such as a loss of SDC.

The staff reviewed the design for the electrical system described in SSAR Section 19.Q.4.4 of the ABWR PRA Shutdown Risk Evaluation Final Report. The ABWR electrical system design includes three diesel generators — one diesel generator for each safety division. A non-safety-related combustion gas turbine is an alternate means of supplying power. The combustion gas turbine can start a feedwater or other pump for DHR or inventory make-up, if required upon a loss of offsite power and diesel generator failure.

Two independent offsite power sources, three unit auxiliary transformers powering three Class 1E and non-1E buses, and four safety divisions of dc power would increase the availability of power for equipment.

The staff finds that the ABWR electrical power system design contains redundant electrical power sources. The availability of electrical power during shutdown is discussed in Section 19.3.7.3 of this report.

19.3.3.5 Reactivity Control

In NUREG-1449, the staff indicated that inadvertent criticality events at BWR plants have resulted in reactor trips. Inadvertent reactivity is most often caused by human error (the operator selecting the wrong control rod) and feedwater transients. GE addressed reactivity excursion events in SSAR Section 19.Q.4.5 of the ABWR PRA Shutdown Risk Final Report, and in Chapter 15 of the SSAR. GE stated that reactivity events during shutdown could result in critical events by moving control rods or making errors in handling fuel, which could jeopardize DHR or fuel integrity. These events could result from any the following:

- Control rod drop
- Control rod ejection
- Refueling error
- Rod withdrawal error
- Fuel loading error.

19.3.3.5.1 Control Rod Drop

A control rod drop event could occur during control rod testing at shutdown. The applicant discussed the control rod drop accidents in details in Chapter 15 of the GE ABWR SSAR. To limit the reactivity increases that would result from a free-falling control rod, the ABWR will include a latch mechanism to restrict the distance of rod free-fall to an acceptable limit. The FMCRD design will detect the separation of the CRD mechanism. A rod block signal will prevent a second control rod from being withdrawn, if any one control rod is fully withdrawn. The latch mechanism would limit to 20.32 cm (8 in.) the distance a rod could drop if (1) the rod block signal failed, (2) the operator incorrectly selected an adjacent control rod for withdrawal, and (3) the incorrectly selected rod became stuck and decoupled from its drive. Two redundant and separate Class 1E switches will detect the separation of either the control rod from the hollow piston or the hollow piston from the ball nut. If either the FMCRD or the Class 1E separation detection system were actuated, an alarm would annunciate in the control room, thus reducing the probability of rod drop accidents going undetected. These features provide acceptable plant response as discussed in Section 15 of the FSER.

19.3.3.5.2 Control Rod Ejection

During shutdown operation, a major break on the CRD housing or associated CRD pipelines from RPV hydrostatic testing of a control rod could cause a control rod to be ejected. The FMCRD system will include redundant brake mechanisms to prevent this accident from causing severe consequences. The rod drop detection system also will provide the same protection against a rod ejection accident and will include alarms in the control room to alert operators to this accident. Details of the control rod ejection accidents and plant response are discussed in Section 15 of the FSER.

19.3.3.5.3 Rod Withdrawal Error

The staff stated that certain inadvertent criticality events resulted in reactor trips. Inadvertent criticality is most often caused by human error (such as the operator's selecting the wrong control rod). GE stated that the reactor could become critical if two adjacent control rods were withdrawn at the same time during refueling operations. To prevent a rod from being inadvertently withdrawn, the ABWR design will include interlocks to ensure that all control rods are inserted while fuel is being handled over the core. The design also will include an interlock to prevent more than one control rod from being withdrawn at a time. If the interlock fails and the control rod is withdrawn, the reactor would trip on a high-flux signal received in the control room. Details of the rod withdrawal errors are discussed in Chapter 15 of the ABWR SSAR and are evaluated in Section 15 of the FSER.

The ABWR refueling interlocks that prevent more than one control rod from being withdrawn at a time and the high-flux signal scram features will provide diverse protection against rod withdrawal errors during shutdown operations. The staff judged that there is only a small probability of the coincident failures of the refueling interlock and RPS occurring together with operator errors.

19.3.3.5.4 Refueling Error

A reactivity excursion event in an ABWR could occur during refueling operations if a fuel bundle is inserted at the maximum fuel grapple speed into a fueled region of the core. Details of the refueling error events are discussed in Chapter 15 of the ABWR SSAR. The ABWR will include the following design features to prevent or mitigate refueling errors:

(1) An interlock with a mode switch in the REFUEL position prevents another fuel assembly from being hoisted over the vessel if a control blade has been removed.

(2) While the mode switch is in the REFUEL position, only one rod can be withdrawn at a time. Any attempt to withdraw a second control rod would result in a rod block signal from the refueling interlocks.

(3) A refueling error will cause the start-up neutron-monitoring system to send an alarm and alert operators to the condition.

Details of the refueling errors are discussed in Section 15 of this report.

19.3.3.5.5 Fuel Loading Error

A fuel loading error is similar to a refueling error. GE stated that if the refueling procedure were not followed, the core reactivity would increase to a value higher than the design value, and if the operator, in performing the core verification process, failed to identify a misplaced fuel bundle, subsequent control rod testing could result in an inadvertent critical event and a power excursion. The likelihood and acceptability of plant response to this event is discussed in Section 15 of this report.

19.3.4 ABWR Reactor Instrumentation

In NUREG-1449, the staff stressed the importance of dedicated shutdown annunciators and instruments used during shutdown operation to provide RCS coolant temperature indication, reactor water level indication, and RHR system status.

The applicant discussed ABWR instruments in SSAR Section 19.Q.5 of the GE ABWR PRA Shutdown Risk Final Report. The applicant stated that, to minimize risk during shutdown, the instrument system must monitor the RPV level, the water temperature, the make-up sources, and heat sinks, and must display these parameters in the control room to the operators in a reliable manner that is easily understood. These instruments can also supply signals to actuate ECCS functions upon receiving a signal for low reactor water level, to automatically insert control rods on high flux, and to close appropriate isolation valves.

The staff asked GE to discuss the reliability of power available for the CUW system to measure coolant temperatures and the adequacy of the DHR system parameters monitored by the ABWR instrument system. The applicant stated in letters dated January 13 and 28, 1993, that resistance temperature detectors (RTDs) in the CUW system suction lines and the RHR pump discharge will be used to measure reactor water temperatures. The temperature indications and alarms will be located in the control room to enable operators to monitor coolant temperature during shutdown operations. Upon a loss of offsite power, the RHR systems will be powered from the EDGs and the CUW system can be powered by either the EDG or an alternate onsite ac power source combustion turbine generator (CTG). When an alarm is received from the CUW system, the operator would retrieve the associated system on a computer display to determine the specific cause for the alarm condition. GE also gave specific DHR system parameters that will be monitored in shutdown conditions.

Instrument features important to shutdown operations include the following:

- Automatic initiation of ECCS to ensure adequate RPV make-up during Modes 4 and 5
- Four channels of instruments to allow for bypass during maintenance and testing while retaining the redundancy of the system (the two-out-of-four logic reverts to two-out-of-three during maintenance bypass)
- Continuous monitoring to detect fires or flooding in safety-related and other areas
- Operability of the RPS during shutdown conditions to mitigate any reactivity excursions
- Interlocked refueling bridge operation to prevent reactivity excursions
- Automatic isolation of SDC valves (F-010 and F-011) on low level in the RPV to prevent fuel from being uncovered
- Interlocked RHR valves (SDC and suppression pool) to reduce the possibility of diverting coolant from the RPV to the suppression pool
- Ability to shut down the plant from the remote control panel if the control room becomes uninhabitable
- Ability to monitor radiation levels throughout the plant to detect breaches in radiological barriers.

Parameters that are monitored by the instrument system include the following:

- RPV level, water temperature, and pressure

- Neutron flux
- Drywell and wetwell pressure and temperature
- Suppression pool temperature and level
- Turbine building condenser pit and reactor component cooling water (RCW) rooms in the control building flooding level
- RHR flow rate, pump motor trip, loop logic power failure
- Fire detection in various buildings
- Electric power distribution system parameters (e.g., power, voltage, current, and frequency)
- Operation of the fire water system
- CUW outlet temperature high.

The staff finds these provisions acceptable and concludes that GE has appropriately addressed concerns in NUREG-1449 related to the importance of dedicated shutdown annunciators and instruments used during shutdown operations to provide RCS coolant temperature indication, reactor water level indication, and RHR system status.

19.3.5 Flooding and Fire Protection

In NUREG-1449, the staff stated that the safety significance of flooding or spills during shutdown depends on the equipment affected by the spills and that such spills are most often caused by human error. Plant activities during shutdown and refueling operations may increase fire hazards in safety-related systems essential to the plant's capability to maintain core cooling. Further, Appendix R to 10 CFR Part 50, Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979, and current NRC fire protection philosophy do not address shutdown and refueling conditions and the effect that a fire may have on the plant's ability to remove decay heat and to maintain shutdown cooling. SSAR Section 19.Q.6 of the ABWR PRA Shutdown Risk Evaluation Final Report addresses the capability of the plant design to protect safe-shutdown equipment from fire and floods.

The physical layout of the ABWR design is the primary means of ensuring that this requirement will be met should these hazardous conditions occur. The ABWR design incorporates such methods as proper plant layout, proper system layout and operation, physical and electrical separation, and the use of administrative controls to be established by the referencing applicant to provide DHR function and to maintain safe shutdown operation.

The layout of the plant will minimize the propagation of fires and floods by locating the control building (CB) between the reactor building (RB) and the turbine building (TB). Most safe-shutdown equipment will be located in the RB and most non-safety-related equipment will be located in the TB; therefore, failures of systems in the TB because of fires or floods will not adversely affect safety-related equipment.

GE stated that the ABWR includes three independent safety-related RHR divisions. These divisions will be separated by 3-hour fire barriers, and divisional equipment will be electrically isolated from redundant divisions of equipment except for equipment in the main control room (MCR), remote shutdown panels (RSPs), primary containment, and several special cases in which divisional equipment is in close proximity to equipment in another division as described in SSAR Section 9A.5, Special Cases. Also, there are some cases where cables of more than one division are in relatively close proximity and require special justification. In each of these areas, the applicant has provided recommendations concerning fire protection during shutdown conditions to ensure that DHR capability remains intact.

The ABWR design and TS in Modes 3, 4, and 5 ensure physical and electrical separation that no more than one of the three redundant divisions of safe-shutdown equipment can be rendered inoperable because of a fire or flood. Physical and electrical separation of redundant safety-related divisions ensures that, if a division is rendered inoperable because of fire or flood, at least one division will be operable and available to provide DHR function. The ABWR will not include fire walls inside primary containment to provide divisional separation because of the need to rapidly equalize pressure among these divisions after a high-energy line break. In the ABWR SSAR, GE stated that the divisions of safe-shutdown equipment will be widely separated (at intervals of 120 degrees around the containment) so that a single fire will not be able to render inoperable any combination of active components that are used to ensure maintenance of a safe-shutdown condition. During normal plant operation, the primary containment will be made inert, therefore, the potential for fire in the containment is reduced substantially. During shutdown or refueling activities, the primary containment will not be made inert and, therefore, would be susceptible to fires because of maintenance activities that involve increased amounts of combustible loading and a number of ignition sources (cutting and welding equipment). The COL applicant referencing the ABWR design will prepare administrative procedures to limit storage of combustibles and control ignition sources inside primary containment.

Although the divisions of safe-shutdown equipment located in the primary containment will not be located in the separated fire areas, other methods will be available to deal with fires in the primary containment such as minimizing the safe-shutdown components, maximizing distance between divisional equipment, and providing sprinkler coverage from the containment spray mode of RHR.

Within the primary containment, check valves and automatic depressurization system (ADS) SRVs are the only components that need to operate to achieve safe shutdown. A fire cannot disable the check valves or the spring-actuated SRVs, and the high-pressure injection pumps will provide enough pressure to lift the SRVs. Therefore, the safe shutdown capability will not be compromised as a result of a fire. GE stated that containment spray can be used to mitigate the consequences of a fire inside primary containment. The staff does not consider this to be a primary fire mitigation feature. It believes that the design of the SRVs and check valves, along with the wide spaces between divisional equipment and administrative procedures, provides sufficient assurance that safe-shutdown equipment will remain functional, should a fire or flood occur inside the primary containment during shutdown conditions.

Although the MCR includes controls and indications for safe-shutdown divisions, redundancy is achieved by providing the RSP in a separate area of the RB. The MCR and the RSP will be separated by 3-hour fire barriers and the RSP will be hard-wired to field devices and power supplies. Fiber optic cables from the MCR will transmit two identical digital control signals to operate equipment. In the event of a major fire in the MCR, the instrumentation and control power for safety-related cooling systems will be transferred from the MCR to the two RSPs, each of which is capable of serving one division of safety-related cooling systems. It is not likely that a fire in the MCR would cause spurious operation of equipment (barring inadvertent control switch operation) because two simultaneous identical signals are needed at the demultiplexer for control action to be taken at the field device (element).

The RSP rooms will be separated from each other by a fire barrier and fire door assembly. The transfer switches will be located in each room. GE stated that if a fire involved one of the safe-shutdown rooms and caused inadvertent operation of safety-related equipment, only one of the two remaining divisions would be needed to bring the reactor to a hot shutdown and then cold shutdown condition.

The heating, ventilating, and air-conditioning (HVAC) systems will be used during shutdown conditions to prevent smoke, hot gases, heat, or fire suppressant in one fire area from migrating into other fire areas to the extent that they could adversely affect the safe shutdown capability of redundant equipment. When the HVAC systems are operated in the smoke-removal mode, the system in the area experiencing fire will be maintained at a lower pressure than the other fire areas. This will ensure that any leakage across a barrier will be drawn into the area experiencing the fire and will provide a clean air space for personnel access to the fire. The fire suppression and detection system used in conjunction with administrative controls will enable the COL applicant to effectively fight the fire.

While the plant is in shutdown modes, the licensee will commonly bring additional personnel and materials to the site for various refueling and maintenance activities. During this time, some safe-shutdown systems may be inoperable for maintenance at a time when fires are more likely to occur. Administrative controls proposed by GE will ensure that at least one safety-related division is operable (the operable division) and that another safety-related division is operable with fire and flood barriers fully in place (the standby division). Thus while one division may be in maintenance, another division will be operable (although its fire or flood barriers may be breached), and the third division will be in standby. If the operable division is rendered inoperable by a fire or flood, the standby division will be available to perform the DHR function. Likewise, if a fire or flood renders the standby division inoperable, the operable division will continue to perform the DHR function. The COL applicant refer to the ABWR design will establish administrative procedures to ensure that an unanticipated breach of a fire barrier will be discovered and compensated for in a timely manner. The staff considers fire protection administrative procedures a plant-specific issue. In addition, the staff will ensure that the COL applicant has appropriately addressed the availability of the safe-shutdown equipment using the physical and electrical separation concept to provide DHR function and to maintain safe-shutdown operation. The staff also will ensure that the COL administrative procedures provide appropriate controls of combustibles and ignition sources during shutdown operations. GE has included this COL action item in SSAR Amendment 34 and the staff finds it to be acceptable.

The physical and electrical separation of redundant safe-shutdown equipment provides for flood protection. The ABWR will include barriers to protect against both flood and fires. Doors that protect against flood are watertight with seals that will seat with increased water pressure from outside the room. These watertight doors also will be alarmed so that if the door is opened (i.e., the flood barrier is breached), security will be alerted and will take appropriate actions. The ABWR will also reduce flood hazards by minimizing sources of flood water near safety-related equipment and by alarming and isolating water sources on indication of high water levels. Safety-related equipment will be installed at a minimum level of 200 mm (8 in.) off the floor with adequate drainage in the floor spaces. GE will give COL applicants guidance on administrative controls to effectively deal with a flood situation.

The staff reviewed the protection philosophy for the ABWR, the design features used to protect safe-shutdown equipment from fires and floods and their effects, and concludes that the design provides for adequate fire and flood protection for systems and components required to achieve and maintain safe-shutdown and is acceptable.

19.3.6 Shutdown Risk Insights

In supporting the design certification of the ABWR design, the NRC required GE to prepare a suitable, systematic risk analysis for those operating modes other than full power. On July 2, 1992, GE submitted its shutdown risk evaluation for the NRC staff to review. The staff reviewed GE's shutdown risk PRA for the ABWR design. This PRA included discussion of dominant accident sequences; calculation of core damage frequency from internally initiated events in Modes 3, 4 and 5; and vulnerabilities while operating the plant in modes other than full power. The staff also considered human reliability insights, important human actions, insights from uncertainty, importance, and sensitivity analyses.

In SSAR Table 19.Q-6 of the shutdown risk evaluation final report, GE performed a failure modes and effects analysis (FMEA) of the ABWR new design features to determine if any vulnerabilities were introduced by the new technology. GE concluded that the new design features would not introduce new vulnerability. The staff discussed these assessments in Section 19.1.3.4 of this report.

The staff reviewed the shutdown risk PRA for the ABWR design. It concluded that the results of the shutdown risk evaluation provide sufficient information to determine that there would not be a disproportionate risk of operating the plant in modes other than full power. The staff also found no unreported shutdown risk vulnerabilities.

19.3.7 Technical Findings in NUREG-1449

In NUREG-1449, the staff identified several operational-related issues for regulatory action. These issues are the following:

- Outage planning and control
- operator training and procedures
- TS

19.3.7.1 Outage Planning and Control

In the absence of strict TS controls, licensees have considerable freedom in planning their outage activities. NUREG-1449 indicated that outage planning dictates what equipment will be available and what and when maintenance activities will be undertaken. It effectively establishes if and when a licensee will encounter circumstances that are likely to challenge safety functions and the level of mitigation equipment available to deal with such a challenge. The staff believes that a safety-oriented approach to planning and controlling an outage is needed and that such an approach will reduce risk during shutdown, thereby reducing the incidence of precursor events and improving the defense-in-depth concept.

In the GE ABWR PRA Shutdown Risk Final Report, SSAR Sections 19.Q.7 and 10, GE discussed the generic procedure guidelines for planning outages. In response to a staff RAI regarding the guidelines for conducting and planning outages, GE stated in a letter dated January 13, 1993, that the plant-specific operating procedures, and the GE ABWR procedure guidelines for outages and planning will endorse NUMARC 91-06, Guidelines for Industry Actions to Assess Shutdown Risk Management. The staff believes that in addition to NUMARC 91-06 guidelines, fire protection and appropriate use of instrumentation also should be considered. Fire protection provisions for plants during shutdown operation and the use of plant instruments important to shutdown to monitor and detect a loss of DHR cooling during reduced inventory are discussed in Sections 19.3.5 and 4 of this report, respectively. The procedure guidelines for outage and control planning will be based on an engineering evaluation including engineering safety analyses and will address the effective planning and control of outages and the maintenance of important shutdown functions: DHR capability, electrical power availability, reactivity control, and containment (primary/secondary) integrity.

The plant-specific guidelines for planning and controlling outages will include the following:

- An outage philosophy including a list of organizations responsible for scheduling outages. These guidelines should address both the initial outage plan and all safety-significant changes to schedule.

- Provisions to ensure that all activities receive adequate resources. The plan should also consider unanticipated changes and increases of the scope.

- Provisions to ensure defense in depth during shutdown and to ensure that safety margins are not reduced. An alternate or backup system must be made available if a safety system is removed from service.

- Provisions to ensure that all personnel involved in outage activities are adequately trained. This should include operator simulator training to the extent practicable. Other plant personnel, including temporary personnel, should receive training commensurate with the outage tasks they will be performing.

- Provisions for an independent safety review team that would be assigned to perform final review and grant approval.

The staff finds that improvement in safe operation of the ABWR plant in low-power and shutdown modes can be reasonably accomplished by implementing GE's guidelines for preparing and implementing an outage plan. It concludes that GE has adequately addressed important areas described in NUREG-1449 regarding outage planning and control. The staff also notes that specific shutdown TS requirements and guidelines for preparing and implementing an outage plan, will significantly improve safe shutdown operation. ABWR shutdown TS requirements are discussed in detail in Section 19.3.7.3 of this report.

The staff considers outage planning and control a plant-specific issue and will verify that each COL applicant has appropriately implemented GE's guidance and recommendations to improve low-power and shutdown operation. The staff will review the COL applicant's outage planning and control program to ensure that the safety principle is clearly defined and documented. The controlled procedure should clearly define an outage planning process and should incorporate preplanning for all outages. GE has included this COL action item in SSAR Amendment 34 and the staff finds it to be acceptable.

19.3.7.2 Operator Training and Procedures

In NUREG-1449, the staff stated that conditions and plant configurations during shutdown outages for refueling can place control room operators in an unfamiliar situation. Personnel that is properly trained and understand the problems that could arise during outages is essential in reducing risk associated with the outage activities.

GE stated in SSAR Section 19.Q.10 of the GE ABWR PRA Shutdown Risk Final Report that each utility must prepare plant-specific operating procedures based on individual site characteristics and training program requirements. GE gave broad guidance to ensure important safety functions are maintained during shutdown operations as follows:

- Decay Heat Removal Capability. The RHR system will be the normal method of removing decay heat. The COL applicant will use a recovery strategy to address loss of normal RHR including determining an alternate DHR system and personnel responsible for executing the recovery. In planning for an outage, the COL applicant will emphasize the need for RHR systems during periods of high heat decay loads and the later maintenance of the RHR system when decay heat loads have been reduced or when the core has been unloaded to the spent-fuel pool. Procedures will be provided to maintain spent-fuel cooling during core unloading.

- Inventory Control. The COL applicant will implement appropriate procedures to ensure that adequate coolant inventory is maintained at all times during shutdown. Plant activities or configurations in which a single failure can result in loss of inventory will be identified, and compensatory measures will be provided. Specific activities that could result in a loss of inventory such as use of freeze seals, removal of control rods, CRDs, reactor internal pumps, and RHR valve actuation or other activities that could lead to diversion of RPV coolant to the suppression pool will be reviewed, compensatory measures will be provided.

- Electrical Power Availability. The COL applicant will implement procedures to ensure the defense in depth of electrical power sources. Maintenance of power sources will reflect the current plant configurations and conditions. GE recommended normal and alternate power sources be made available during high-risk conditions. The COL applicant will review all maintenance and switchyard activities to identify single failures or procedural errors that could result in loss of power to vital buses during shutdown. Procedures will be written to govern the use of alternate sources of power.

- Reactivity Control. Shutdown reactivity control for the ABWR will be maintained by core design analysis and interlocks to restrict the movement of fuel and CRDs. The COL applicant will provide procedures to ensure that the core is loaded according to design requirements and that fuel movements are not to be permitted while a CRD mechanism is in maintenance. If a refueling sequence must be altered, a new shutdown margin analysis must be performed. All fuel movements will be verified by knowledgeable trained personnel.

- Containment Integrity. The COL applicant will develop procedures to ensure the availability of primary containment during Modes 3 and 4 (if appropriate). In addition, procedures will be available to ensure that the secondary containment function can be maintained, as required, in all modes.

GE has recommended ways for COL applicants to maintain key safety functions during shutdown. However, the effectiveness of these recommendations would depend on the procedures, characteristics and training program requirements of each plant. The staff considers plant-specific procedures a COL action item and will require COL applicants to appropriately address and incorporate plant-specific safety-related issues and the vendor's operating guidance on safe operations during shutdown. GE has included this COL action item in SSAR Amendment 34 and the staff finds it to be acceptable.

19.3.7.3 Technical Specifications.

The TS for current operating plants are the primary sources of operational requirements. The standard technical specifications (STS) include general requirements for reactivity control, inventory control, RHR, and containment capability during all plant conditions. The STS also include general requirements for fire protection.

The staff asked GE to provide shutdown TS requirements for the ABWR design. In letters of June 30 and July 9, 1993, GE submitted shutdown TS for the staff to review.

In the ABWR shutdown TS, GE established systematic requirements for operating the plant in modes other than full-power conditions. The staff reviewed the ABWR shutdown TS using insights from technical findings discussed in NUREG-1449 and the staff's proposed model TS improvements to enhance the safe shutdown operation of all nuclear plants. These proposed improvements were made available for public comment and industry review in July 1993. Table 19.3-1 of this report indicates additional limiting conditions, beyond those currently listed in the improved STS (NUREG-1434), for operation during reduced inventory that GE proposed. The staff compared TS requirement improvements for ABWR design with the BWR-6 TS. The ABWR shutdown TS reflects redundant onsite AC power sources (diesel generators), one offsite power source, and associated support systems to ensure the DHR capability can be maintained and to minimize the loss of DHR from a loss of electrical power. The ABWR shutdown TS closely follows the staff's guidance on the proposed model TS improvements. Therefore, the staff concludes that the ABWR shutdown TS will include requirements needed for managing risk during shutdown operations.

19.3.8 Summary

The ABWR design provides flexible combinations of DHR systems, alternate features, and associated safety parameters monitored by the ABWR instrumentation and control system during shutdown operations. These combinations of safety-related systems and normally operating non-safety-related systems ensure that the ABWR is adequately protected against accidents during low-power and shutdown operations.

GE considered fire and flood hazards and plant damage that could occur during shutdown operation as evidenced by proper plant layout, proper system layout and operation, physical separation, and electrical separation. GE took appropriate actions for adequate fire and flood protection to ensure that at least one safety-related division is operable and the standby safety-related division also is operable with fire and flood barriers fully in place.

Outage planning and control will include specific operating procedures to address key safety features such as DHR capability, inventory control, electrical power availability, reactivity control, and containment control. Specific guidelines for planning and controlling outages will include organizations responsible for scheduling outages. Personnel involved in outage activities will be adequately trained and proper safety reviews will be conducted. Implementations of these operating procedures are COL action items.

The ABWR design includes specific TS requirements for operating the plant in modes other than full power. These TS requirements will provide appropriate redundancy in equipment during higher-risk evolutions during shutdown. These TS are consistent with the staff's proposed model TS developed from evaluation of shutdown and low-power operations.

19.3.9 Conclusion

Based on the above, the staff finds the ABWR PRA Shutdown Risk Evaluation Final Report acceptable, and meets the staff's proposed applicable regulation for shutdown risk. Further, the staff concludes that GE has adequately addressed the shutdown risk concerns in NUREG-1449 and has demonstrated that the ABWR design will not introduce significant risk during shutdown operations.

Table 19.3-1 Comparison of ABWR shutdown TS and BWR-6 TS

ABWR Shutdown TS	Modes	BWR-6 TS	Modes
*RCW/RSW and UHS — Shutdown Two RCW/RSW divisions and UHS operable	4 and 5 except when water level ≥23 ft	No requirements	
*RCW/RSW and UHS — Refueling One RCW/RSW division and UHS subsystems operable	5 with water level ≥23 ft	SSW and UHS — Refueling No requirements	
*AC Sources — Shutdown (low level) One offsite power source Two diesel generator (DG) (onsite) power sources	4 and 5 with water level ≤23 ft	AC source — shutdown one offsite power source One DG (onsite) power source	4 and 5
*AC Sources — Shutdown (high level) One offsite power source One DG (onsite) power source	4 and 5 with water level ≥23 ft		

* A new LCO has been added as a result of the shutdown risk program.

20 GENERIC ISSUES

This chapter covers the staff's evaluation of several topics: (1) unresolved safety issues (USIs) and generic safety issues (GSIs), (2) construction permit/manufacturing license (CP/ML) rule compliance, and (3) incorporation of operating experience in the advanced boiling water reactor (ABWR) design.

Since there is a large number of USIs, GSIs, and CP/ML rule items, the staff has grouped its evaluations according to issue type. Section 20.1 contains task action plan items, which include both USIs and GSIs. Sections 20.2 and 20.3 address new generic issues and human factors issues, respectively, all of which belong to the GSI category. Section 20.4 includes items listed by Three Mile Island (TMI) Action Plan (NUREG-0660) item number, but many are actually evaluated in Section 20.5 since they are also CP/ML items. TMI Action Plan items are in the GSI category. Section 20.5 deals with the CP/ML items, which are listed by their 10 CFR 50.34(f) paragraph number. CP/ML items can also be considered as GSIs. All issues are listed consecutively within each section.

Because of the considerable amount of overlap among CP/ML rule items and USIs and GSIs, Table 20-1 lists all the USIs and GSIs that are technically relevant to the ABWR design or that the staff needed to otherwise address. Table 20-2 lists all the CP/ML issues. These tables provide the issue designation, title, and a reference to the appropriate section(s) of this report containing the evaluation.

Section 20.6 covers incorporation of operating experience in the ABWR design.

Compliance with paragraph (1)(iv) of 10 CFR 52.47(a)

Paragraph (1)(iv) of 10 CFR 52.47(a) requires an application for design certification to include proposed technical resolutions of those USIs and medium- and high-priority GSIs identified in the version of NUREG-0933, "A Prioritization of Generic Safety Issues," current on the date 6 months prior to application and which are technically relevant to the design.

In the draft final safety evaluation report (DFSER), the staff required GE to modify the standard safety analysis report (SSAR) to explicitly discuss the resolution of each technically relevant USI and GSI per 10 CFR 52.47(a)(1)(iv) in the appropriate SSAR section for clarity and to enable the staff to evaluate each item. These were identified in the DFSER as Open Items 20.1-1 and 20.2-1, respectively.

As a result of the ABWR licensing review bases document (letter from T. Murley, NRC to R. Artigas, GE, dated August 7, 1987), GE agreed to address issues beyond the date 6 months prior to the ABWR application. During a conference call on January 13, 1993, the staff and GE mutually agreed that issues identified in NUREG-0933, with Supplements 1 through 15, would be addressed in the ABWR design certification review. It was also agreed that the list of issues contained in NUREG-0933, Appendix B, "Applicability of NUREG-0933 Issues to Operating Reactors and Future Plants," would be used as the baseline list of issues to be addressed for the ABWR design, excluding any issues that were shown in the list to be not applicable to BWR vendors or to future plants. In addition, GE agreed to address five other issues (A-17, A-29, B-5, 29, and 82) that were resolved without the issuance of new requirements, but for which the Office of Nuclear Reactor Research had recommended the development of specific guidance for future plants (although action to develop such guidance is suspended at this time). The staff also asked GE to address one other issue (C-8), the subject of which was an important ABWR review topic.

During the time frame of the review, several issues (113, 120, 121, and 151) were resolved by the NRC without the issuance of new requirements. Since the staff was already pursuing an ABWR response for those items, they were evaluated.

The advance safety evaluation report (SER) stated that based on the staff's evaluation of the issues listed in Table 20-1 of the SER and contingent on GE's incorporation of agreed-to issue mark-ups in the SSAR, the staff concluded that GE adequately demonstrated compliance with or proposed a method of compliance for the USIs and medium- and high-priority GSIs that are technically relevant to the ABWR design as required by 10 CFR 52.47(a)(1)(iv), with some exceptions that required further GE or staff action. These exceptions were that (1) Issue II.F.2 was still open pending the resolution of the differences of views between the staff and GE on the need for diverse instrumentation for reactor pressure vessel water level indication, and (2) Issues II.B.1 and II.K.3(15) required incorporation of COL action items in the ABWR SSAR. These exceptions have been addressed as discussed in Sections 20.5.30, 20.5.18, and 20.4.64, respectively, of this report, and GE has incorporated agreed-to issue mark-ups in the ABWR SSAR. Based on this information and the staff's review of the issues listed in Table 20-1 of this report, the staff concludes that GE has adequately demonstrated compliance with 10 CFR 52.47(a)(1)(iv) for the ABWR design. Therefore, DFSER Open Items 20.1-1 and 20.2-1 are resolved.

Table 20-1 ABWR - Relevant USIs and GSIs

Issue Designation	Title	FSER Section(s)
A-1	Water Hammer (former USI)	20.1.1
A-7	Mark I Long-Term Program (former USI)	20.1.2
A-8	Mark II Containment Pool Dynamic Loads - Long-Term Program	20.1.3
A-9	ATWS (former USI)	20.1.4
A-10	BWR Feedwater Nozzle Cracking (former USI)	20.1.5
A-13	Snubber Operability Assurance	20.1.6
A-17	Systems Interaction (former USI)	20.1.7
A-24	Qualification of Class 1E Safety-Related Equipment (former USI)	20.1.8
A-25	Non-Safety Loads on Class 1E Power Sources	20.1.9
A-29	Nuclear Power Plant Design for the Reduction of Vulnerability to Industrial Sabotage	20.1.10
A-31	RHR Shutdown Requirements (former USI)	20.1.11
A-35	Adequacy of Offsite Power Systems	20.1.12
A-36	Control of Heavy Loads Near Spent Fuel (former USI)	20.1.13
A-39	Determination of Safety Relief Valve Pool Dynamic Loads and Temperature Limits (former USI)	20.1.14
A-40	Seismic Design Criteria - Short-Term Program (former USI)	20.1.15
A-42	Pipe Cracks in BWR (former USI)	20.1.16
A-44	Station Blackout (former USI)	20.1.17
A-47	Safety Implications of Control Systems (former USI)	20.1.18
A-48	Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment	20.1.19
B-5	Ductility of Two-Way Slabs and Shells and Buckling Behavior of Steel Containments	20.1.20
B-10	Behavior of BWR Mark III Containments	20.1.21
B-17	Criteria for Safety-Related Operator Actions	20.1.22
B-36	Develop Design, Testing, and Maintenance Criteria for Atmosphere Cleanup System Air Filtration and Adsorption Units for ESF Systems and Normal Ventilation Systems	20.1.25
B-55	Improved Reliability of Target Rock SRVs	20.1.26

Table 20-1 (Continued)

Issue Designation	Title	FSER Section(s)
B-56	Diesel Reliability	20.1.27
B-61	Allowable ECCS Equipment Outage Periods	20.1.28
B-63	Isolation of Low-Pressure Systems Connected to the Reactor Coolant Pressure Boundary	20.1.29
B-66	Control Room Infiltration Measurements	20.1.30
C-1	Assurance of Continuous Long-Term Capability of Hermetic Seals on Instrumentation and Electrical Equipment	20.1.31
C-8	Main Steam Line Leakage Control Systems	20.1.32
C-10	Effective Operation of Containment Sprays in a LOCA	20.1.33
C-17	Interim Acceptance Criteria for Solidification Agents for Radioactive Solid Wastes	20.1.34
15	Radiation Effects on Reactor Vessel Supports	20.2.1
23	Reactor Coolant Pump Seal Failures	20.2.2
25	Automatic Air Header Dump on BWR Scram System	20.2.3
29	Bolting Degradation or Failure in Nuclear Power Plants	20.2.4
40	Safety Concerns Associated With Pipe Breaks in the BWR Scram System	20.2.5
45	Inoperability of Instrumentation Due to Extreme Cold Weather	20.2.6
51	Proposed Requirements for Improving the Reliability of Open Cycle SWSs	20.2.7
57	Effects of Fire Protection System Actuation on Safety-Related Equipment	20.2.8
67.3.3	Steam Generator Staff Actions - Improved Accident Monitoring	20.2.9
75	Generic Implications of ATWS Events at the Salem Nuclear Plant	20.2.10
78	Monitoring of Fatigue Transient Limits for RCS	20.2.11
82	Beyond Design-Basis Accidents in Spent Fuel Pools	20.2.12
83	Control Room Habitability	20.2.13
86	Long-Range Plan for Dealing with Stress Corrosion Cracking in BWR Piping	20.2.14
87	Failure of HPCI Steam Line Without Isolation	20.2.15
89	Stiff Pipe Clamps	20.2.16

Table 20-1 (Continued)

Issue Designation	Title	FSER Section(s)
103	Design for Probable Maximum Precipitation	20.2.17
105	Interfacing Systems LOCA at LWRs	20.1.29 20.2.19
106	Piping and Use of Highly Combustible	20.2.20
113	Dynamic Qualification Testing of LBHSs	20.2.21
118	Tendon Anchorage Failure	20.2.22
120	On-Line Testability of Protection Systems	20.2.23
121	Hydrogen Control for Large, Dry PWR Containments	20.2.24
124	Auxiliary Feedwater System Reliability	20.2.25
128	Electrical Power Reliability	20.2.26
142	Leakage Through Electrical Isolators in Instrumentation Circuits	20.2.28
143	Availability of Chilled Water Systems	20.2.29
145	Actions to Reduce Common Cause Failures	20.2.30
151	Reliability of RPT During an ATWS	20.2.31
153	Loss of Essential Service Water in LWRs	20.2.32
155.1	More Realistic Source Term Assumptions	20.2.33
HF 1.1	Staffing and Qualifications - Shift Staffing	20.3.1
HF 4.4	Procedures - Guidelines for Upgrading Other Procedures	20.3.2
HF 5.1	Man-Machine Interface - LCSs	20.3.3
HF 5.2	Man-Machine Interface - Review Criteria for Human Factors Aspects of Advanced Controls and Instrumentation	20.3.4
I.A.1.1	Operating Personnel -- Operating Personnel and Staffing - STA	20.4.1
I.A.1.2	Operating Personnel -- Operating Personnel and Staffing - Shift Supervisor Administrative Duties	20.4.2
I.A.1.3	Operating Personnel -- Operating Personnel and Staffing - Shift Manning	20.4.3
I.A.1.4	Operating Personnel -- Operating Personnel and Staffing - Long-Term Upgrading	20.4.4

Table 20-1 (Continued)

Issue Designation	Title	FSER Section(s)
I.A.2.1	Operating Personnel -- Training and Qualification of Operating Personnel - Immediate Upgrading of Operator and Senior Operator Training and Qualifications	20.4.5
I.A.2.3	Operating Personnel -- Training and Qualifications of Operating Personnel - Administration of Training Programs	20.4.6
I.A.2.6(1)	Operating Personnel -- Training and Qualifications of Operating Personnel - Long-Term Upgrading of Training and Qualifications; Revise RG 1.8	20.4.7
I.A.3.1	Operating Personnel -- Licensing and Requalification of Operating Personnel - Revise Scope of Criteria for Licensing Examinations	20.4.8
I.A.4.1(2)	Operating Personnel -- Licensing and Requalification of Operating Personnel - Initial Simulator Improvement; Interim Changes in Training Simulators	20.4.9
I.A.4.2	Operating Personnel -- Simulator Use and Development - Long-Term Training Upgrade	18.7.2.2 20.4.10 20.5.13
I.C.1	Operating Procedures - Short-Term Accident Analysis and Procedures Revision	20.4.11
I.C.2	Operating Procedures - Shift Relief and Turnover Procedures	20.4.12
I.C.3	Operating Procedures - Shift Supervisor Responsibilities	20.4.13
I.C.4	Operating Procedures - Control Room Access	20.4.14
I.C.5	Operating Procedures - Procedures for Feedback of Operating Experience to Plant Staff	13.2 13.5 20.4.15 20.5.41
I.C.6	Operating Procedures - Procedures for Verification of Correct Performance of Operating Activities	20.4.16 20.5.41
I.C.7	Operating Procedures -- NSSS Vendor Review of Procedures	20.4.17
I.C.8	Operating Procedures -- Pilot Monitoring of Selected Emergency Procedures for NTOL Applicants	20.4.18
I.D.1	Control Room Design - Control Room Design Reviews	18.7.2.2 20.4.20 20.5.15
I.D.2	Control Room Design - Plant Safety Parameter Display Console	7.5.2 18.7.2.2 20.4.21 20.5.16

Table 20-1 (Continued)

Issue Designation	Title	FSER Section(s)
I.D.3	Control Room Design - Safety System Status Monitoring	18.7.2.2 20.4.22 20.5.17
I.D.5(2)	Control Room Design - Plant Status and Post-Accident Monitoring	20.4.23
I.D.5(3)	Control Room Design - On-Line Reactor Surveillance System	20.4.24
I.F.2	Quality Assurance - Develop More Detailed Quality Assurance (QA) Criteria	20.4.26 20.5.43
I.G.1	Preoperational and Low-Power Testing - Training Requirements	13.2 20.4.27
I.G.2	Preoperational and Low-Power Testing - Scope of Test Program	20.4.28 14.2
II.B.1	Consideration of Degraded or Melted Cores in Safety Review - RCS Vents	5.2.2 20.4.29 20.5.18
II.B.2	Consideration of Degraded or Melted Cores in Safety Review - Plant Shielding to Provide Access to Vital Areas and Protect Safety Equipment for Post-Accident Operation	12.3.5.1 12.3.6 13.6.3.5 20.4.30 20.5.19
II.B.3	Consideration of Degraded or Melted Cores in Safety Review - Post-Accident Sampling	9.3.2.2 20.4.31 20.5.20
II.B.4	Consideration of Degraded or Melted Cores in Safety Review - Training for Mitigating Core Damage	20.4.32
II.B.8	Consideration of Degraded or Melted Cores in Safety Review - Rulemaking Proceeding on Degraded Core Accidents	6.2.5 9.3.1 20.4.33 20.5.1 20.5.21 20.5.44 20.5.45
II.D.1	RCS Relief and Safety Valves - Testing Requirements	20.4.34 20.5.22 5.2.2 3.9.3.2

Table 20-1 (Continued)

Issue Designation	Title	FSER Section(s)
II.D.3	RCS Relief and Safety Valves - Relief and Safety Valve Position Indication	20.4.35 20.5.23 6.3.3 5.2.2
II.E.1.3	System Design -- AFW System - Update the SRP and Develop Regulatory Guidance	20.4.38
II.E.4.1	System Design -- Containment Design - Dedicated Penetrations	20.4.40 20.5.46 6.3.5
II.E.4.2	System Design -- Containment Design - Isolation Dependability	20.4.41 20.5.26 6.2.4.1
II.E.6.1	System Design -- In-Situ Testing of Valves - Test Adequacy Study	20.4.44
II.F.1	Instrumentation and Controls (I&C) - Additional Accident Monitoring Instrumentation	20.4.46 20.5.29 12.3.6 12.3.4 11.5.2 11.5.1
II.F.2	I&C - Identification of and Recovery from Conditions Leading to Inadequate Core Cooling	20.4.47 20.5.30 6.3
II.F.3	I&C - Instruments for Monitoring Accident Conditions	20.4.48 20.5.31
II.J.4.1	General Implications of TMI for Design and Construction Activities - Revise Deficiency Reporting Requirements	20.4.51
II.K.1(5)	Measures to Mitigate SBLOCAs and Loss-of-Feedwater Accidents -- IE Bulletins - Safety-Related Valve Position Description	20.4.52 18.7.2.2
II.K.1(10)	Measures to Mitigate SBLOCAs and Loss-of-Feedwater Accidents -- IE Bulletins - Review and Modify Procedures for Removing Safety-Related Systems from Service	20.4.53
II.K.1(13)	Measures to Mitigate SBLOCAs and Loss-of-Feedwater Accidents -- IE Bulletins - Proposed TS Changes Reflecting Implementation of All Bulletin Items	20.4.54

Table 20-1 (Continued)

Issue Designation	Title	FSER Section(s)
II.K.1.(22)	Measures to Mitigate SBLOCAs and Loss-of-Feedwater Accidents -- IE Bulletins - Describe Automatic and Manual Actions for Proper Functioning of Auxiliary Heat Removal Systems When Feedwater System Not Operable	20.4.55 20.5.33 5.4.6
II.K.1(23)	Measures to Mitigate SBLOCAs and Loss-of-Feedwater Accidents -- IE Bulletins - Describe Uses and Types of Reactor Vessel Level Indication for Automatic and Manual Initiation Safety Systems	20.4.56
II.K.3(3)	Measures to Mitigate SBLOCAs and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Report Safety and Relief Valve Failures Promptly and Challenges Annually	20.4.61
II.K.3(11)	Measures to Mitigate SBLOCAs and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Control Use of Power-Operated Relief Valves Supplied by Control Components, Inc., Until Further Review Complete	20.4.62
II.K.3(13)	Measures to Mitigate SBLOCAs and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Separation of HPCI and RCIC System Initiation Levels	20.4.63 20.5.5 5.4.6
II.K.3(15)	Measures to Mitigate SBLOCAs and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Modify Break Detection Logic to Prevent Spurious Isolation of HPCI and RCIC Systems	20.4.64
II.K.3(16)	Measures to Mitigate SBLOCAs and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Reduction of Challenges and Failures of Relief Valves; Feasibility Study and System Modification	20.4.65 20.5.6 5.2.2
II.K.3(17)	Measures to Mitigate SBLOCAs and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Report on Outage of ECCSs; Licensee Report and TS Changes	20.4.66
II.K.3(18)	Measures to Mitigate SBLOCAs and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Modification of ADS Logic; Feasi bility Study and Modification for Increased Diversity for Some Event Sequences	20.4.67 20.5.7 6.3.3
II.K.3(21)	Measures to Mitigate SBLOCAs and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Restart of Core Spray and LPCI Systems on Low Level; Design and Modification	20.4.68 20.5.8 6.3
II.K.3(22)	Measures to Mitigate SBLOCAs and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Automatic Switchover of RCIC System Suction; Verify Procedures and Modify Design	20.4.69
II.K.3(24)	Measures to Mitigate SBLOCAs and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Confirm Adequacy of Space Cooling for HPCI and RCIC Systems	20.4.71 20.5.9 6.2.5 5.4.6

Table 20-1 (Continued)

Issue Designation	Title	FSER Section(s)
II.K.3(25)	Measures to Mitigate SBLOCAs and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Effect of Loss of ac Power on Pump Seals	20.4.72 20.5.3 5.4.1
II.K.3(27)	Measures to Mitigate SBLOCAs and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Provide Common Reference Level for Vessel Level Instrumentation	20.4.73
II.K.3(28)	Measures to Mitigate SBLOCAs and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Study and Verify Qualification of Accumulators on ADS Valves	20.4.74 20.5.10 7.3 6.3 5.2.2 3.11
II.K.3(30)	Measures to Mitigate SBLOCAs and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Revised SBLOCA Methods to Show Compliance with 10 CFR Part 50, Appendix A	20.4.75
II.K.3(31)	Measures to Mitigate SBLOCAs and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Plant-Specific Calculations to Show Compliance with 10 CFR 50.46	20.4.76 6.3
II.K.3(44)	Measures to Mitigate SBLOCAs and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Evaluation of Anticipated Transients with Single Failure to Verify no Significant Fuel Failure	20.4.77 15.1
II.K.3(45)	Measures to Mitigate SBLOCAs and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Evaluate Depressurization with Other Than Full ADS	20.4.78 20.5.11 9.6.3 6.3.3
II.K.3(46)	Measures to Mitigate SBLOCAs and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Response to List of Concerns from ACRS Consultant	20.4.79
III.A.1.1(1)	Emergency Preparedness and Radiation Effects -- Improve Licensee Emergency Preparedness - Short Term; Upgrade Emergency Preparedness, Implement Action Plan Requirements for Promptly Improving Licensee Emergency Preparedness	20.4.80
III.A.1.2	Emergency Preparedness and Radiation Effects -- Improve Licensee Emergency Preparedness - Short Term; Upgrade Emergency Preparedness	20.4.81 20.5.37 13.3
III.A.2.1	Emergency Preparedness and Radiation Effects -- Improve Licensee Emergency Preparedness - Long Term; Amend 10 CFR Part 50 and 10 CFR Part 50, Appendix E	20.4.82
III.A.2.2	Emergency Preparedness and Radiation Effects -- Improve Licensee Emergency Preparedness - Long Term; Development of Guidance and Criteria	20.4.83

Table 20-1 (Continued)

Issue Designation	Title	FSER Section(s)
III.A.3.3	Emergency Preparedness and Radiation Effects -- Improving NRC Emergency Preparedness - Communications	20.4.84
III.D.1.1	Radiation Protection -- Radiation Source Control - Primary Coolant Sources Outside the Containment Structure	20.5.85 20.5.38
III.D.3.3	Radiation Protection --Worker Radiation Protection Improvement - Inplant Radiation Monitoring	20.4.86 20.5.39 12.5.1
III.D.3.4	Radiation Protection --Worker Radiation Protection Improvement - Control Room Habitability	20.4.87 20.5.40 9.4.1.1 6.4

Table 20-2 ABWR - CP/ML Rule Items

10 CFR 50.34(f) Paragraph	TMI Item	Title	FSER Section(s)
(1)(i)	II.B.8	Consideration of Degraded or Melted Cores in Safety Review - Rulemaking Proceeding on Degraded Core Accidents, "Design Alternatives from PRA"	20.5.1
(1)(ii)	II.E.1.1	System Design -- AFWS - AFWS Evaluation	20.5.2 20.4.36
(1)(iii)	II.K.2(16)	Measures to Mitigate SBLOCAs and Loss-of-Feedwater Accidents -- Commission Orders on B&W Plants - Impact of RCP Seal Damage Following SBLOCA With Loss of Offsite Power	20.5.3 20.4.59 5.41
(1)(iii)	II.K.3(25)	Measures to Mitigate SBLOCAs and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Effect of Loss of ac Power on Pump Seals	20.5.3 20.4.72 5.4.1
(1)(iv)	II.K.3(2)	Measures to Mitigate SBLOCAs and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Report on Overall Safety Effect of Power-Operated Relief Valve Isolation	20.5.4 20.4.60
(1)(v)	II.K.3(13)	Measures to Mitigate SBLOCAs and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Separation of HPCI and RCIC System Initiation Levels	20.5.5 20.4.63 5.4.6
(1)(vi)	II.K.3(16)	Measures to Mitigate SBLOCAs and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Reduction of Challenges and Failures of Relief Valves; Feasibility Study and System Modification	20.5.6 20.4.65 5.2.2

Table 20-2 (Continued)

10 CFR 50.34(f) Paragraph	TMI Item	Title	FSER Section(s)
(1)(vii)	II.K.3(18)	Measures to Mitigate SBLOCAs and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Modification of ADS Logic; Feasibility Study and Modification for Increased Diversity for Some Event Sequences	20.5.7 20.4.67 6.3.3
(1)(viii)	II.K.3(21)	Measures to Mitigate SBLOCAs and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Restart of Core Spray and LPCI Systems on Low Level; Design and Modification	20.5.8 20.4.68 6.3
(1)(ix)	II.K.3(24)	Measures to Mitigate SBLOCAs and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Confirm Adequacy of Space Cooling for HPCI and RCIC Systems	20.5.9 20.4.71 6.2.5 5.4.6
(1)(x)	II.K.3(28)	Measures to Mitigate SBLOCAs and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Study and Verify Qualification of Accumulators on ADS Valves	20.5.10 20.4.74 7.3 6.3 5.2.2 3.11
(1)(xi)	II.K.3(45)	Measures to Mitigate SBLOCAs and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Evaluate Depressurization with Other Than Full ADS	20.5.11 20.4.78 9.6.3 6.3.3
(1)(xii)	N/A	Evaluation of Alternative Hydrogen Control Systems	20.5.12
(2)(i)	I.A.4.2	Operating Personnel -- Simulator Use and Development - Long-Term Training Upgrade	20.5.13 20.4.10 18.7.2.2
(2)(ii)	I.C.9	Operating Procedures - Long-Term Program Plan Procedures for Upgrading of Procedures	20.5.14 20.4.19 18.7.2.2 13.5

Table 20-2 (Continued)

10 CFR 50.34(f) Paragraph	TMI Item	Title	FSER Section(s)
(2)(iii)	I.D.1	Control Room Design - Control Room Design Reviews	20.5.15 20.4.20 18.7.2.2
(2)(iv)	I.D.2	Control Room Design - Plant Safety Parameter Display Console	20.5.16 20.4.21 18.7.2.2 7.5.2
(2)(v)	I.D.3	Control Room Design - Safety System Status Monitoring	20.5.17 20.4.22 18.7.2.2
(2)(vi)	II.B.1	Consideration of Degraded or Melted Cores in Safety Review - RCS Vents	20.5.18 20.4.29 5.2.2
(2)(vii)	II.B.2	Consideration of Degraded or Melted Cores in Safety Review - Plant Shielding to Provide Access to Vital Areas and Protect Safety Equipment for Post-Accident Operation	20.5.19 20.4.30 13.6.3.5 12.3.6 12.3.5.1
(2)(viii)	II.B.3	Consideration of Degraded or Melted Cores in Safety Review - Post-Accident Sampling	20.5.20 20.4.31 9.3.2.2
(2)(ix)	II.B.8	Consideration of Degraded or Melted Cores in Safety Review - Rulemaking Proceeding on Degraded Core Accidents, "Hydrogen Control System"	20.5.21
(2)(x)	II.D.1	RCS Relief and Safety Valves - Testing Requirements	20.5.22 20.4.34 5.2.2 3.9.3.2
(2)(xi)	II.D.3	RCS Relief and Safety Valves - Relief and Safety Valve Position Indication	20.5.23 20.4.35 6.3.3 5.2.2
(2)(xii)	II.E.1.2	System Design -- AFWS - AFWS System Automatic Initiation and Flow Indication	20.5.247 20.4.37

Table 20-2 (Continued)

10 CFR 50.34(f) Paragraph	TMI Item	Title	FSER Section(s)
(2)(xiii)	II.E.3.1	System Design -- Decay Heat Removal - Reliability of Power Supplies for Natural Circulation	20.5.25 20.4.39
(2)(xiv)	II.E.4.2	System Design -- Containment Design - Isolation Dependability	20.5.26 20.4.41 6.2.4.1
(2)(xv)	II.E.4.4	System Design -- Containment Design - Purging	20.5.27 20.4.42 6.2.5
(2)(xvi)	II.E.5.1	System Design -- Design Sensitivity of B&W Reactors - Design Evaluation	20.5.28 20.4.43
(2)(xvii)	II.F.1	I&C - Additional Accident Monitoring Instrumentation	20.5.29 20.4.46 12.3.6 12.3.4 11.5.2 11.5.1
(2)(xviii)	II.F.2	I&C - Identification of and Recovery from Conditions Leading to Inadequate Core Cooling	20.5.30 20.4.47 6.3
(2)(xix)	II.F.3	I&C - Instruments for Monitoring Accident Conditions	20.5.31 20.4.48
(2)(xx)	II.G.1	Electrical Power - Power Supplies for Pressurizer Relief Valves, Block Valves, and Level Indicators	20.5.32 20.4.49
(2)(xxi)	II.K.1(22)	Measures to Mitigate SBLOCAs and Loss-of-Feedwater Accidents -- IE Bulletins - Describe Automatic and Manual Actions for Proper Functioning of Auxiliary Heat Removal Systems When Feedwater System Not Operable	20.5.33 20.4.55 5.4.6
(2)(xxii)	II.K.2(9)	Measures to Mitigate SBLOCAs and Loss-of-Feedwater Accidents -- Commission Orders on B&W Plants - Analysis and Upgrading of Integrated Control System	20.5.34 20.4.57

Table 20-2 (Continued)

10 CFR 50.34(f) Paragraph	TMI Item	Title	FSER Section(s)
(2)(xxiii)	II.K.2(10)	Measures to Mitigate SBLOCAs and Loss-of-Feedwater Accidents -- Commission Orders on B&W Plants - Hard-Wired Safety-Grade Anticipatory Reactor Trips	20.5.35 20.4.58
(2)(xxiv)	II.K.3(23)	Measures to Mitigate SBLOCAs and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Central Water Level Recording	20.5.36 20.4.70
(2)(xxv)	III.A.1.2	Emergency Preparedness and Radiation Effects -- Improve Licensee Emergency Preparedness - Short Term; Upgrade Emergency Preparedness	20.5.37 20.4.81 13.3
(2)(xxvi)	III.D.1.1	Radiation Protection -- Radiation Source Control - Primary Coolant Sources Outside the Containment Structure	20.5.38 20.4.85
(2)(xxvii)	III.D.3.3	Radiation Protection --Worker Radiation Protection Improvement - Inplant Radiation Monitoring	20.5.39 20.4.86 12.5.1
(2)(xxviii)	III.D.3.4	Radiation Protection --Worker Radiation Protection Improvement - Control Room Habitability	20.5.40 20.4.87 9.4.1.1 6.4
(3)(i)	I.C.5	Operating Procedures - Procedures for Feedback of Operating Experience to Plant Staff	20.5.41 20.4.16
(3)(ii)	I.F.1	Quality Assurance (QA) - Expand QA List	20.5.42 20.4.25
(3)(iii)	I.F.2	Quality Assurance - Develop More Detailed QA Criteria	20.5.43 20.4.26
(3)(iv)	II.B.8	Consideration of Degraded or Melted Cores in Safety Review - Rulemaking Proceeding on Degraded Core Accidents, ".91-Meter (3-Foot) Diameter Equivalent Dedicated Containment Penetration"	20.5.44

Table 20-2 (Continued)

10 CFR 50.34(f) Paragraph	TMI Item	Title	FSER Section(s)
(3)(v)	II.B.8	Consideration of Degraded or Melted Cores in Safety Review - Rulemaking Proceeding on Degraded Core Accidents, "Containment Integrity During an Accident Involving 100-Percent Fuel Clad Metal-Water Reaction"	20.5.45
(3)(vi)	II.E.4.1	System Design -- Containment Design - Dedicated Penetrations	20.5.46 20.4.40 6.3.5
(3)(vii)	II.J.3.1	General Implications of TMI for Design and Construction Activities -- Management for Design and Construction - Organization and Staffing to Oversee Design and Construction	20.5.47 20.4.50

Compliance with paragraph (1)(ii) of 10 CFR 52.47(a)

Paragraph (1)(ii) of 10 CFR 52.47(a) requires an application for a design certification to include a demonstration of compliance with any technically relevant portions of the TMI requirements identified in 10 CFR 50.34(f), often called the CP/ML rule.

GE addressed the TMI requirements of paragraph (1)(ii) of 10 CFR 52.47(a) in SSAR Appendix 19A and in other SSAR sections. In the DFSER, the staff provided evaluations for the majority of GE's submittal relating to compliance with this regulatory requirement and indicated that the additional items not included in the DFSER would be included in the final safety evaluation report (FSER). A number of the staff's evaluations contained open items, combined operating license (COL) action items, and/or technical specification (TS) items that needed to be addressed by GE. The closure of these items for the ABWR design certification review is discussed in the appropriate issue evaluations.

The advance SER stated that based on the staff's evaluation of the issues listed in Table 20-2 of the SER and contingent on GE's incorporation of agreed-to issue mark-ups in the SSAR, the staff concluded that GE adequately demonstrated compliance with or proposed a method of compliance for the technically relevant portions of 10 CFR 50.34(f) as required by 10 CFR 52.47(a)(1)(ii) for the ABWR design, with some exceptions that required further GE or staff action. These exceptions were that (1) 10 CFR50.34(f)(2)(xviii) (TMI Item II.F.2) was still open pending the resolution of the differences of views between the staff and GE on the need for diverse instrumentation for reactor pressure vessel water level indication, and (2) 10 CFR 50.34(f)(2)(vi) (TMI Item II.B.1) required incorporation of a COL action item in the ABWR SSAR. These exceptions have been addressed as discussed in Sections 20.5.30 and 20.5.18, respectively, of this report, and GE has incorporated agreed-to issue mark-ups in the ABWR SSAR. Based on this information and the staff's review of the issues listed in Table 20-2 of this report, the staff concludes that GE has adequately demonstrated compliance with 10 CFR 52.47(a)(1)(ii) for the ABWR design.

Incorporation of operating experience in the ABWR design

In a staff requirements memorandum dated February 15, 1991, on SECY-90-377, "Requirements for Design Certification Under 10 CFR Part 52," the Commission directed the staff to ensure that the design certification process preserves operating experience insights in the certified design. As discussed in Section 20.6 of this report, the staff concludes that GE has adequately considered operating experience identified by generic letters or bulletins issued since 1980 in the ABWR design.

20.1 Task Action Plan Items

This section addresses staff evaluation of USIs and GSIs that are categorized as "task action plan items" in NUREG-0933. All the following issues, with the exception of Issues B-29 and B-32, are relevant to the ABWR design. Issues B-29 and B-32 were evaluated in the DFSER and are discussed here for continuity only.

20.1.1 Issue A-1: Water Hammer (former USI)

Water hammer is defined as a rapid change in pressure caused by a change in velocity of a fluid in a closed volume. Water hammer occurs in various piping systems, such as emergency core cooling, residual heat removal (RHR), containment spray, service water, and feedwater and in steam lines. Water hammer may be caused by rapid condensation of steam pockets, steam-driven slugs of water, pump startup with partially empty lines, and rapid valve motion. Regardless of the cause, water hammer may result in a rapid acceleration of the fluid and may affect the piping system. Severity of the damage from water hammer may range from overstressing of pipe hangers to major damage of restraints, piping, and components.

The review criteria for this issue are stated in the following sections of NUREG-0800, "Standard Review Plan" (SRP): 5.4.7, 6.3, 9.2.1, 9.2.2, and 10.4.7 and Branch Technical Position (BTP) ASB 10-2. Specifically, the feedwater system, containment spray system, shutdown cooling, and other safety-related systems that may be adversely affected by water hammer should be designed to withstand the dynamic loads associated with water hammer. BTP ASB 10-2 requires that the feedwater system be subjected to preoperational testing to demonstrate the effectiveness of the design and operating procedures to mitigate the effects of water hammer.

SSAR Section 19B.2.2 indicates that all of the ABWR systems having potential for water hammer have been analyzed. Different forms of initiating events that could occur and which cause water hammer were considered, such as steam condensation, steam-driven slugs of water, pump startup with partially empty lines, and rapid valve cycling. Section 19B.2.2 states that GE has evaluated various systems for potential water hammer, including the condensate and feedwater system, main steam lines, and all components of the main steam supply system.

With regard to leak before break, GE states that feedwater lines were demonstrated to be immune to failure from water hammer effects. Reactor core isolation cooling (RCIC), high pressure core flooder (HPCF), and RHR systems are precluded from water hammer by keep-full features and the absence of fast-acting valves.

GE also states that the systems susceptible to water hammer effects will be kept full of water, thus preventing water hammer when pumps are started from a steady condition. These systems include the reactor service water (RSW); turbine service water; RHR; HPCF; RCIC; and heating, ventilation, and air conditioning (HVAC) emergency cooling water system. Based on the above, the staff concludes that the ABWR design meets the guidelines in the SRP sections listed above with respect to the dynamic effects associated with possible fluid flow instabilities, such as water hammer. The staff further concludes that GE adequately addressed this issue for the ABWR design.

20.1.2 Issue A-7: Mark I Long-Term Program (former USI)

During testing for an advanced BWR containment system design (Mark III), GE identified suppression pool hydrodynamic loads that had not been considered in the original design of the Mark I containment system. To address this issue, a Mark I Owners Group was formed and the assessment was divided into a short-term and a long-term program. The long-term program was conducted to provide a generic basis to define suppression pool hydrodynamic loads and the related structural acceptance criteria so that a comprehensive reassessment of each Mark I containment system would be performed. A series of experimental and analytical programs was conducted by the Mark I Owners Group. The program proposed to the NRC and reviewed and modified by the staff was to be used to perform plant-unique analyses and identify modifications, as necessary.

The review criteria for this issue are to establish design-basis, conservative loads that are appropriate for the anticipated life of each Mark I BWR containment and to restore the originally intended design margin of safety for the containment system. The principal thrust of the long-term program has been the development of generic methods for the definition of suppression pool hydrodynamic loadings and the associated structural assessment techniques for the Mark I configuration.

It is recognized that the Mark I torus pool and vent configuration is different from the ABWR annular pool and vent design. Therefore, the local loads evaluated within the issue are not applicable to the ABWR design. However, while the results of the Mark I Owners Group investigation cannot be directly applicable for definition of the safety-relief valve (SRV) loads in the ABWR design, they are used as a data base for definition of the SRV loads that are specified in Issue A-39 as discussed in Section 20.1.14 of this report.

20.1.3 Issue A-8: Mark II Containment Pool Dynamic Loads - Long-Term Program

This issue deals with the new containment loads associated with the postulated loss-of-coolant accidents (LOCAs) that were identified as a result of tests by GE. These loads result from the dynamic effects of drywell air and steam being rapidly forced into the suppression pool during a postulated LOCA event. These loads, as well as the loads from actuation of SRVs in the Mark II containment, had not been previously accounted for.

The review criteria for this issue are contained in NUREG-0808, "Mark II Containment Program Evaluation and Acceptance Criteria," and SRP Section 6.2.1.1C and Appendices A and B. SRP Section 6.2.1.1C, Appendix A pertains to steam bypass from the drywell to the suppression pool air volume in the Mark I, II, and III containment designs and states that the system used to quench steam bypassing the suppression pool should be designed so that the steam bypass capability for small breaks satisfies the specified criteria. It also states that the bypass leakage should not substantially increase over the life of a plant. SRP Section 6.2.1.1C, Appendix B summarizes the generic loads acceptable to the NRC and provides information regarding load identification, a summary of the load specification, load specification clarifying criteria, and a reference to the NRC NUREG section that describes the NRC-specific load evaluation.

SSAR Section 3B.4.2.1 states that pool swell response calculations to quantify pool swell loads were based on a simplified, one-dimensional analytical model (described in NEDE-21544-P, "Mark II Pressure Suppression Containment Systems: Analytical Model of the Pool Swell Phenomenon"), which is the same as that reviewed and accepted by the staff in NUREG-0808 for application to Mark II plants. This analytical model was qualified against Mark II full-scale test data. It is recognized that although ABWR wetwell airspace is similar to that of the Mark II design, its vent system design is quite different. Therefore, recognizing the difference in vent system design, additional studies comparing model versus Mark III horizontal vent test data were performed to assure adequacy of the model for application to the ABWR. The staff concludes that this approach adequately addresses this issue for the ABWR since the differences between the ABWR design and the Mark II and Mark III containments have been taken into account in the analysis.

20.1.4 Issue A-9: Anticipated Transient Without Scram (ATWS) (former USI)

This issue deals with the problem of occurrence of transients that require scram but for which scram does not occur. It involves devising measures, both design and operational, that can be taken to avoid or compensate for such occurrences.

The staff's technical findings on this issue were documented in NUREG-0460, "Anticipated Transients Without Scram for Light Water Reactors," and the requirements for resolution are contained in 10 CFR 50.62 (also known as the ATWS rule). For BWRs, 10 CFR 50.62 requires the reactor to have an alternate rod injection (ARI) system that is diverse (from the reactor trip system) from the sensor output to the final actuation device. This system is also required to have redundant scram air header exhaust valves and must be designed to perform its function reliably and be independent of the existing reactor trip system from sensor output to final actuation device. The ATWS rule also requires that each BWR have a standby liquid control system (SLCS) that has the capability of injecting a borated water solution into the reactor pressure vessel (RPV). The borated water must be of such flow rate, boron concentration, and boron-10 isotope enrichment that, when accounting for the RPV volume, the resulting reactivity control is equivalent to that resulting from the injection of 326 Lpm (86 gpm) of 13 weight percent sodium pentaborate decahydrate solution at the natural boron-10 isotope abundance into a 638 cm (251 in.) internal diameter reactor pressure vessel for a given core design. The SLCS must be automatically initiated and the system and its injection location must be designed to perform their functions in a reliable manner. Each BWR must also have equipment designed to reliably trip the reactor coolant recirculation pumps under conditions indicative of an ATWS.

SSAR Sections 15.8 and 19B.2.5 indicate that there are two ways to provide scram in the ABWR design: a motor-driven way and a hydraulic way. In response to a scram signal, the control rods are inserted hydraulically, by the stored energy in the scram accumulator, similar to the current operating BWRs. A scram signal is given simultaneously to insert the fine motion control rod drives (FMCRDs) electrically, via the FMCRD motor drive. This diversity (hydraulic and electric methods of scramming) provides a high degree of assurance that control rods will be inserted when needed.

The ABWR has equipment to trip the reactor internal pumps (RIPs) automatically under ATWS conditions. The RIPs are automatically tripped on reactor high pressure 7860 kPa (1125 psig) (RIPs not connected to the motor-generator set) and RPV Level 2 (RIPs connected to the motor-generator set).

The ABWR design provides recirculation runback for all scram signals, feedwater runback on reactor high pressure and startup range neutron monitoring system ATWS permissive for 2 minutes.

In SSAR Appendix 15E, GE provides ATWS performance evaluation for fuel integrity, containment integrity, primary system, and long-term shutdown cooling. GE states that all requirements of satisfactory performance in the case of an ATWS are met.

As discussed in Section 15.5.2 of this report, the staff performed audit calculations to verify that the ABWR design is satisfactory to mitigate the effects of an ATWS. The review focused on the consequences of manual SLCS actuation and no recirculation pump runback on scram signals other than reactor high pressure and reactor low level. Under some circumstances, a problem was identified regarding the power shift to the top of the core. The staff concluded that the new design of recirculation runback on any scram signal and any ARI FMCRD run-in signal ensure that there is no potential for any unacceptable power shift to the top of the core.

Based on the information provided by GE and the staff evaluation cited above, the staff concludes that GE has adequately addressed this issue for the ABWR. The ABWR complies with the ATWS rule (10 CFR 50.62), as discussed in Section 15.5 of this report.

20.1.5 Issue A-10: BWR Feedwater Nozzle Cracking (former USI)

Inspections of operating BWRs conducted up to April 1978 revealed cracks in the feedwater nozzles of 20 reactor vessels. These cracks ranged in depth from 1.3-1.90 cm (0.5-0.75 in.), including cladding. One crack penetrated the cladding to the base metal for a total depth of approximately 3.8 cm (1.5 in.). It was determined that cracking resulted from high-cycle thermal fatigue caused by fluctuations in water temperature within the vessel in the nozzle region. These fluctuations occurred during periods of low feedwater circulation when the flow was unsteady and intermittent. Once started, the cracks grew because of thermal cycling during startups and shutdowns.

The review criteria for this issue are stated in NUREG-0619, "BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking." This document states that the thermal fatigue and crack initiation of feedwater nozzles are caused by the incoming feedwater, which is considerably colder than the water in the reactor vessel. This is especially true during reactor startup (before feedwater heaters are in service) and during shutdown (after heaters are taken out of service). Turbulent mixing of the hot water returning from the steam separators and dryers and the incoming cold feedwater causes thermal stress cycling of the nozzle unless it is protected by a thermal sleeve.

The proposed design for the ABWR will require that an inner thermal sleeve leading the cooler feedwater to the feedwater sparger be welded to the nozzle safe end. The welded thermal sleeve will assure that there is no leakage of cold feedwater between the thermal sleeve and the safe end. A secondary thermal sleeve is to be placed concentrically in the annulus between the inner thermal sleeve and the nozzle bore to prevent cold water that may be shedding from the outside surface of the inner sleeve from impinging on the nozzle bore and the inside nozzle corner. This proposed double-sleeve design gives a low fatigue usage factor in the nozzle bore and at the inner nozzle corner.

The material of the nozzle forging is SA-508, Class 3, low-alloy steel and the material of the safe end is SA-508, Class 1, carbon steel. The carbon steel safe end is welded to the nozzle forging with a carbon steel weld.

The double thermal sleeve as applied to the ABWR has not been used in earlier plants, although the Monticello (U.S.) and Tsuruga (Japan) plants are using similar designs. A telephone conference with Northern States Power, owner of the Monticello Power Plant, disclosed that the double thermal sleeve performs satisfactorily.

GE proposed an inservice inspection (ISI) program consisting of the following:

- Ultrasonic examination from the external surface of the nozzle ends, nozzle bores, and nozzle blend radius every second outage. If indications are found in the safe ends, the indications will be evaluated per the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI.

- Visual inspection of flow holes and welds in sparger arms and sparger tees every fourth outage.

- Visual inspection of accessible areas of the nozzles from the inner diameter surface on the ASME Code, Section XI, schedule as core internal components.

The ISI program described above is acceptable to the staff because the reactor feedwater nozzles will be ultrasonically and visually examined in service according to ASME Code, Section XI, schedules and inspection criteria. This should ensure that the reactor vessel nozzles will perform in service as designed.

On the basis of the above, the staff concludes that GE adequately addressed this issue for the ABWR design.

20.1.6 Issue A-13: Snubber Operability Assurance

In May 1978, the Advisory Committee on Reactor Safeguards (ACRS) staff observed that there are many licensee event reports (LERs) related to the malfunction of snubbers. The most common LERs involve (1) seal leakage in hydraulic snubbers and (2) high rejection rate during functional testing of snubbers. Snubbers are used as seismic and pipe whip restraints at operating plants. Their safety function is to provide supports to systems or components under dynamic load conditions such as earthquakes and severe hydraulic transients, e.g., pipe breaks. When snubbers are used as vibration arrestors, their fatigue strength must be considered.

The review criteria for this issue are contained in SRP Section 3.9.3. This section states that systems and components that utilize snubbers as shock and vibration arrestors must be analyzed to ascertain their interaction with the systems and components to which they are attached. Snubbers used as shock arrestors do not require fatigue evaluation if it can be demonstrated that certain conditions are satisfied. The criteria for inspection and testing of snubbers are also provided.

SSAR Section 19B.2.7 refers to Section 3.9.3.4.1(3), which in turn, provides the information pertinent to snubber operability assurance. The information consists of the design parameters regarding the required load capacity and snubber location, inspection, testing, replacement, design and testing, installation requirements, and preservice examination. As discussed in Section 3.12.6.6 of this report, the staff concluded that the information provided by GE is consistent with applicable portions of SRP Section 3.9.3 and, therefore, is acceptable. On the basis of the above, the staff concludes that GE adequately addressed this issue for the ABWR design.

20.1.7 Issue A-17: Systems Interaction (former USI)

Nuclear power plants contain many structures, systems, and components (SSCs), some of which are safety related. Some of these SSCs are designed to interact to perform their intended functions and are usually well recognized and accounted for in the evaluation of plant safety by designers and in plant safety assessments. Several significant, plant-specific events have involved unintended or unrecognized dependencies among various SSCs. Some of these events have involved subtle dependencies between safety-related and non-safety-related SSCs and some have even involved subtle dependencies between redundant and supposedly independent SSCs. These dependencies have been termed "adverse systems interactions" (ASIs). This issue was instituted to investigate the potential that these ASIs have remained hidden and could lead to safety-significant events. Seismically induced systems interactions, originally covered in Issue A-46 (which applies only to operating plants), is covered for the ABWR in current licensing requirements.

The staff's technical findings are documented in NUREG-1174, "Evaluation of Systems Interactions in Nuclear Power Plants," and the regulatory analysis in NUREG-1229, "Regulatory Analysis for Resolution of USI A-17." The staff informed the Commission of the resolution in SECY-89-230, "Unresolved Safety Issue A-17, 'Systems Interactions in Nuclear Power Plants'." These documents provide adequate guidance on this issue, although it was resolved by the staff without the establishment of new requirements. Generic Letter (GL) 89-18 was issued to licensees and applicants on September 6, 1989, to inform them about the resolution of this issue.

SSAR Section 19B.2.59 describes the studies on the subject of ASIs that were carried out over the last 10 years and the ABWR design features that could prevent and/or mitigate them. These studies were performed by various organizations, such as NRC's Office of Analysis and Evaluation of Operational Data (AEOD), the Institute of Nuclear Power Operations (INPO), nuclear steam supply system (NSSS) vendors, and the NRC staff based on the operating experience that is available in various publications, such as generic letters, information notices, and bulletins. These studies allowed plant system designers to formulate certain attributes that could be incorporated into design of the ABWR. These attributes consist of the separation criteria, consideration of failure aspects (such as fail safe, diversity, redundancy), protective actions (such as auto versus manual), and so on.

GE states that consideration of ASIs has resulted in designing the ABWR to explicitly avoid unwanted, unacceptable, or unknown ASIs. This has been accomplished through such features as multiple fission barriers, inherent shutdown features and mechanisms, a redundant and diverse engineered safety features (ESFs) network, and a redundant and diverse instrumentation and controls (I&C) protection network.

The SSAR also compares the ABWR-unique features that are designed to prevent, mitigate, and accommodate ASIs with the features of other BWRs. These features include more redundant, diverse and independent power sources, RPV and containment makeup and cooling capabilities, decay heat removal capabilities, and operator action capabilities. Also, the ABWR has more redundant fault-tolerant I&C protection and a more secure and protected ESF housing from fire and flood.

Based on review of the design aspects of the ABWR described above, the staff concludes that the ABWR reflects the proven technology and accepted design requirements and that GE adequately addressed the concerns of this issue.

20.1.8 Issue A-24: Qualification of Class 1E Safety-Related Equipment (former USI)

Construction permit (CP) applicants for which SERs were issued after July 1974 were required by the NRC to qualify all safety-related equipment in accordance with IEEE 323-1974, "Qualifying Class 1E Equipment for Nuclear Power Generating Stations." From the time that this standard was originated, methods to qualify equipment to IEEE 323 were developed by the industry, but some of them, such as testing margins, aging effects, and the simulation of the worst-case environments, have not been accepted by the NRC.

All major NSSS vendors and architect engineers submitted topical reports that describe their methods of qualification. These reports were reviewed by the NRC and the results documented in NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment." These requirements were later established in 10 CFR 50.49 and revised Regulatory Guide (RG) 1.89, "Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants," which describes acceptable methods for complying with 10 CFR 50.49. Dynamic and seismic qualification of Class 1E electrical equipment was not included in the scope of 10 CFR 50.49. Guidance on dynamic and seismic qualification is contained in RG 1.100 (Rev. 2), "Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants."

The criteria for this issue are contained in NUREG-0588 and in 10 CFR 50.49 for environmental qualification and RG 1.100 (Rev. 2), for dynamic and seismic qualification of Class 1E electrical equipment.

Dynamic and seismic qualification testing and analysis of the electrical equipment identified in SSAR Appendix 3I are addressed in SSAR Section 3.10, except for pump motors and valve motor operators which are addressed in Section 3.9. The tests and analyses are to be performed in accordance with IEEE-344, "IEEE Recommended Practices for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations," as modified and endorsed by RG 1.100.

Environmental qualification of safety-related mechanical and electrical equipment is described in SSAR Section 3.11. ABWR Class 1E electrical equipment, including pump and valve motors and electrical accessories, will be environmentally qualified by the methods documented in the NRC-approved report, NEDE-24326-1-P, "General Electric Environmental Qualification Program." These methods are in accordance with the guidance of IEEE 323-1974, NUREG-0588, RG 1.89 (Rev. 1), and the generic requirements of 10 CFR 50.49. Typical environmental conditions (temperature, pressure, humidity, integrated radiation dose, and exposure to chemicals) are provided in SSAR Appendix 3I and cover the design lifetime. Conditions are tabulated for normal operation in and outside of containment and for LOCAs and HELBs inside containment.

Environmental qualification tests and analyses are addressed in SSAR Section 3.11.2. The safety-related equipment in the areas of SSAR Appendix 3I is required to remain functional in the environmental conditions expected at the equipment location during and after the limiting DBA. Qualification tests and analyses of electrical equipment for the effects of aging, radiation, temperature, humidity, chemical spray, submergence, and power supply variation, as applicable, are to be performed and the results documented in accordance with NEDE-24326-1-P.

The proposed qualification program for ABWR electrical equipment is acceptable as discussed further in Sections 3.10 and 3.11 of this report. Therefore, the staff concludes that GE adequately addressed this issue for the ABWR design.

20.1.9 Issue A-25: Non-Safety Loads on Class 1E Power Sources

Class 1E power sources provide the electric power for the plant systems that are essential to reactor shutdown, containment isolation, reactor core cooling, and containment heat removal, and are otherwise essential in preventing significant release of radioactive material to the environment. In some cases, non-safety loads are supplied from the Class 1E power sources and if this is allowed, it is possible that the non-safety loads may cause degradation of the Class 1E power system by introducing loss of redundancy or by other failure mechanisms.

Resolution of this issue required that non-safety-related electrical equipment connected to the Class 1E power systems be limited and, if connected, conform to requirements (for example, independence, electrical isolation, and physical separation) so that the Class 1E system to which the non-safety-related equipment is connected continues to meet the capacity, capability, independence, redundancy, and testability requirements of GDC 17.

The ABWR design incorporates three independent Class 1E diesel generators and a non-Class 1E combustion turbine generator. The combustion turbine generator is designed to automatically assume the majority of non-safety-related electrical equipment independently from the Class 1E diesel generators. Therefore, it is not necessary for non-safety-related electrical equipment to be connected to the Class 1E system.

The ABWR design excludes non-safety-related electrical equipment from the Class 1E system, with the exception of the fine-motion control rod drive subsystem and a portion of the lighting subsystem. The fine motion control rod drive subsystem meets Class 1E requirements from the Class 1E system buses to and including zone-select interlocks (isolation devices). In addition, the fine motion control rod drive subsystem is restricted to Division I in order to assure that the Class 1E subsystems do not violate their independence requirements. The lighting subsystem meets Class 1E requirements from the Class 1E system buses to and including the subsystem load. Because lighting fixtures and bulbs are not seismically qualified in accordance with Class 1E requirements, protective devices (breakers or fuses) and their coordination is provided to assure that Class 1E systems meet their independence and redundancy requirements. The Class 1E system is also sized with sufficient capacity to accommodate operation and failure of the connected non-safety-related subsystems.

The staff concludes that the connection of non-safety-related electrical equipment to the Class 1E system has been appropriately limited. It also concludes that the Class 1E system (with the limited number of connected non-safety-related electrical subsystems) meets the capacity, capability, independence, redundancy, and testability requirements of GDC 17. Therefore, the staff concludes that GE adequately addressed this issue for the ABWR design.

20.1.10 Issue A-29: Nuclear Power Plant Design for the Reduction of Vulnerability to Industrial Sabotage

Reduction of the vulnerability of reactors to radiological sabotage is currently treated as a plant physical security function and not as a plant design requirement. Although present reactor designs provide a great deal of inherent protection against industrial sabotage, extensive physical security measures are still required to provide an acceptable level of protection. An alternative approach would be to consider more fully reactor vulnerabilities to sabotage during the preliminary design phase. Because emphasis is being placed on standardizing plants, it is especially important to consider measures that could reduce the vulnerability of reactors to sabotage. Any design features to enhance physical protection must be consistent with system safety requirements.

The staff resolved this issue without the establishment of new requirements. However, future plants may decrease vulnerability to sabotage by several means. Each division of safety system functions should be totally independent and separated, both mechanically and electrically. Each division should also include 3-hour fire barriers, physical protection of each division from flooding, and physical protection from pipe breaks, both inside and outside the containment. The site security system requirements should be compatible with the plant arrangement and safety system design, definition of vital systems, layout of vital components, security barriers, intrusion detection systems, isolation zone requirements, security alarms, access control, security communications, power supply, and data management. Since there exists a potential for sabotage by a "knowledgeable insider" with authorized access or for acts of sabotage that could occur during maintenance activities, advanced light water reactor (ALWR) plant designers should also analyze the vulnerability of their designs to insider sabotage before finalizing the designs.

SSAR Section 19B.2.4 contains a summary that describes this issue, the Electric Power Research Institute (EPRI) ALWR requirements document, and the proposed resolution of Issue A-29 for the ABWR. It states that the ABWR design will comply with the ALWR requirements document as defined in the SSAR. It also indicates that the ABWR design will mitigate the acts of sabotage through physical separations in the plant arrangement of engineering safety systems and the design and location of barriers to resist threats.

In SSAR Section 19B.2.4, GE states that a sabotage vulnerability analysis will be conducted before the design is finalized. The staff verified that GE established a COL action item in Table 1.9-1 to perform this analysis. This is an acceptable approach. Based on the above, the staff concludes that GE adequately addressed this issue for the ABWR design.

20.1.11 Issue A-31: RHR Shutdown Requirements (former USI)

This issue addresses the safe shutdown of the reactor following an accident or abnormal condition other than a LOCA from a hot standby (that is, the reactor is shut down, but the primary system temperature and pressure are still at or near normal operating values) to a cold shutdown condition. Considerable emphasis has been placed on long-term cooling, which is achieved by the RHR system. The RHR starts to operate when the reactor coolant system (RCS) pressure and temperature are substantially lower than their hot-standby condition values.

Even though it may generally be considered safe to maintain a reactor in a hot-standby condition for a long time, experience has shown that there have been events that required eventual cooldown and long-term cooling until the reactor coolant is cool enough to perform inspection and repairs. For this reason, the ability to transfer heat from the reactor to the environment after a shutdown is an important safety function. It is essential that a power plant be able to go from hot-standby to a cold-shutdown condition after any abnormal occurrence.

The review criteria for this issue are contained in SRP Section 5.4.7, Revision 3. Specifically, the RHR system should meet the intent of the following:

- The design should be such that the reactor can be taken from normal operating to cold shutdown using only safety-grade systems that satisfy the criteria of GDC 1 through 5 of 10 CFR Part 50, Appendix A.

- The system(s) should have suitable redundancy in components and features, and suitable interconnections, leak connections, and isolation capabilities to assure that for onsite electrical power system operation (assuming that offsite power is not available), the system function can be accomplished assuming a single failure.

- The system is capable of being operated from the control room with either onsite or offsite power available.

- The system(s) should be capable of bringing the reactor to a cold shutdown condition, with either onsite or offsite power available, within a reasonable time following a shutdown, assuming the most limiting single failure.

In SSAR Section 19B.2.10, GE stated that the RHR system consists of three electrically and mechanically independent divisions, except for the outboard containment isolation valves, which are in different electrical divisions than the inboard valves. The system will be redundant so that its functional integrity will be assured for onsite electrical power system operation, when offsite power is not available, assuming a single failure. The RHR shutdown cooling subsystem will be activated manually by the operator from the control room following insertion of the control rods and normal blowdown to the main condenser. For emergency conditions, when one of the RHR loops has failed, the RHR system will be capable of bringing the reactor to the cold shutdown condition, i.e., 100 °C (212 °F), within 36 hours following reactor shutdown with two divisions. When all three RHR loops are functioning together, the RHR can remove residual heat (decay and sensible) from the reactor vessel water at a rate sufficient to cool it to 60 °C (140 °F) within 24 hours after the rods are inserted.

The ABWR RHR design does not meet SRP Section 5.4.7, BTP RSB 5-1, Sections B.1(b) and (c), which require the RHR suction side isolation valves to have independent diverse interlocks to prevent the valves from being opened unless RCS pressure is below the RHR system design pressure. Instead, the pressure signal that provides the interlock function is supplied from a 2-out-of-4 logic, which has four independent pressure sensor and transmitter inputs, each of which is in a separate instrument division. The staff concluded in Section 5.4.7 of this report that they satisfy the intent of BTP RSB 5-1. Therefore, the staff concludes that GE adequately addressed this issue for the ABWR design.

20.1.12 Issue A-35: Adequacy of Offsite Power Systems

GDC 17 of 10 CFR Part 50, Appendix A, requires provisions be included in the design to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated by the nuclear power unit, the loss of power from the transmission network, or the loss of power from the onsite electric power supplies.

To meet these requirements of GDC 17, the NRC, in the past, depended on (1) the results of transient and steady-state stability analyses documented in the safety analysis reports for license applications which indicated that the offsite power source remained stable and (2) on design provisions for the disconnection of the offsite power source on its loss of voltage. However, abnormal occurrences at several operating plants indicated that a sustained undervoltage condition of the offsite power can occur and not be detected by the existing loss of voltage design provisions. Failure to disconnect from the undervoltage condition could cause redundant safety-related equipment to be exposed to voltage levels below that for which they are designed. The sustained undervoltage condition could thus result in failure of redundant safety-related equipment.

To resolve this issue, the NRC evaluated the power systems of operating plants to determine the susceptibility of safety-related electrical equipment to: (1) sustained undervoltage condition on the offsite power source, (2) interaction of the offsite and onsite power sources, and (3) adequacy of the existing testing requirements. From this evaluation, the NRC developed an additional requirement for a second level of loss of voltage protection. This second level of protection assures disconnection from offsite power when there is a sustained undervoltage condition on the offsite power source. This additional requirement for a second level of protection was imposed on all operating and future plant designs. In order to assure implementation on future plant designs, Appendix A to SRP 8.3.1 was revised to incorporate this requirement as BTP PSB-1, "Adequacy of Station Electric Distribution System Voltages." In the advance SER, the staff also stated that in resolving A-35 it evaluated the susceptibility of safety-related electrical equipment to the rapid rate of decay of the offsite power source. The advance SER stated that the staff determined this was not a significant safety concern. Details of this determination may be found in an NRC memorandum for K. Kniel from M. Srinivasan dated July 31, 1981.

The review guideline for this issue is that the design of the undervoltage protection schemes for the Class 1E buses of the onsite power system conform to the requirements of BTP PSB-1. Specifically, a second level of voltage protection should be provided for Class 1E equipment, in addition to the existing protection based on detecting the complete loss of offsite power to the Class 1E buses. The second level should have two separate time delays: one before alerting the control room operator and the other automatically separating the Class 1E buses from the offsite power source. The time delays should be long enough to ensure protection from sustained low voltage while avoiding disconnection from the offsite source because of short-term transients such as motor starting. The undervoltage protection scheme should have the capability to be tested and calibrated during power operation. Voltage levels at the safety-related buses should be optimized for the maximum and minimum load conditions that are expected throughout the anticipated range of offsite power source voltage variations. TS are to include limiting conditions of operation, surveillance requirements, and protection equipment set points.

The ABWR design provides two levels of protection for independence of offsite and onsite systems during loss of or degraded voltage conditions. During loss of voltage condition, that is, when the bus voltage decays to less than 70 percent of its normal rated value, a bus transfer to the diesel generator is initiated by the first level of protection. During degraded voltage conditions, that is, when the bus voltage decays to between 70 and 90 percent of its normal rated value for a sustained period of time, the bus is tripped by the second level of protection. With the bus tripped, bus transfer to the diesel generator is initiated by the first level of protection. Equipment will be qualified for voltages below 90 percent for the period of time the equipment will be subjected to these voltage conditions.

The staff reviewed the ABWR design for a second level of protection in accordance with the guidelines of BTP PSB-1, and concluded in Section 8.2.3.2 of this report that the design meets the requirements of GDC 17 defined above. Therefore, the staff concludes that GE adequately addressed this issue for the ABWR design.

20.1.13 Issue A-36: Control of Heavy Loads Near Spent Fuel (former USI)

Overhead cranes are used at all nuclear plants to lift heavy objects in the vicinity of spent fuel. If a heavy object such as a spent fuel shipping cask or shielding block were to fall onto spent fuel in the storage pool or reactor core during refueling and damage the fuel, radioactivity could be released to the environment. Such an occurrence also has the potential of overexposing plant personnel to radiation. If the dropped object were large and the damaged fuel contained a considerable amount of undecayed fission products, radiation releases could exceed the guidelines of 10 CFR Part 100. With the advent of increased and longer-term storage of spent fuel, the NRC determined that there is a need for a systematic review of requirements, facility designs, and TS regarding the movement of heavy loads to assess safety margins and improve them where necessary.

The review criteria for this issue are stated in NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants." They provide for the safe path of the load, training of the operators, inspection and tests of the equipment involved, safety interlocks, and limit switches, etc. The review of the proposed resolution of this issue consists of determining incorporation of the guidelines provided in NUREG-0612 in the ABWR design.

SSAR Section 19B.2.12 states that a number of measures are required to preclude an accident involving heavy loads, in general, and in the vicinity of the storage pool, in particular. These measures include the following:

- The COL applicant will perform a study on all planned heavy-load-handling moves to evaluate and minimize safety risks. The study will establish the heavy-loads-handling paths and routing plans. The COL applicant will also be required to provide the NRC a confirmatory structural evaluation of the spent fuel racks.
- The major heavy-load-handling equipment components (such as cranes and hoists) will be provided with an operating instruction and maintenance manual, in conformance with the guidelines of NUREG-0612, for reference and utilization by operation and maintenance personnel.
- Crane inspections and testing will comply with the requirements of American National Standards Institute (ANSI) B30.2, "Overhead Gantry Cranes," and NUREG-0612, Section 5.1.1(6).
- The equipment-handling components used over the spent fuel pool are designed to meet the single-failure-proof criteria, according to the guidelines of NUREG-0554, "Single Failure Proof Crances for Nuclear Power Plants." Safety interlocks and limit switches are provided to prevent transporting heavy loads, other than spent fuel by the refueling platform crane, over any spent fuel that is stored in the spent fuel storage pool.
- The reactor vessel head lifting strongback and the dryer/separator lifting strongback are designed in accordance with the guidelines of NUREG-0612 and ANSI N14.6, "Standard for Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 Kg) or More for Nuclear Materials."
- The design bases of the system will conform to the requirements of GDC 2, as it relates to the ability of structures, systems, and mechanisms to withstand the effects of earthquakes; GDC 4 as it relates to protection of safety-related equipment from the effects of internally generated missiles (i.e., dropped loads); and GDC 61 as it relates to the safe handling and storage of fuel.

The staff verified that GE established a COL action item in Table 1.9-1 to provide design details for the load handling equipment. This approach is acceptable to the staff. Based on the above information, the staff concludes that GE adequately addressed this issue for the ABWR design.

20.1.14 Issue A-39: Determination of Safety Relief Valve Pool Dynamic Loads and Temperature Limits (former USI)

Operation of BWR primary system pressure relief valves can result in hydrodynamic loads on the suppression pool retaining structures or those structures located within the pool. These loads result from initial vent clearing of relief valve piping and steam quenching because of high local pool temperatures. Overall, the definition methodology used for the ABWR containment is similar to that used for prior BWR containment designs. In spite of the unique features in the ABWR, such as pressurization of the wetwell gas space, the presence of a lower drywell, the smaller number of horizontal vents (30 in the ABWR versus 120 in the Mark III containment), and extension of horizontal vents into the pool, the hydrodynamic loads addressed in this issue are similar to those in other BWR designs.

The review criteria pertinent to this issue are contained in SRP Section 6.2.1.1.C, Appendix B, which lists the generic loads acceptable to the NRC, including load identification, a summary of load specification, load specification clarifying criteria, and reference to the NRC NUREG section that describes the NRC-specific load evaluation. The staff considers that this issue summarizes and incorporates the pertinent results of the studies described in Issues A-7 and B-10, Sections 20.1.1 and 20.1.21, respectively of this report.

SSAR Appendix 3B describes containment hydrodynamic loads, such as those resulting from the SRVs, quencher discharge loads, pressure and temperature transients, and submerged structure loads. The ABWR containment design has some unique features that differ from previously approved designs, such as the Mark III containment. These unique features include pressurization of the wetwell airspace, the presence of a lower drywell, the smaller number of horizontal vents into the pool, vent submergence, and suppression pool width. GE states in the SSAR that SRV discharge is completely condensed in the pool and steam condensation loads are low compared to those of other submerged structures loads because of SRV line air clearing and LOCAs. Consequently, dynamic loads on submerged structures during quencher steam condensation will not be defined and considered for containment evaluation. This is appropriate for the dynamic loads associated with the SRV.

Based on this information and the staff's evaluation of the ABWR containment analysis in Section 6.2.1.6 of this report, the staff concludes that GE adequately addressed this issue for the ABWR design.

20.1.15 Issue A-40: Seismic Design Criteria - Short-Term Program (former USI)

Seismic design of nuclear plants is reviewed in accordance with the provisions of SRP Sections 2.5.2, 3.7.1, 3.7.2, and 3.7.3. Over the years, there has been an evolution of seismic design requirements and technology. The objective of this issue was to investigate selected areas of the seismic design sequence to determine their conservatism for all types of sites, to investigate alternative approaches where desirable, to quantify the overall conservatism of the design sequence, and to modify the NRC criteria in the SRP where justified. Studies were conducted and the results were documented in NUREG/CR-1161, "Recommended Revisions to Nuclear Regulatory Commission Seismic Design Criteria," with specific recommendations for changes in seismic design requirements.

SRP sections were then revised with the following principal areas of change: Section 2.5.2 was updated to reflect the current NRC staff review practice; Section 3.7.1, to reflect design time history criteria; Section 3.7.2, to reflect development of floor response criteria, damping values, soil-structure interaction uncertainties, and combination of modal responses; and Section 3.7.3, to reflect seismic analysis of above-ground tanks and Category 1 buried piping.

The review criterion for the resolution of this issue is conformance with the seismic design acceptance criteria of Revision 2 to SRP Sections 2.5.2, 3.7.1, 3.7.2, and 3.7.3. Specifically, these SRP sections cover review of the site characteristics and earthquake potential, the parameters to be used in seismic design, methods to be used in seismic analysis of the overall plant, and methods to be used in seismic analysis of individual systems and components.

SSAR Section 19B.2.14 states that the design ground motions, site parameters, and system and subsystem analyses criteria and methods described in SSAR Sections 2.3.2.22, 3.7.1, 3.7.2, and 3.7.3 meet the intent of the corresponding SRP sections, except that the operating-basis earthquake (OBE) requirement is not a requirement for the ABWR. Elimination of the OBE from the ABWR design is consistent with the Commission-approved staff position on the policy issue regarding elimination of the OBE addressed in SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light Water (ALWR) Designs," April 2, 1993.

Based on this information, the staff concludes that GE adequately addressed this issue for the ABWR design.

20.1.16 Issue A-42: Pipe Cracks in BWR (former USI)

Pipe cracking has occurred in the heat-affected zones of welds in primary system piping in BWRs since mid-1980. These cracks have occurred mainly in Type 304 unstabilized austenitic stainless steel, which is the pipe material used in most operating BWRs. The major problem is recognized to be intergranular stress corrosion cracking (IGSCC) of austenitic stainless steel components. These components have been made susceptible to this failure by being exposed to a sensitizing temperature range 427-816 °C (800-1500 °F) during welding or post-weld heat treatment.

The review criteria for this issue are contained in NUREG-0313, Revision 2, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping."

The ABWR design complies with RG 1.44, "Control of the Use of Sensitized Stainless Steel," and with the guidelines of NUREG-0313, Revision 2. These documents specify that low-carbon austenitic stainless steels are used in the construction of BWR piping and that low carbon (with a minimum of 8 percent ferrite) weld metal as deposited, is utilized in the fabrication of BWR piping. This will ensure that sensitization of ABWR components will be avoided. Based on this information, the staff concludes that GE adequately addressed this issue for the ABWR design.

20.1.17 Issue A-44: Station Blackout (former USI)

The complete loss of ac electrical power to the essential and nonessential switchgear buses in a nuclear power plant is referred to as a "station blackout" (SBO). Because many safety systems required for reactor core decay heat removal are dependent on ac power, the consequences of an SBO could lead to a severe core damage accident. This issue involves the likelihood and duration of the loss of all ac power and the potential for severe core damage after a loss of all ac power.

The issue arose because of experience with the reliability of ac power supplies. Numerous reports of standby diesel generators failing to start and run had been received and a number of operating plants had experienced a total loss of offsite electrical power. In almost every one of these latter events, the onsite ac power supplies were available to supply power to the safety equipment. However, in some instances, one of the redundant onsite ac power supplies had not been available and, in a few cases, ac power was completely lost (although during these latter events, ac power was restored in a short time and no serious consequences resulted).

The results of WASH-1400/NUREG-75/014, "Reactor Safety Study, An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," showed that for one of the evaluated plants, an SBO could be an important contributor to the total risk from nuclear power plant accidents. Although the total risk was found to be small, the relative importance of an SBO was established.

To resolve this issue, the NRC designated SBO as an unresolved safety issue and implemented a task action plan to determine the need for additional safety requirements. The results, described in NUREG-1109, "Regulatory/Backfit Analysis for the Resolution of Unresolved Safety Issue A-44, Station Blackout," indicated that actions could be taken to reduce the risk from SBO events. The NRC amended its regulations in 10 CFR Part 50 to include the SBO rule (10 CFR 50.63) and issued an associated regulatory guide (RG 1.155, "Station Blackout") that provides guidance on an acceptable means to comply with the SBO rule.

Paragraph (a) of 10 CFR 50.63 requires that each light-water-cooled nuclear power plant be able to withstand and recover from an SBO of a specific duration. The specified SBO duration must be based on (1) the redundancy of the onsite standby ac power sources, (2) the reliability of the onsite standby ac power sources, (3) the expected frequency of loss of offsite power, and (4) the probable time needed to restore offsite power. During the specified SBO duration, the reactor core and associated coolant, control, and protection systems, including station batteries and any other necessary support systems, must provide sufficient capacity and capability to ensure that the core is cooled and appropriate containment integrity is maintained.

Paragraph (c)(2) of 10 CFR 50.63 allows an alternate ac (AAC) source to be used to meet the above defined requirements of 10 CFR 50.63(a) provided that:

- Either the AAC source can be demonstrated by test to be available within 10 minutes to supply power to switchgear buses that are capable of supplying power to required shutdown equipment, or the reactor core and associated coolant, control, and protection systems, including station batteries and any other necessary support systems, have sufficient capacity and capability to ensure that the core is cooled and appropriate containment integrity is maintained from the onset of SBO until the AAC source and required shutdown equipment are started and lined up to operate,
- The time required for startup and alignment of the AAC source and shutdown equipment is demonstrated by test, and
- The AAC source, as a minimum, has the capacity and capability to ensure the plant can be brought to and maintained in safe shutdown.

To meet the SBO rule, GE has provided a combustion turbine generator (CTG) as an AAC source. The CTG has the capability of being aligned with any one of the three

Class 1E divisions within 10 minutes. The CTG has sufficient capacity and capability to supply the loads that can be connected to one Class 1E division. Each of the three Class 1F divisions has sufficient capacity and capability to ensure the plant can be brought to and maintained in safe shutdown. The capability of aligning the CTG with a Class 1E bus within 10 minutes can be demonstrated by test. In addition, the time required for startup and alignment of the CTG and shutdown equipment can be demonstrated by test.

The staff reviewed the ABWR design for its ability to withstand and recover from an SBO in accordance with the guidelines of RG 1.155, and concluded in Section 8.3.9 of this report that the design meets the above defined requirements of 10 CFR 50.63(c)(2). Therefore, the staff concludes that GE adequately addressed this issue for the ABWR design.

20.1.18 Issue A-47: Safety Implications of Control Systems (former USI)

Concerns have been raised regarding the potential for accidents or transients being made more severe as a result of control system failures, including control and instrumentation power supply faults. During the licensing review process, the staff performs an audit review of the non-safety-grade control systems to ensure that an adequate degree of separation and independence is provided between these non-safety-grade systems and the safety systems. On this basis, it is generally believed that control system failures are not likely to result in safety function losses that could lead to serious events or result in conditions that the safety systems are not able to mitigate. However, in-depth studies for all non-safety-grade systems have not been performed.

Generic Letter 89-19, "Request for Action Related to Resolution of Unresolved Safety Issue A-47, 'Safety Implication of Control Systems in LWR Nuclear Power Plants'," recommended that all GE BWR plant designs provide (1) automatic reactor vessel overfill protection to mitigate main feedwater (MFW) overfeed events and (2) plant procedures and TS to periodically verify the operability of overfill protection during power operation.

The ABWR reactor vessel overfill protection is described in SSAR Section 7.7.1.4(9). The level control system provides interlocks and control functions to other systems. When the reactor water level reaches the Level 8 trip set point, the feedwater control system (FWCS) simultaneously (1) annunciates a control room alarm, (2) sends a trip signal to the turbine control system to trip the turbine generator, and (3) sends trip signals to the condensate, feedwater, and condensate air extraction (CF&CAE) system to trip all feed pumps and to close the MFW discharge valves. This interlock is enacted to protect the turbine from damage from high moisture content in the steam caused by excessive carry over while preventing the reactor water level from rising any higher.

In the event that the feedwater pump discharge valves fail to close following the Level 8 trip signal, the FWCS automatically issues another signal to the CF&CAE system to trip all condensate pumps in order to avoid overpressurization of the vessel.

Based on the information above and in SSAR Section 19B.2.17, which states that the COL applicant will develop plant procedures including reactor vessel overfill considerations as shown in SSAR Figure 7.7-8, "Feedwater Control System IED," the staff finds that the proposed Issue A-47 resolutions are acceptable. Therefore, the staff concludes that GE adequately addressed this issue for the ABWR design.

20.1.19 Issue A-48: Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment

This issue concerns the control of large quantities of hydrogen in reactors with small volume containments. As a result of the accident at TMI-2, the Commission issued requirements on hydrogen control in 10 CFR 50.34(f) and 50.44. 10 CFR 50.34(f) requires a hydrogen control system. This system must be based on a 100-percent, fuel clad metal-water reaction and a hydrogen concentration limit of 10 percent on uniformly distributed hydrogen in the containment or on a post-accident atmosphere that will not support hydrogen combustion.

Provision of a noncombustible containment atmosphere (inerting) is an acceptable approach to addressing this issue and is mandated for those reactors with a Mark I or II type of containment. These plants must also have hydrogen recombiners, either internal or external. Reactors with a Mark III type of containment are required to provide a hydrogen control system.

During the TMI-2 accident, it became apparent that metal-water reactions generated hydrogen in excess of the amounts specified in 10 CFR 50.44. In June 1990, the Commission approved the staff's recommendations in SECY-90-016 as the requirements for evolutionary LWRs. These requirements are set forth in 10 CFR 50.34(f)(2)(ix).

The plant-specific design must also comply with 10 CFR 50.34(f) for combustible gas control. 10 CFR 50.34(f)(2)(ix) requires that a hydrogen control system be based on a 100-percent, fuel clad metal-water reaction and a hydrogen concentration limit of 10 percent on uniformly distributed hydrogen in the containment or on a post-accident atmosphere that will not support hydrogen combustion. 10 CFR 50.34(f)(3)(v) requires that containment integrity be maintained below Service Level C limits for steel containments and the factored load category for concrete containments during an accident that releases hydrogen generated from a 100-percent, fuel clad metal-water reaction.

This issue was resolved by the requirements contained in 10 CFR 50.34(f) and 10 CFR 50.44. SSAR Section 19B.2.18 states that there are no design-basis events for the ABWR that result in core uncovery or core heatup sufficient to cause significant metal-water reaction. It further states that this is equivalent to the reaction of the active clad to a depth of 0.00058 cm (0.00023 in.) or 0.72 percent of the active clad. SSAR Section 6.2.5.3 states that the atmospheric control system (ACS) is designed to maintain the containment in an inert condition, except for nitrogen make-up needed to maintain a positive containment pressure and prevent air leakage from the secondary into the primary containment.

GE analyzed consequences of hydrogen release and concluded that for 100-percent, fuel clad metal-water reaction, the resulting peak containment pressure would be about 517 kPa (75 psig). The ABWR has a concrete containment with a steel upper drywell head. This head, based on GE structural analysis, has been shown to be the most limiting structural component of the containment, with a Service Level C limit of 669 kPa (97 psig). Therefore, the containment pressurization resulting from a 100-percent, fuel clad metal-water reaction, coupled with a large-break LOCA, is below the Service Level C limit.

On the basis of this information and the staff's evaluation of ABWR compliance with 10 CFR 50.34(f)(2)(ix), as discussed in Sections 20.5.21 and 19.2.3.3.1 of this report, the staff concludes that GE adequately addressed this issue for the ABWR design.

20.1.20 Issue B-5: Ductility of Two-Way Slabs and Shells and Buckling Behavior of Steel Containments

This issue addresses two concerns relating to containment design: (1) that sufficient information is not available to predict the behavior of two-way reinforced concrete slabs, and (2) that the structural design of a steel containment vessel subjected to unsymmetrical dynamic loadings may be governed by the instability of the shell. The safety significance of the first concern is that in the event of the collapse of a floor that may be caused by an earthquake or a LOCA, there would be a possibility that other portions of the RCS or safety-related systems could be damaged. The damage could lead to an accident sequence resulting in the release of radioactivity to the environment.

The other concern, identified in NUREG-0471, "Generic Task Problem Descriptions (Categories B, C, and D)," is over the lack of a uniform, well-defined approach to design evaluation of steel containments. The structural design of a steel containment vessel subjected to unsymmetrical dynamic loadings may be governed by the instability of the shell. For these types of loads, the current criteria and the current analytical techniques may not be as comprehensive as they should be.

The review criterion for the first concern is that the design code, American Concrete Institute code ACI 349, "Code Requirements for Nuclear Safety Related Structures," contains sufficient information pertaining to the design of two-way slabs subjected to dynamic loads and biaxial tension to enable a reasonably accurate analysis. ACI 349 should be used in conjunction with the pertinent regulatory documents such as SRP Section 3.5.3, Appendix A, and RG 1.142, "Safety-Related Concrete Structures for Nuclear Power Plants (Other than Reactor Vessels and Containments)."

The review criterion for the second concern is that all applied loads must be adequately addressed by the steel containment design. RG 1.57, "Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components," recommends a minimum factor of safety of 2 against buckling for the worst loading condition, provided that a detailed, rigorous analysis that considers inelastic behavior is performed. Also, the allowable stress values for buckling are contained in the ASME Code, Section III, Subsection NE-3222.

SSAR Sections 3.8.3, 3.8.4, and 19B.2.61 state that the design of the ABWR safety-related structures (other than the containment vessel), including consideration of the ductility requirements for the two-way slabs, is based on the ACI 349-80 Code. The approach used by GE for the design of two-way reinforced concrete slabs meets the guidelines of SRP Sections 3.8.3 and 3.8.4 and is, therefore, acceptable.

The applicant indicated that the ABWR containment is a concrete structure and the steel component not backed by concrete, that is, the ABWR reactor closure head, is designed in accordance with the ASME Code, Section III, Subsection NE. This approach meets the guidelines of SRP Section 3.8.2 for shell buckling and, therefore, is acceptable.

Based on the above discussions, the staff concludes that GE adequately addressed this issue for the ABWR design.

20.1.21 Issue B-10: Behavior of BWR Mark III Containments

This issue deals with the dynamic loads present in the Mark III containment following a postulated LOCA when escaping steam forces the suppression pool water out of the drywell and into the wetwell. This action results in pool swell and loads from vent clearing, jets, chugging, impact of water, impact of froth impingement, pool fallback, condensation, and containment pressure. The concern is that these loadings may damage structures and components located within the wetwell. Although many of these structures (e.g., walkways) are by themselves not related to safety, the various emergency core cooling systems (ECCSs) take suction from the wetwell and, therefore, damage in the wetwell may affect the performance of the ECCSs.

The review criteria for this issue are provided in NUREG-0978, "Mark III LOCA-Related Hydrodynamic Load Definition," Appendix C. These criteria have been developed on the basis of large-scale tests conducted between 1973 and 1979 by GE in order to define the LOCA-related hydrodynamic loads for use in the design of the standard Mark III containment.

The ABWR horizontal vent confirmatory test program was performed to obtain data that could be used to determine condensation oscillation and chugging loads for design evaluation of containment structures. The test matrix included tests at conditions that produce bounding loads and additional tests to examine the sensitivity of the loads to system parameters. The test specifically documents work performed, including general evaluation of the test data and the specification of procedures that can be used to define containment loads.

The ABWR design utilizes a horizontal vent system, which is similar to that of the Mark III containment design, but includes some ABWR-unique design features. These unique features include pressurization of the wetwell airspace, the presence of a lower drywell, a smaller number of horizontal vents (30 in the ABWR containment versus 120 in the Mark III containment), extension of horizontal vents into the pool, vent submergence, and suppression pool width.

The ABWR horizontal vent test program has been based on resolution of several other issues (A-7, A-8, and B-10), which produced test data to confirm and define condensation oscillation and chugging loads for design application. SSAR Section 3B.2.2 describes a spectrum of postulated LOCAs that was considered in assessing the design adequacy of the ABWR containment system. The results obtained from small-scale tests conducted within the scope of the ABWR design confirmed the applicability of Issue B-10 test data.

Based on this information and the staff's evaluation of the ABWR containment analysis in Section 6.2.1.6 of this report, the staff concludes that GE adequately addressed this issue for the ABWR design.

20.1.22 Issue B-17: Criteria for Safety-Related Operator Actions

Current plant designs are such that reliance on the operator to take action in response to certain transients is necessary. Consequently, it becomes necessary to develop appropriate criteria for safety-related operator actions (SROAs). The criteria would include a determination of actions that should be automated rather than manual and development of a time criterion for SROAs.

The review criteria for this issue are contained in ANSI/American Nuclear Society (ANS) 58.8-1984, "Time Response Design Criteria for Nuclear Safety Related Operator Actions," and ANSI/ANS 52.2-1983, "Nuclear Safety Criteria for the Design of Stationary Boiling Water Reactor Plants." Plants should perform task analysis, simulator studies, and analysis and evaluation of operational data to assess ESF and safety-related control system designs for conformance to the criteria. Where nonconformance is identified, modification of the design and hardware may be required.

SSAR Appendix 18E describes the program of human-factors-related activities conducted throughout the development of the ABWR plant system designs, including the development of the main control room (MCR) and remote shutdown system (RSS) designs. Appendix 18E describes the process through which the MCR and RSS human-system interface (HSI) design implementations will be conducted and evaluated through the application of human factors engineering (HFE) practices and principles.

The COL applicant is responsible for addressing B-17 as part of the detailed design implementation. The staff verified that GE has established a general COL action item in SSAR Section 18.8 (Item 18.8.1) to conduct the detailed HFE design according to design and implementation as defined by the ABWR certified design material (CDM) Table 3.1 Inspection, Tests, Analyses, and Acceptance Criteria (ITAAC) and SSAR Appendix 18E. The staff considers the SSAR to include commitments for the COL applicant to conduct the necessary analyses of the critical operator tasks. In addition, the staff verified that GE established another COL action item in SSAR Section 18.8 (Item 18.8.15) to evaluate the adequacy of the HSI to provide the necessary controls, displays, and alarms for the timely performance of critical tasks. This approach is acceptable to the staff as discussed in Section 18.7.2.2 of this report.

Based on this information, the staff concludes that GE adequately addressed this issue for the ABWR design.

20.1.23 Issue B-29: Effectiveness of Ultimate Heat Sinks

This issue addressed the adequacy of existing NRC mathematical models for the prediction of the ultimate heat sink (UHS) performance by comparing model performance with field data. The issue also addressed the selection of site-specific meteorological data for use as UHS design basis meteorology.

The DFSER stated that the design of the UHS is not within the scope of the ABWR design. GE provided several interface requirements to be used as guidance for the design of systems not within the ABWR design scope. The DFSER contained a lengthy discussion of these interface requirements and concluded that GE provided sufficient information to allow the COL applicant to provide a plant-specific response to Issue B-29. The plant-specific response was identified in the DFSER as COL Action Item 20.1-1.

Issue B-29 is categorized in NUREG-0933 as a licensing issue and as such, is not required to be considered in meeting the requirements of 10 CFR 52.47(1)(a)(iv). The purpose of the issue was to confirm the validity of the NRC models used to make an independent assessment of UHS design safety. NRC's studies have confirmed that the capability of NRC's models and existing guidance are adequate. The determination of the adequacy of a specific UHS design is to be made by the staff during the review process using the models and guidance. The design of the UHS for the ABWR will be provided by the COL applicant as part of the COL application in accordance with the interface requirements discussed in Section 9.2.5 of this report. The UHS design will be reviewed at the COL application stage. This approach is acceptable to the staff.

Upon further consideration, the staff determined that because Issue B-29 is a licensing issue and not required to be considered in meeting the requirements of 10 CFR 52.47(1)(a)(iv) as discussed above, no specific response to Issue B-29 is necessary from either GE or the COL applicant. On this basis, the staff concludes that DFSER COL Action Item 20.1-1 was not warranted and need not be addressed.

20.1.24 Issue B-32: Ice Effects on Safety-Related Water Supplies

This issue addressed the potential effects associated with extreme cold weather and ice buildup on the reliability of various plant water supplies. Of particular concern are events that could affect safety-related water systems and affect the ability of the plant operations staff to safely shut down the plant and provide adequate core cooling.

The DFSER contained a discussion of interface requirements for the UHS necessary to the reliability of this water source. The evaluation concluded that GE needed to require the COL applicant to use site weather conditions to establish the severe weather design envelope for the site. This was identified in the DFSER as COL Action Item 20.1-2.

The staff's evaluation in the DFSER also identified that the interface requirements for the UHS did not explicitly include the RSW system, portions of which have exposed piping that may be vulnerable to the adverse effects of ice, and required GE to address this concern. This was identified in the DFSER as Open Item 20.1-2. The DFSER stated that subject to the acceptable resolution of DFSER Open Item 20.1-2, the staff could conclude that the ABWR design addresses the concerns of Issue B-32.

In the DFSER, the staff also required GE to establish a requirement for the COL applicant to address the capability of the plant-specific RSW system and UHS designs to address the concerns of Issue B-32. This was identified in the DFSER as COL Action Item 20.1-3.

NUREG-0933 shows that Issue B-32 was subsumed by Issue 153 (discussed in Section 20.2.32 of this report). As such, Issue B-32 is not required to be considered separately in meeting the requirements of 10 CFR 52.47(a)(1)(iv). The UHS design and the out-of-scope portions of the RSW will be provided by the COL applicant as part of the COL application in accordance with the interface requirements discussed in Sections 9.2.5 and 9.2.15 of this report. The UHS design and the out-of-scope portions of the RSW will be reviewed at the COL application stage. This approach is acceptable to the staff. In addition, GE has included a COL action item in SSAR Section 2.3.2.14 for the COL applicant to demonstrate that safety-related facilities and water supply are not affected by ice flooding or blockage.

Because Issue B-32 was subsumed by Issue 153, upon further consideration of this information, the staff determined that no specific response to Issue B-32 is necessary from either GE or the COL applicant. On this basis, the staff concludes that

(1) DFSER Open Item 20.1-2 was not warranted and need not be addressed. Therefore, DFSER Open Item 20.1-2 is resolved.

(2) DFSER COL Action Items 20.1-2 and 20.1-3 were not warranted and need not be addressed.

See Section 20.2.32 of this report for the staff's evaluation of Issue 153.

20.1.25 Issue B-36: Develop Design, Testing, and Maintenance Criteria for Atmosphere Cleanup System Air Filtration and Adsorption Units for Engineered Safety Features Systems and Normal Ventilation Systems

This issue involves developing revisions to current guidance and staff technical positions regarding ESF and normal ventilation system air filtration and adsorption units. Any technological advances leading to better methods and/or standards for the design, testing, and maintenance for these systems in light water-cooled nuclear power plants need to be documented for NRC staff guidance and technical positions.

Guidance on controlling the release of gaseous radioactive effluents to the environment is contained in Revision 1 to RG 1.140, "Design, Testing, and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," and Revision 2 to RG 1.52, "Design, Testing and Maintenance Criteria for Postaccident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Plants." RG 1.52 provides guidance relating to the design, testing, and maintenance of post-accident ESFs, whereas RG 1.140 applies to the normal ventilation exhaust features.

GE provided the information pertinent to this issue in SSAR Sections 6.4 and 6.5. SSAR Section 6.5.1 states that the filter systems required to perform safety-related functions following a design-basis accident are the standby gas treatment system (SGTS) and the control room portion of the HVAC system.

The SGTS has the capacity to filter the gaseous effluent from the primary containment or from the secondary containment, when required, to limit the discharge of radioactivity to the environment to meet the guidelines of 10 CFR Part 100. GE described the SGTS power generation design basis, safety design basis, system design, SGTS operation (automatic and manual), and the design evaluation. Compliance with RG 1.52 is also discussed.

SSAR Sections 9.4.1 and 6.4 and Appendices 6A, 6B, 9C, and 9D describe the HVAC system pertinent to control room habitability. The system is designed in conformance with the requirements of GDC 19 and guidelines of RG 1.52 related to the design, testing, and maintenance of post-accident ESFs and ASME N509 and N510.

The radwaste building incinerator offgas exhaust is directed to a separate monitored vent, as described in SSAR Section 9.4.6.5.3. The COL applicant is to provide conformance with RG 1.140 for the radwaste building HVAC system.

Based on this information, the staff concludes that GE adequately addressed this issue for the ABWR design.

20.1.26 Issue B-55: Improved Reliability of Target Rock Safety-Relief Valves

The majority of valves in BWR pressure-relief systems are Target Rock SRVs. A significant number of failures of these valves have occurred, which include valves that (1) failed to open properly on demand, (2) opened spuriously and then failed to reseat properly, and (3) opened properly and then failed to reseat properly. The performance of these valves is under continual surveillance and the consequences of their failures are subject to review.

SSAR Section 19B.2.22 states that Target Rock SRVs are not to be used in the ABWR design. The SRVs to be used in the ABWR do not have a pilot stage such as that present in the Target Rock pilot-operated SRVs. The ABWR will use a direct-acting SRV design. Therefore, the mechanisms that cause the pilot valve to fail to open properly have been eliminated from the ABWR design.

Based on this information, the staff concludes that GE adequately addressed this issue for the ABWR design.

20.1.27 Issue B-56: Diesel Reliability

If a loss of normally available ac power from the offsite preferred systems occurs at a nuclear power plant, redundant onsite standby ac power sources (diesel generators) provide power for necessary safety functions, which include reactor core decay heat removal, emergency core cooling, and containment heat removal. Therefore, the reliability of the diesel generators is a major factor in ensuring acceptable plant safety.

This issue was promulgated by a review of licensee event reports (LERs) which indicated that the diesel generators at operating plants were demonstrating an average starting reliability of about 0.94 per demand. The NRC's goal for new plants, as expressed in Regulatory Guide 1.108, "Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants," is a diesel generator starting reliability of 0.99 per demand. As part of the resolution of Issue A-44 on station blackout (see Section 20.1.17 of this report), diesel reliability was considered one of the main factors affecting plant risk from station blackout. Thus, attaining and maintaining high diesel generator reliability was a necessary part of the resolution of Issue A-44. The Federal Register notice (53 FR 23217) which accompanied 10 CFR 50.63 (station blackout rule) indicated that the resolution of Issue B-56 would provide specific guidance for diesel generator reliability consistent with the resolution of Issue A-44. The specific guidance (the resolution of B-56) has been included as part of Revision 3 of RG 1.9, "Selection, Design, and Qualification of Diesel-Generator Units Used as Standby (Onsite) Electric Power Systems at Nuclear Power Plants" and as part of RG 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." RG 1.160 endorses the guidelines in NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants" (May 1993) as an acceptable method for meeting the requirements of 10 CFR 50.65 (the maintenance rule). With regard to systems and components such as diesel generators, 10 CFR 50.65 requires that:

- the performance or condition of diesel generators be monitored against established goals in a manner sufficient to provide reasonable assurance that the diesel generators are capable of fulfilling their intended functions,

- goals be established commensurate with safety and to take into account industry wide operating experience,

- appropriate corrective action be taken when the performance or condition of the diesel generators do not meet established goals,

- diesel generator performance and condition monitoring activities and associated goals and preventive maintenance activities be periodically evaluated, taking into account industry wide operating experience, and

- adjustments be made where necessary to ensure that the objective of preventing failure of diesel generators through maintenance is appropriately balanced against the objective of minimizing unavailability of diesel generators due to monitoring or preventive maintenance.

Revision 3 of RG 1.9 integrated into a single regulatory guide guidance previously addressed in other documents such as Revision 2 of RG 1.9, Revision 1 of RG 1.108, and Generic Letter 84-15.

With respect to diesel generator reliability (compliance with Issue B-56), the ABWR SSAR indicates the following:

a. Diesel generators will be required to be capable of reaching full speed and voltage within 20 seconds after the signal to start (Section 8.3.1.1.8.2(4) of SSAR Amendment 34).

b. For the COL license, the COL applicant will be required to have appropriate plant procedures for periodic testing of diesel generator start capability (Section 8.3.4.36 of SSAR Amendment 33). In particular,

   - Appropriate plant procedures will include the requirement that the interval between periodic start test for diesel generators will be no longer than 31 days (Section 6.5.1 of IEEE 387-1984), and

   - Diesel generator start testing may, once per 6 months, be replaced with a modified diesel generator start involving idling and gradual acceleration to synchronous speed as recommended by the manufacturer (Technical Specification guidelines in Section 16.3.8 (SR 3.8.1.2) of SSAR Amendment 34).

c. As part of the COL license, the COL applicant will be required to demonstrate the start reliability of the diesel generators (Section 8.3.4.2 of SSAR Amendment 34). Specifically,

   - The preoperational test program will demonstrate the required reliability by means of 25 start demands without failure on each installed diesel generator unit (Section 14.2.12.1.45.3(l) of SSAR Amendment 34 and Position C.2.3.1 of RG 1.9 (Rev. 3)).

   - Periodic testing at intervals of no longer than 31 days will commence within 31 days following completion of preoperational testing for diesel generator start reliability (Position C.2.3.2.1 of RG 1.9, Rev. 3)).

   - Performance criteria and goals for diesel generator start and loading reliability will be set at 0.975 per demand as part of the COL licensee's program for assuring diesel generator reliability. The SSAR indicates:

- Diesel generator start reliability of 0.986 per demand can be achieved as shown by industry experience (Section 19B.2.23 of SSAR Amendment 34).

- A diesel generator reliability of 0.975 per demand was assumed for station blackout considerations (Table 1C-1 of SSAR Amendment 32).

- A diesel generator reliability of 0.975 per demand was used in the ABWR PRA analysis (Table 1C-1 of SSAR Amendment 32).

- Performance criteria for both diesel generator reliability and unavailability will be established to assure that the performance or condition of the diesel generators is being effectively controlled through the performance of appropriate preventive maintenance, such that the diesel generator remains capable of performing its intended function (Section B of RG 1.160). The SSAR states:

  - Performance criteria for reliability will be met by the absence of a maintenance-preventable failure, or the occurrence of a single maintenance-preventable failure, followed by appropriate root cause determination and corrective action.

  - Performance criteria for unavailability will be met by having fewer unavailable hours, on a rolling 1-year basis, than required by the established performance criteria.

- If performance criteria is not met or a second diesel generator maintenance-preventable failure occurs, diesel generator performance goals will be established and the performance or condition of the diesel generators will be monitored in a manner sufficient to provide reasonable assurance that the diesel generators are capable of fulfilling their intended functions consistent with an appropriate balance between diesel generator reliability and unavailability (Section B of RG 1.160).

- Periodic adjustments will be made where necessary to ensure that the objective of preventing failures of the diesel generator through maintenance will be appropriately balanced against the objective of minimizing unavailability of the diesel generator due to monitoring or preventive maintenance (Section B of RG 1.160).

The staff concludes that a diesel generator testing and reliability program which meets the above described commitments will assure an acceptable level of diesel reliability in accordance with the objectives of Issues B-56 and A-44 and will meet the above defined requirements of 10 CFR 50.65. Because COL applicants will be required to have a diesel generator test and reliability program for their license which meets the above described commitments, or an acceptable alternative method, the staff concludes that GE adequately addressed this issue for the ABWR design.

20.1.28 Issue B-61: Allowable ECCS Equipment Outage Periods

This issue concerns establishing surveillance test intervals and allowable equipment outage periods, using analytically based criteria and methods for TS. The present TS allowable equipment outage intervals and test intervals were determined primarily on the basis of engineering judgement. Studies performed by the NRC on operating reactors indicated that from 30 to 80 percent of the ECCS system unavailability was a result of testing, maintenance, and allowed outage periods. Therefore, by optimizing the allowed outage period and the test and maintenance interval, the equipment unavailability and public risk can be reduced.

The review criteria for this issue consist of the techniques and methods available and the modeling from the Interim Reliability Evaluation Program (IREP) and the National Reliability Evaluation Program (NREP), the optimum equipment test intervals and allowable equipment downtimes. Also, since the ABWR evolved primarily from the BWR/6 design, most of the ABWR TS that control the ECCS outage periods, were modeled after NUREG-1433, "Standard Technical Specifications General Electric Plants, BWR/6." Furthermore, the criteria for resolution of this issue incorporated the accumulated experience from currently operating light water reactors.

SSAR Section 19B.2.24 states that the ABWR design incorporates many design enhancements to improve the operation and safety of the plant, and the most significant advances are in the area of ESFs. Based on the review of the information provided in the SSAR, the staff agreed with GE that the ABWR design includes redundancy for the ESF systems beyond that for currently operating BWR plants. This added redundancy allows for extending the associated completion times (CTs) beyond those specified for ESF systems on currently operating BWR plants, thereby facilitating maintenance on certain subsystems. This relaxation ranges from 8 hours to 14 days, and is based on probabilistic risk analysis (PRA), engineering evaluation, operating experience, and judgement for various components and combinations of components.

The PRA performed by GE used a system fault tree approach to quantify system accident sequences that result in severe core damage. Data related to the ESF used in the quantification included:

- Component failure rates
- Component repair times and maintenance frequencies
- Component inspection and test times and frequencies
- Allowable equipment completion times

The data used were in accordance with the guidance in NUREG/CR-2815, "Probabilistic Safety Analysis Procedures Guide," and basic failure rate data were obtained from the EPRI ALWR Requirements Document, supplemented by other nuclear sources. The staff reviewed the ABWR TS and concluded that the CTs contained therein are considered appropriate by the staff. The staff concludes that GE adequately addressed this issue for the ABWR design.

20.1.29 Issue B-63: Isolation of Low-Pressure Systems Connected to the Reactor Coolant Pressure Boundary

These generic issues, combined with Issue 96, "RHR Suction Valve Testing," (subsumed by Issue 105, "Interfacing Systems LOCA at LWRs") concern the common area of the ABWR, namely, the systems connected to the RCS pressure boundary that are considerably below the RCS operating pressure. The NRC has required that valves forming the interface between these high and low systems and the associated piping have sufficient redundancy to assure that the low-pressure systems are not subjected to pressure that exceeds their design limits.

Issue B-63 was resolved by the staff and implemented by MPA B-45, "Event V, Primary Coolant System Pressure Isolation Valves," which required leak testing of these check valves in accordance with plant-specific technical specifications. For the ABWR, SSAR Section 19B.2.25 states that test requirements for these valves are in SSAR Section 3.9.6, "Testing of Pumps and Valves." SSAR Table 3.9-9 provides a list of all the ABWR reactor coolant system pressure isolation valves. Section 3.9.6 states that all of the valves in this table will be periodically leak tested in accordance with the surveillance requirements in SSAR Chapter 16, "Technical Specifications." As discussed in Section 3.9.6.2.4 of this report, the staff reviewed the applicable information in SSAR Section 3.9.6 and concluded that it is acceptable.

On the basis the above discussion, the staff concludes that GE adequately addressed these issues for the ABWR.

20.1.30 Issue B-66: Control Room Infiltration Measurements

This issue addresses the concerns that the control room may not be in a safe, habitable condition under accident conditions and may not provide adequate protection for the plant operators against the effects of accidental releases of airborne radioactivity and toxic gases. The rate of air infiltration into the control room is a significant factor in maintaining habitability, and the NRC measured air exchange rates in selected operating reactor plant control rooms to improve the data base for evaluating its effects. No new design requirements were established by the NRC as a result of this and other work related to control room habitability in an accident. However, more specific review procedures were incorporated in SRP Sections 6.4, 6.5.1, and 9.4.1, including the habitability review provisions of TMI Action Plan Item III.D.3.4 (see Section 20.4.87 of this report) regarding analyses of toxic gas concentrations and operator exposures from airborne radioactive material and direct radiation, to ensure more effective implementation of existing requirements.

The review criterion for the resolution of this issue is that the control room ventilation and air conditioning systems be designed to maintain the room's environment within acceptable limits for the operation, testing, and maintenance of the unit controls and the uninterrupted safe occupancy during normal and accident conditions.

SSAR Sections 9.4.1, 6.4, and 19B.2.26 state that the systems incorporated in the design of the control room ventilation and air conditioning will meet the intent of the guidance given in SRP Sections 6.4, 6.5.1, and 9.4.1. More specifically, these systems are designed to meet the intent of the guidance given in SRP Sections 6.4, 6.5.1, 9.4.1, and 15.6.5.5 (all Rev. 2). Under normal operation, the control room HVAC provides HVAC functions and pressurization inside the main control area envelope (MCAE) using a combination of filtered outdoor air and recirculated indoor air.

The emergency recirculation system consists of an electric heating coil, a prefilter, pre-high-efficiency particulate (HEPA) filter, charcoal absorber, post-HEPA, and two 100-percent capacity circulating fans. Independent and separate discharge to and return from the MCAE to each filtration unit is provided. All control room HVAC equipment, including the ductwork (which is termed ESF), and surrounding structures are seismic Category I design and operable during loss of offsite power supply.

ABWR design features regarding control room habitability and the HVAC are described in the SSAR Sections 9.4.1 and 6.4. The filtration units conform to the guidance of RG 1.52, and the requirements of ANSI ASME N509 and ASME N510, "Testing of Nuclear Air-Cleaning Systems." SSAR Appendices 9C and 9D provide detailed information regarding ABWR conformance with RG 1.52 and SRP Table 6.5.1-1.

In Amendment 32, GE revised SSAR Sections 9.4.1.1.4 and 9.4.1.1.5 to state that the galvanized steel (American Society for Testing and Materials (ASTM) A526 or A527) is used for outdoor air intake and exhaust ducts and all other ducts (CRHS HVAC system) are welded, black steel ASTM A570, Grade A or Grade D. In Amendment 34, GE revised SSAR Section 9.4.1.1.5 to state that unfiltered in-leakage testing will be periodically performed on all ductworks and housing outside the MCAE in accordance with ASME N510. Based on the information provided in SSAR Sections 19B.2.26, 9.4.1, and 6.4 and Appendices 9C and 9D that the appropriate plant-specific procedures will be developed to address MCR infiltration measurements, to preclude any unfiltered in-leakage not credited in dose analysis, in accordance with the above requirements.

On the basis of this information, the staff concludes that GE adequately addressed this issue for the ABWR design.

20.1.31 Issue C-1: Assurance of Continuous Long-Term Capability of Hermetic Seals on Instrumentation and Electrical Equipment

This issue concerns the long-term capability of hermetically sealed instruments and equipment that must function in post-accident conditions. More specifically, certain classes of instrumentation that are equipped with seals are sensitive to steam and vapor. If the seals become defective as a result of personnel error in the maintenance of such equipment, such errors could lead to the loss of a seal and of equipment functionality. The objective of this issue is to establish confidence that sensitive equipment has an effective seal for the lifetime of the plant.

The review criterion for this issue is compliance with the review criteria of SRP Section 3.11 for environmental qualification of electrical equipment.

SSAR Section 19B.2.17 refers to SSAR Section 3.11, "Environmental Qualification of Safety-Related Mechanical and Electrical Equipment," which defines the environmental conditions with respect to limiting design conditions for all safety-related mechanical and electrical equipment. Safety-related equipment located in a harsh environment must perform its proper safety function during normal, abnormal, test, DBA, and post-accident environments, as applicable. A list of all safety-related electrical and mechanical equipment required for safe shutdown that is located in a harsh environment area will be included in the Environmental Qualification Document as stated in SSAR Section 3.11.6.1.

Environmental conditions for the zones where safety-related equipment is located are calculated for normal, abnormal, test, accident, and post-accident conditions and are documented in SSAR Appendix 3I, "Equipment Qualification Environmental Design Criteria." Environmental conditions are tabulated by zones contained in the referenced building arrangements.

Safety-related electrical equipment that is located in a harsh environment is qualified by test or other methods, as described in IEEE 323 and permitted by 10 CFR 50.49(f). The qualification methodology is described in detail in the NRC-approved report, NEDE-24326-1-P. This report also addresses compliance with the applicable portions of 10 CFR Part 50, Appendix A and the quality assurance criteria of 10 CFR Part 50, Appendix B. Additionally, the report describes conformance to NUREG-0588 and the RGs and IEEE standards referenced in SRP Section 3.11.

Based on the above discussion, since safety-related electrical equipment for the ABWR will be qualified in accordance with applicable guidance, including NUREG-0588, the staff concludes that GE adequately addressed this issue for the ABWR design.

Details on the staff's evaluation of environmental qualification of safety-related electrical equipment are provided in Section 3.11 of this report.

20.1.32 Issue C-8: Main Steam Line Leakage Control Systems

Dose calculations in 1975 by NRC's former Accident Analysis Branch indicated that operation of the main steam isolation valve leakage control system (MSIVLCS) required for some BWRs could result in higher offsite accident doses than if the system were not used and the integrity of the steam lines and condenser was maintained. This issue was initiated to investigate whether the MSIVLCS recommended in RG 1.96, "Design of Main Steam Isolation Valve Leakage Control Systems for Boiling Water Reactor Nuclear Power Plants," was desirable. This issue was resolved without the establishment of new requirements. The main steam system of BWR designs is expected to meet GDC 54. RG 1.96 describes a means acceptable to the staff for implementing GDC 54 with regard to the design of an MSIVLCS that ensures that total radiological effects do not exceed the guidelines of 10 CFR Part 100 in the event of a postulated design-basis LOCA.

Since this issue was resolved without the establishment of new requirements, it does not need to be addressed in the ABWR design certification review. A discussion of the ABWR approach to the MSIVLCS issue is provided here for information purposes only.

The ABWR design does not incorporate a MSIVLCS, but utilizes an alternative approach as permitted by RG 1.96. RG 1.96 indicates that an LCS is not required if the main steam line leakage path can be relied on to remain intact and capable of providing significant dose reduction factors for postulated accident conditions. A description and the staff's evaluation of the ABWR's method of containing and holding up MSIV leakages following a design-basis LOCA are addressed in Section 10.3.1 of this report. Evaluation of the seismic and radiological analyses of the main steam and condenser systems' capability to perform this post-LOCA function is provided in Sections 3.2.1 and 15.4.4.2, respectively, of this report.

20.1.33 Issue C-10: Effective Operation of Containment Sprays in a LOCA

This issue deals with the effectiveness of various containment sprays to remove airborne radioactive materials that could be present within the containment following a LOCA. This concern includes the possible damage to equipment located inside the containment because of inadvertent actuation of the sprays.

The review criteria for this issue are that the containment spray system will be designed to meet the requirements of GDC 41, 42, and 43 of 10 CFR Part 50, Appendix A, related to fission product removal, periodic inspection, and functional testing, respectively, by conforming to the guidance of SRP Section 6.5.2, Revision 2.

SSAR Section 19B.2.28 states that the RHR system provides two independent containment spray cooling systems (on loops B and C), each having a common header in the wetwell and a common spray in the drywell, and sufficient capacity for containment depressurization by removing heat and condensing steam in both the drywell and wetwell air volumes following a LOCA. All components of the RHR containment spray system can be inspected and tested during normal plant operation or during refueling and maintenance outages. The ABWR design does not take credit for any fission product removal provided by the drywell and wetwell spray portion of the RHR system. The removal of fission products is controlled by the SGTS, which has the redundancy and capability to filter the gaseous effluent from the primary and secondary containment. However, the drywell sprays off the RHR system also function to provide removal of fission products during a LOCA, as well as in the event of failure of the drywell head. The drywell spray is initiated by operator action post-LOCA in the presence of high drywell pressure and is terminated by operator action. It is also terminated automatically as the RHR injection valve starts to open.

The water in the 304L stainless-steel-lined suppression pool is maintained at high purity by the suppression pool cleanup (SPCU) system. The pH range is maintained between 5.3 and 8.6 to minimize any corrosive attack on the pool liner over the life of the plant. The protective epoxy coatings applied on the carbon steel containment liner, internal steel structures and equipment inside the drywell and wetwell meet the guidance of RG 1.54, "Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants," and are qualified using the tests in accordance with ANSI N101.4, "Quality Assurance for Protective Coatings Applied to Nuclear Facilities."

On the basis of this information, the staff concludes that GE adequately addressed this issue for the ABWR design.

20.1.34 Issue C-17: Interim Acceptance Criteria for Solidification Agents for Radioactive Solid Wastes

This issue concerns lack of criteria for acceptability of solidification agents for radioactive solid wastes.

The review criteria for this issue are contained in 10 CFR Part 61, which was published in the Federal Register on December 27, 1982, and includes Section 61.56, which addresses waste characteristics. Also, BTP ETSB-11-3, "Design Guidance for Solid Radioactive Waste Management Systems Installed in Light-Water-Cooled Nuclear Power Plants," was developed under TMI Action Plan Item IV.C.1 before the issuance of 10 CFR 61.56. Evaluation of this issue is based on these documents.

SSAR Section 19B.2.29 states that the ABWR design will comply with the requirements of the 10 CFR 61.56 regarding waste characteristics. The establishment and implementation of a process control program (PCP) for solidifying the evaporator concentrates (the applicable plant waste will be solidified) using an approved solidification agent is dependent on the as-procured equipment for the ABWR standard design. Therefore, the staff will review the PCP on a plant-specific basis. The staff verified that GE established a COL action item in SSAR

Section 11.4.3.1 to provide a PCP for solidifying the evaporator concentrates using an approved solidification agent and to demonstrate that the wet solidification process will result in a product that complies with 10 CFR 61.56 regarding waste characteristics. The staff finds this approach acceptable, and therefore, concludes that GE adequately addressed this issue for the ABWR design.

20.1.35 Other Issues

20.1.35.1 Issue A-43: Containment Emergency Sump Performance

This issue concerns the availability of adequate recirculation cooling water following a LOCA when long-term recirculation of cooling water (from the PWR containment sump or the BWR RHR system suction intake) must be initiated to prevent core melt. This water must be sufficiently free of LOCA-generated debris and potential air ingestion so that pump performance is not impaired, thereby seriously degrading long-term recirculation flow capability. Issue A-43 was resolved with the issuance of RG 1.82 (Rev. 1), "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident."

NUREG-0933, Appendix B indicates that this issue is not applicable to BWRs, therefore, GE did not specifically address it in SSAR Appendix 19B. However, the staff evaluated GE's compliance with RG 1.82 (Rev. 1) and supplemental staff positions and found it to be acceptable. Refer to Sections 6.2.1.9 and 6.2.2 of this report for the staff's evaluation.

20.2 New Generic Issues

This section addresses staff evaluation of GSIs that are categorized as "new generic issues" in NUREG-0933. All the following issues, with the exception of Issue 130 (which was evaluated in the DFSER) are relevant to the ABWR design. Issue 130 is discussed here for continuity only.

20.2.1 Issue 15: Radiation Effects on Reactor Vessel Supports

This issue addresses the potential problem of radiation embrittlement of reactor vessel support structures. Neutron damage of structural materials causes embrittlement that may increase the potential for propagation of flaws that might exist in the materials. The potential for brittle fracture of these materials is typically measured in terms of the material's nil ductility transition temperature (NDTT), which is the lowest temperature at which the material would not be susceptible to failure by brittle fracture. As long as the operating environment in which the materials are used has a higher temperature than the material's NDTT, no failure by brittle fracture would be expected. Recent studies of steel materials indicate that the NDTT may shift upwards (towards the operating temperature) as a result of exposure to neutron irradiation.

For a structural material to be susceptible to a brittle failure, several conditions must be met simultaneously. These conditions are that (a) there must be a flaw of critical size, (b) there must be a load that develops a tensile stress, and (c) the service temperature must be at or below the NDTT of the material. This last condition is affected by the neutron flux to which the element is exposed. Thus, the evaluation of the resolution of this issue depends on the assessment of the conditions stated above.

The ABWR RPV support consists of a support skirt bolted to the support pedestal. The skirt is located below the core beltline and slightly below the core support plate. It follows that the skirt is in a low neutron flux area because it is located below the core beltline. Additional shielding is provided by water flow between the vessel shroud and the vessel wall. In this situation, the effects of neutron flux would be negligible. Based on this information, the staff determined that this issue is not applicable to the ABWR design and concludes that GE adequately addressed this issue for the ABWR design.

20.2.2 Issue 23: Reactor Coolant Pump Seal Failures

This issue concerns the high rate of reactor coolant pump (RCP) seal failures that challenge the makeup capacity of the ECCS in pressurized water reactors (PWRs) which could result in a small-break loss-of-coolant-accident (SBLOCA) and possibly in core damage. RCP seal failures in BWRs occur at a frequency similar to that experienced in PWRs, but the operating experience indicates that the problem in BWRs is mitigated by smaller leak rate, larger RCIC, high-pressure coolant injection (HPCI), and feedwater makeup capabilities.

The review criteria for RCP seal acceptability is to limit the possibility of an SBLOCA (which might lead to core damage) resulting from an RCP shaft seal failure. In particular, susceptibility of the auxiliary systems to failure because of an SBO should be addressed.

SSAR Section 1A.2.30 states that the ABWR wet motor RIPs do not include seals. During a loss of preferred power (LOPP), the RIPs shutdown automatically. There are no shaft seals which require cooling water restoration. Based on this information, the staff determined that this issue does not apply to the ABWR design and concludes that GE adequately addressed this issue for the ABWR design.

20.2.3 Issue 25: Automatic Air Header Dump on BWR Scram System

This issue concerns the slow loss of control air pressure in the scram system of BWRs. Air pressure dropping at a certain rate will first allow some of the control rod drive (CRD) scram outlet valves to open slightly, thus filling the scram discharge volume (SDV) with water, but allowing little or no control rod movement. Eventually, the rods will try to scram but the scram will be impaired. Meanwhile, the dropping air pressure may cause a transient (e.g., via controller lockup), which would normally call for a scram.

The review criteria for this issue are stated in an NRC memorandum for G. Lainas, et al., from P. Check, "BWR Scram Discharge System Safety Evaluation," dated December 1, 1980.

SSAR Section 19B.2.32 states that the ABWR is different from other BWRs in that there is no SDV employing the locking piston control rod drive in this design. For the ABWR fine motion control rod drive (FMCRD) design, scram water is discharged directly into the reactor vessel, instead of the discharge volume, as was done in previous BWR designs. The ABWR has no SDV as does previous BWR designs.

Two other features are incorporated in the design of the ABWR:

- A scram air header low-pressure alarm to alert the operator of a low pressure in the header. The pressure in the header is maintained higher than that at which the scram valves start to open and the set point is set at a sufficient margin to alert the operator to take corrective action.
- A rod block and alarm initiated by low pressure and a scram initiated by low-low pressure in the common header supplying the charging water to the scram accumulators. The accumulators have sufficient water volume to scram the associated control rods as long as the CRD system pump maintains the pressure in the charging header above the minimum required accumulator charging pressure. If pressure in the header drops below the acceptable level, the instrumentation located in the charging header will initiate an immediate scram. Thus, the accumulator charging header low pressure causes the automatic shutdown before the accumulator is depleted.

The ABWR design incorporates two features to prevent the loss or impairment of the scram function because of slow loss of control air in the system: (1) a low pressure alarm to alert the operator to trouble in the scram air header and (2) an accumulator charging header low pressure scram to automatically shut down the plant before the accumulator is depleted. Based on this information, the staff concludes that GE adequately addressed this issue for the ABWR design.

20.2.4 Issue 29: Bolting Degradation or Failure in Nuclear Power Plants

This issue addresses degradation of bolts in nuclear power plants, especially those constituting an integral part of the primary pressure boundary, such as closure studs and bolts on reactor vessels, reactor coolant pumps, and other safety-related equipment and components. Failure of these bolts or studs could result in a loss of reactor coolant and thus jeopardize the safe operation of a plant. There is also a concern regarding bolts used as component supports or embedment anchor bolts that are essential for withstanding transient loads resulting from abnormal or accident conditions. This issue was resolved without the establishment of any new requirements.

The review criteria used for evaluating this issue are taken from NUREG-1339, "Resolution of Generic Safety Issue 29: Bolting Degradation or Failure in Nuclear Power Plants." More specifically, proven bolting designs and selection of proper materials and fabrication techniques should be employed to assure acceptable performance of bolts and studs used in vital areas of nuclear power plants.

In SSAR Section 19B.2.62, GE states that for the ABWR designs only proven materials for the specific application and environment will be employed. The materials will be selected after evaluation of the potential for corrosion wastage and IGSCC. Also, the RCPB components and their integral bolts, including the reactor vessel, reactor coolant pumps, and piping will be fabricated, tested, and installed in accordance with the requirements of the ASME Code, Sections III and XI. The ABWR design will also comply with the guidelines of NUREG-1339; EPRI NP-5769, "Degradation and Failure of Bolting in Nuclear Power Plants;" and GL 91-17, "Generic Safety Issue 29, 'Bolting Degradation of Failure in Nuclear Power Plants'."

This approach is acceptable to the staff and, therefore, the staff concludes that GE adequately addressed this issue for the ABWR design.

20.2.5 Issue 40: Safety Concerns Associated With Pipe Breaks in the BWR Scram System

This issue is concerned with a possibility of a break or leak in the scram discharge volume (SDV) during a reactor scram. If such a leak or break develops, it would result in the release of water and steam at 100 °C (212 °F) into the reactor building at a maximum flow rate of 2,082 Lpm (550 gpm) and it is postulated to result in 100-percent relative humidity in the reactor building. This could be mitigated by closing the scram exhaust valves that are located on the hydraulic control units, but this depends upon ability to reset the scram, which cannot be absolutely assured immediately following the scram. Therefore, a rupture of the SDV could result in an unisolable break outside of primary containment. This break is postulated to threaten emergency core cooling equipment by flooding areas in which this equipment is located and by causing ambient temperature and relative humidity conditions for which this equipment is not qualified.

The review criteria for this issue are stated in the NUREG-0803, "Generic Safety Evaluation Report Regarding Integrity of BWR Scram System Piping," which provides guidance to ensure pipe integrity, detection capability, mitigation capability, and qualification of the emergency equipment to the expected environment.

This issue is not applicable to the ABWR design. SSAR Sections 4.6 and 19B.2.33 state that for the FMCRD design, scram water is discharged through the drive directly into the reactor vessel. There are no CRD withdraw lines or SDV as used in previous BWR designs employing the locking piston control rod drive. Consequently, the safety concerns associated with pipe break are not applicable to the ABWR. The staff concludes that GE adequately addressed this issue for the ABWR design.

20.2.6 Issue 45: Inoperability of Instrumentation Due to Extreme Cold Weather

This issue was raised after an event at Arkansas Nuclear One, Unit 2, in which all four refueling water storage tank (RWST) instrumentation channels were lost when the level transmitters froze. The system heat-tracing circuit was deenergized because the main line fuse was removed. This situation would have prevented the automatic changeover of the ECC from the injection to the recirculation mode under LOCA conditions, that is, a loss of safety function could have occurred. Typical safety-related systems employ pressure and level sensors that use small bore instrumentation lines. Most operating plants contain safety-related equipment and systems, parts of which are exposed to the ambient environment. These lines contain liquid that is susceptible to freezing. Where systems or components and their associated instrumentation are exposed to subfreezing temperatures, heat tracing and/or insulation should be used to mitigate the effects of cold temperatures.

The review criterion for this issue is that the fluid in safety-related instrument sensing lines must be protected from freezing and maintained above the precipitation point. Guidance on the design of protective measures against freezing in instrument lines of safety-related systems are stated in the RG 1.151, "Instrument Sensing Lines." RG 1.151 endorses and augments the Instrument Society of America (ISA) standard ISA-S67.02 (1980), "Nuclear-Safety-Related Instrument Sensing Line Piping and Tubing Standards for Use in Nuclear Power Plants." RG 1.151 augments ISA-S67 by indicating that freezing temperatures be added to the environmental and installation conditions of the standard and that special considerations in the design of instrument sensing lines include freezing temperatures. Further guidance is provided in SRP Sections 7.1, Revision 3; Section 7.1, Appendix A, Revision 1; Section 7.5, Revision 3; and Section 7.7, Revision 3. SRP Section 7.1 provides for identification of the I&C systems important to safety. The acceptance criteria consist of the GDC (identified is SRP Table 7.1) and IEEE 279, "Criteria for Protection Systems for Nuclear Power Generating Stations." SRP Section 7.5 provides that the information systems important to safety provide the operator with the information on the status of the plant to allow manual safety actions to be performed when necessary. SRP Section 7.7 provides that the control systems used for normal operation, that are not relied upon to perform safety functions, but which control plant processes having a significant impact on plant safety, be acceptable and meet the relevant requirements of GDC 13 and 19.

As a proposed resolution of this issue, SSAR Section 19B.2.34 states that all safety-related systems and components used in the ABWR design, including instrument sensing lines, will be located in temperature-controlled environments. These environments will be maintained above the freezing (or precipitation) point of the contained fluid. SSAR Appendix 3I demonstrates that the temperature of these environments is not expected to be below 10° C (50 °F). The section also states that the ABWR will meet the guidance of RG 1.151.

Based on this information, the staff concludes that GE adequately addressed this issue for the ABWR design.

20.2.7 Issue 51: Proposed Requirements for Improving the Reliability of Open Cycle Service Water Systems

This issue addresses the subject of service water system (SWS) fouling at operating plants by aquatic bivalves. The SWS is the UHS that, during an accident or transient, cools the reactor building component cooling water heat exchangers, which in turn cool the RHR heat exchangers, as well as provide cooling for safety-related pumps and area cooling coils. Fouling of the safety-related SWS either by mud, silt, corrosion products, or aquatic bivalves has led to plant shutdowns, reduced power operation for repairs and modifications, and degraded modes of operation.

The review criteria for this issue consist of elimination of the possible effects of fouling of SWS and UHSs.

SSAR Section 19B.2.35 indicates that the COL applicant is given specific requirements and guidance on achieving this goal, including instructions to consider designs and new requirements that further mitigate the fouling effects. Additionally, the COL applicant is directed to investigate the problem with ice as a flow blockage mechanism and to dispose of and/or dissolve such ice, as required.

The staff studied the conditions that allow fouling and compared alternative surveillance and control programs to minimize SWS fouling. The staff's technical findings were published in NUREG/CR-5210, "Technical Findings Document for Generic Issue 51: Improving the Reliability of Open-Cycle Service-Water Systems."

SSAR Section 19B.2.35 states that the design basis for the SWS is in accordance with the EPRI ALWR Requirements Document and RG 1.27, "Ultimate Heat Sink for Nuclear Power Plants" (Rev. 2). The specifics of the design basis are given in SSAR Section 19B.2.35 as follows. Section 19B.2.35 states that the direct service water will not be used for component cooling. A closed-loop component cooling system will be used to transfer heat from the component heat loads via a heat exchanger to the SWS and ultimate heat sink. This design will minimize the number of pieces of equipment that could be in contact with the problem-causing service water. Additionally, the COL applicant will treat raw service water to reduce the effects of mud, silt and/or organisms, will select the materials for piping, pumps, and heat exchangers to offer greater resistance to the probable water chemistry, and will provide for inspections and replacements of piping during plant life. The COL applicant will also provide sufficient redundancy of makeup pumps for the UHS to allow for malfunction of one of them according to the guidelines of RG 1.27 (Rev. 2) and provide the safety-related portions of the systems to meet the design bases during a loss of power. These systems will be designed to perform their cooling function assuming a single active failure in any mechanical or electrical system.

GE identified this issue as a COL applicant action in SSAR, Section 9.2.15.2.2, "Power Generation Design Bases (Interface Requirements)," and in SSAR Section 9.2.15.2.3, "System Description (Conceptual Design)." The staff verified that GE established a COL action item (Item 9-12) in SSAR Table 1.9-1. The staff also verified that SSAR Section 9.2.17.2, "COL License Information, Reactor Service Water System Requirements," addresses measures that will be used to prevent organic fouling, erosion, and corrosion. This approach is acceptable to the staff. However, the staff will review the COL applicant's proposed resolution of this issue on a case-by-case basis.

Based on this information, the staff concludes that GE adequately addressed this issue for the ABWR design.

20.2.8 Issue 57: Effects of Fire Protection System Actuation on Safety-Related Equipment

This issue addresses fire protection system (FPS) actuations that have resulted in adverse interactions with safety-related equipment at operating nuclear power plants. Events have shown that safety-related equipment subjected to FPS water spray could be rendered inoperable and that numerous spurious actuations of the FPS have been initiated by operator testing errors or by maintenance activities, steam, or high humidity in the vicinity of FPS detectors. The NRC issued Office of Inspection and Enforcement (IE) Information Notice 83-41, "Actuation of Fire Suppression System Causing Inoperability of Safety-Related Equipment," to alert licensees and provide recent examples in which FPS actuations caused damage or inoperability of systems important to safety. In addition, the staff is considering the need for modifying FPS requirements or licensing review procedures.

The review criterion for this issue is to verify, per GDC 3 of 10 CFR Part 50, Appendix A, that a fire detection and fighting system of appropriate capacity shall be provided and designed to minimize the adverse effects of fires on structures, systems, and components important to safety. It further states that fire fighting systems shall be designed to assure that their rupture or inadvertent operation does not significantly impair the safety capability of those structures and components. Criteria in BTP CMEB 9.5-1, Section C.7.1, state that automatic fire suppression should be installed to combat any diesel generator or lubricating oil fires; such systems should be designed for operation when the diesel is running without affecting the diesel.

SSAR Section 19B.2.36 states that the ABWR incorporates design features that prevent the inadvertent actuation of fire protection systems and limit the effects of water spray onto safety-related equipment. The automatic fire suppression systems protecting the safety-related equipment are of the preaction automatic type that require the detection of a fire by infrared and/or rate-of-rise heat detectors, and that require the opening of the fusible link sprinkler heads. Furthermore, each division has its own dedicated detection and actuation equipment for the control of the automatic closed head sprinklers in that divisional area. The first of the two actuation signals required to initiate the fire suppression system will annunciate an alarm to alert the operator to any potential problem. The operator has the capability of terminating the flow of fire suppressant locally by manual isolation valve.

The safety-related equipment that could be damaged because of flooding discharge of a sprinkler system is further protected by being elevated and by providing adequate drainage.

Based on this information, the staff concludes that GE adequately addressed this issue for the ABWR design.

20.2.9 Issue 67.3.3: Steam Generator Staff Actions - Improved Accident Monitoring

This issue addresses several weaknesses in accident monitoring that were observed at the Ginna Nuclear Plant. More precisely, they included (1) non-redundant monitoring of RCS pressure, (2) failure of the position indication for the steam generator relief and safety valves, and (3) the limited range of the charging pump flow indicator for monitoring charging flow during accidents. Under these conditions, it is difficult for the operating personnel to decide what corrective action they should take in situations when such an action is needed.

The review criteria used for this issue are those contained in the RG 1.97, (Rev. 3), "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions and Environs During and Following an Accident." RG 1.97 describes the methods acceptable to the NRC staff for complying with the Commission's requirements to provide instrumentation to monitor plant variables and systems during and following an accident in a nuclear plant. It describes the parameters that are necessary for the operating personnel to be monitored so that an appropriate corrective action could be taken. It also sets the requirements for the instrumentation to be functional in case of an emergency.

SSAR Section 19B.2.37 states that the ABWR has implemented into its basic design the guidance of RG 1.97 and the TMI Action Plan requirements of NUREG-0737 and NUREG-0737, Supplement 1. Section 19B.2.37 refers also to Section 7.5.1.1, Table 7.5.2, and SSAR Section 18.2. Section 7.5.1.1 describes the design features of the plant that indicate its conformance to the provisions of RG 1.97. Plant variables are defined in conformance to the definitions contained in the RG and consist of five "types" and three "categories," also according to the RG. SSAR Table 7.5-3 lists 12 Type A "variables," such as neutron flux, RPV water level, and RPV pressure.

SSAR Section 18.2 states that during all phases of normal plant operation, abnormal events, and emergency conditions, the ABWR will be operable by two reactor operators. In addition, the operating crew will include one assistant control room shift supervisor, one control room shift supervisor, and two or more auxiliary equipment operators. Four licensed operators will be on shift at all times, consistent with the staffing requirements of 10 CFR 50.54(m). The main control room staff size and roles will be evaluated and implemented by the COL applicant. The staff verified that GE established a COL action item in SSAR Section 18.8.2 regarding the adequacy of control room staffing. The design acceptance criteria (DAC)/ITAAC will further ensure compliance with the HFE design and implementation process with regard to the control room and the remote shutdown station.

Based on this information, the staff concludes that GE adequately addressed this issue for the ABWR design.

20.2.10 Issue 75: Generic Implications of ATWS Events at the Salem Nuclear Plant

On two occasions in 1983, Salem Unit 1 failed to scram automatically because both reactor trip breakers (RTBs) failed to open on receipt of an actuation signal. In both cases, the unit was successfully tripped manually. The failure of the breakers was attributed to excessive wear from improper maintenance of the undervoltage relays that receive the trip signal from the protection system and result in the breaker's opening mechanically.

Three separate actions were initiated to address this problem. One was plant specific and was addressed before restart of Salem Unit 1. The second action was an investigation of the Salem events and the circumstances leading to them. The third action was the formation of an NRC task force to study the overall generic implications of this event. The results of the task force's work were reported in NUREG-1000, Volume 1, "Generic Implications of ATWS Events at the Salem Nuclear Power Plant."

The results of investigations of the Salem Generic Implications Task Force published in NUREG-1000, Volume 1, were later outlined in proposed actions for licensees, applicants, and the NRC staff in SECY-83-248, "Generic Actions for Licensees and Staff in Response to the ATWS Events at Salem Unit 1." Furthermore, in July 1983, NRC issued the required actions for licensees and applicants in GL 83-28, "Required Actions Based on Generic Implications of Salem ATWS Events," and the internal staff actions as NUREG-1000, Volume 2.

SSAR Section 19B.2.38 describes acceptable criteria pertinent to the issue that would satisfy the regulatory requirements. These criteria state that:

- The plant must have a program for a post-trip review of unscheduled reactor shutdowns.
- The plant must have a program for safety-related equipment classification and vendor interface.
- The plant must have a program for post-maintenance operability testing.
- The plant must have a program to control vendor-related modifications, preventive maintenance, and surveillance for reactor trip breakers.

SSAR Section 19B.3.1 indicates that the COL applicant is responsible for providing resolutions of issues identified as "COL Applicant" in the "Safety Issues Index" of SSAR Appendix 19B. The SSAR indicates that issues are identified for COL action because they pertain to operating personnel issues, operating procedures, and other topics beyond the scope of the ABWR design certification review. SSAR Section 19B.1.1 lists specific documentation the COL applicant is to provide for resolution of such issues.

GE identified this issue for COL applicant action in the "Safety Issues Index." The staff verified that GE established a COL action item (Item 19-28) in SSAR Table 1.9-1 to address unresolved generic and TMI safety issues. This approach is acceptable to the staff. The staff will review the COL applicant's proposed resolution of this issue on a case-by-case basis.

20.2.11 Issue 78: Monitoring of Fatigue Transient Limits for Reactor Coolant System

This issue concerns the fact that repeated thermal cycling of RCS components produces some degree of fatigue degradation of the material that could lead to failure, thereby increasing the likelihood of a LOCA. The staff expressed the concern that for many older operating reactors, no TS requirements exist for monitoring the actual number of transient occurrences. For newer operating reactors, it is required that the licensees keep account of the number of transient occurrences to ensure that transient limits, based on design assumptions, are not exceeded. Additionally, the staff determined that the fatigue curves used in ASME Code, Section III, may not be adequate in taking into account environmental effects. Recent data indicated that the existing code fatigue curves may have less margin than originally intended when the effects of fatigue induced by the operating environment were considered.

The review criteria for this issue are that plants implement TS to monitor plant transients and environmental effects on the fatigue life of ASME Code, Section III, Class I carbon steel piping. For ASME Code, Class 2 and 3 or Quality Group D components that are subjected to cyclic loading, an appropriate analysis is required.

SSAR Section 19B.2.39 states that the ABWR TS 5.7.2.9 requires that the monitoring of plant transients be performed to ensure that RCPB components are maintained within design limits, and that environmental effects will be included in the design bases for materials. The calculated core damage frequency (CDF) includes the environmental effects on fatigue resistance of materials.

The tentative procedure to evaluate the environmental effects on material fatigue that is currently used for a foreign BWR plant design was presented to the staff during an audit at the GE offices in San Jose, California, on March 23 through 26, 1992. In SSAR Section 3.9.3.1.1.7, GE commits to perform additional evaluations for environmental effects on the fatigue design of ASME Code, Section III, Class 1 carbon steel piping in accordance with GE document 408HA414 (non-proprietary). As discussed in Section 3.12.5.7 of this report, the staff found that the conditions and the methodology proposed by GE constitute supplemental guidelines that enhance the design margins beyond the requirements of the ASME Code, Section III, for fatigue evaluation and are acceptable.

Based on this information, the staff concludes that GE adequately addressed this issue for the ABWR design.

20.2.12 Issue 82: Beyond Design-Basis Accidents in Spent Fuel Pools

A typical spent fuel storage pool with high-density storage racks can hold about five times the fuel in the core. If the pool were to be drained of water, the discharged fuel from the last two refuelings might still be "fresh" enough to melt under decay heat. Additionally, the zircaloy cladding of this fuel could be ignited during the heatup. The resulting fire, in a pool equipped with high-density storage racks, could spread to most or all of the fuel in the pool. This could cause a release of fission products from the fuel matrix.

The review criteria for this issue are contained in SRP Sections 9.1.2 through 9.1.5 and RG 1.13, "Spent Fuel Storage Facility Design Basis." The highlights of the guidelines contained in these documents are as follows:

- The spent fuel pool should be a seismic Category I structure.
- The spent fuel pool should be designed to withstand heavy load drops without pool leakage that would uncover the top of the fuel. The spent fuel pool will be arranged to prevent cask movement over the pool.
- There should be no connections to the pool that could allow the pool to be drained below the minimum level over the spent fuel.

Although the likelihood of the complete draining of the spent fuel pool is low, the use of high-density storage racks does increase the probability of a zircaloy cladding fire as compared with the use of low-density or open frame racks. The use of low-density storage racks, for the most recently off loaded fuel, as a minimum, is justified by a favorable value-impact ratio for new designs.

In SSAR Section 19B.2.63, GE listed several design features that have been incorporated into the ABWR to provide the degree of safety for the spent fuel pool required for resolution of the issue. These features are as follows:

- The spent fuel pool is located inside the reactor building, a seismic Category I structure, and is, therefore, protected against seismic loads, tornadic winds and the associated missiles, as well as turbine missiles. It is also protected from other type of missiles because of the absence of non-seismic systems, high- or moderate-energy piping, and rotating machinery in the vicinity of the spent fuel pool. Also, connections from the RHR system to the fuel pool cooling and cleanup (FPC) and suppression pool cleanup (SPCU) systems provide a seismic Category I, safety-related makeup capability to the spent fuel pool. In SSAR Section 9.1.3.3, GE states that following an accident or seismic event, the filter-demineralizers are isolated from the cooling portion of the fuel pool cooling and cleanup system and the SPCU system by two block valves in series at both the inlet and outlet of the common filter-demineralizer portion. Seismic Category I Quality Group C bypass lines are provided on both the fuel pool cooling and cleanup system and SPCU system to allow continued flow of cooling and makeup water to the spent fuel pool.
- Spent fuel is protected against heavy loads, including the fuel cask, by means of interlocks that prevent travel of the reactor building crane over the spent fuel storage pool.
- The SGTS limits the potential release of radioactive iodine and other radioactive materials because it is located inside the reactor building.
- No inlets, outlets, or drains are provided that might permit the pool to be drained below a safe shielding level.

In addition to the above features, SSAR Section 9.1.3 states the following:

- Fire protection is provided by means of standpipes in the reactor building and the water supplies of which are seismically designed. GE states that an analysis indicates that under the maximum abnormal heat load with the pool gates closed and no pool cooling taking place, the pool temperature will reach about 100 °C (212 °F) in about 16 hours. This provides sufficient time for the operator to hook up fire hoses for the pool makeup.

Based on the above information, the staff determined that the ABWR meets the guidelines or the pertinent regulatory documents listed above. Based on this information, the staff concludes that GE adequately addressed this issue for the ABWR design.

20.2.13 Issue 83: Control Room Habitability

This issue emphasizes the significant discrepancies found during a survey of existing plant control rooms. These discrepancies highlighted deficiencies in the maintenance and testing of ESFs designed to maintain control room habitability. They also provided examples of design and installation errors, including degradation of control room leak tightness, and pointed out a shortage of NRC and licensee personnel knowledgeable about HVAC systems and nuclear air-cleaning technology.

Loss of control room habitability following an accidental release of external airborne toxic or radioactive material or smoke can impair or cause loss of the control room operator's capability to safely control the reactor and could lead to a core damaging accident. Use of the remote shutdown station outside the control room following such events in unreliable since this station has no emergency habitability or radiation protection provisions similar to the control room's.

The review criterion for this issue is to verify that the control room is designed to provide adequate protection to the operating personnel during and following an accident. The design must meet the guidance given in the SRP Sections 6.4, 6.5.1, 9.4.1, and 15.6.5.5. The design must be in accordance with the requirements of GDC 2, 4, and 19 of 10 CFR Part 50, Appendix A, and ASME AG-1, "Code on Nuclear Air and Gas Treatment" (1991) and the ASME AG-1a-92 Addenda, ASME 509, and ASME N510.

SSAR Sections 9.4.1 and 6.4 describe control room habitability. The control room is designed to withstand the effects of natural phenomena, missiles, and postulated accidents in accordance with GDC 2 and 4. Design of the ambient conditions (HVAC system) permits safe occupancy during abnormal conditions. Radiation exposure of control room personnel during any of the postulated design-basis accidents does not exceed the requirements of GDC 19, that is, 0.05 Sv (5 rem) whole-body radiation exposure. Smoke and toxic gas protection will be provided by the use of noncombustible materials, purging by the HVAC, individual respirators, and site-specific considerations of potential chemical releases. SSAR Section 9.4.1.1.7 states that ESF filter trains comply with the design, testing, and maintenance provisions of RG 1.52. SSAR Appendices 9C and 9D provide information on how the ABWR design meets RG 1.52 and SRP Table 6.5.1-1, respectively. The staff's review of this section found that the ABWR conforms with the design, testing, and maintenance provisions of RG 1.52.

Based on this information, the staff finds that the ABWR is adequately designed to ensure the habitability of the control room under normal and accident conditions and meets the requirements of GDC 2 regarding the systems being capable of withstanding the effects of earthquakes; GDC 4 as it relates to maintaining environmental conditions in the control room compatible with the design limits of essential equipment located therein during normal, transient, and accident conditions; and GDC 19 with regard to providing adequate protection to permit access and occupancy of the control room under accident conditions. Therefore, the staff concludes that GE adequately addressed this issue for the ABWR design.

Three other issues are related to this issue: B-36, B-66, and III.D.3.4. They are discussed in Sections 20.1.25, 20.1.30, and 20.4.87, respectively, of this report.

20.2.14 Issue 86: Long-Range Plan for Dealing with Stress Corrosion Cracking in BWR Piping

This issue addresses leaks that were detected in the heat-affected zones of the safe-end-to-pipe welds in two of the 71-cm (28-in.) diameter recirculation loop safe ends at Nine Mile Point Unit 1. Subsequent ultrasonic examination revealed extensive cracking at many weld joints in the recirculation system. The cause of the cracking was determined to be IGSCC. Addressing existing power plants, this issue offers four possible solutions for preventing or mitigating the effects of IGSCC. These recommendations are contained in NUREG-0313, Revision 2, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping."

The ABWR design meets RG 1.44, "Control of the Use of Sensitized Stainless Steel," and NUREG-0313. This will ensure that significant sensitization in ABWR components will be avoided. Further, the ABWR design does not utilize recirculation piping and, therefore, the issue as it relates to cracking of the recirculation loop piping in BWRs does not apply to the ABWR. Based on this information, the staff concludes that GE adequately addressed this issue for the ABWR design.

20.2.15 Issue 87: Failure of High Pressure Coolant Injection Steam Line Without Isolation

This issue addresses a postulated break in the HPCI steam supply line and the uncertainty regarding the operability of the HPCI steam supply line isolation valves under those conditions. A similar situation can occur in the RWCU system. The HPCI steam supply line has two containment isolation valves in series (one inside the containment and one outside), both of which are normally open in most plants (two plants do operate with the outboard isolation valve normally closed). An HPCI supply valve located adjacent to the turbine and the turbine stop valve are normally closed. The RWCU system also has two normally open containment isolation valves that must remain open if the system is to function.

At the valve manufacturers' facilities, only the opening characteristics are tested under operating conditions (because of flow limitations). Although the operation of the valves is tested periodically without steam, the capability of the valves to close against the forces created by the steam flow resulting from a downstream line break has not been demonstrated. The valve type is not under

GE's (BWR vendor) scope of control, but is selected by the plant architect-engineer. This results in a diversity of valves and valve types (Y-type globe valves and gate valves) and increases the difficulty of demonstrating valve operating capability.

This issue is addressed in GL 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance," dated June 28, 1989, with Supplements 1-4. The staff's specific concern is the adequacy of testing, maintenance, and inspection of motor-operated valves (MOVs) so as to assure that they will remain functional when subjected to the design-basis conditions that are to be considered during both normal operation and abnormal events within the design basis of the plant.

SSAR Section 3.9.6.2.2 provides commitments for the design, qualification, testing, and inspection of MOVs. The details of the staff's evaluations are contained in Section 3.9.6.2.2 of this report. The staff's evaluation concludes that the commitments as described in SSAR Section 3.9.6.2.2 provide reasonable assurance for demonstrating the adequacy of the MOV isolation capability for the design-basis conditions and are, therefore, acceptable. Furthermore, SSAR Section 3.9.7.3 also states that the COL applicant will address the design qualification and testing for MOVs as discussed in SSAR Section 3.9.6.2.2 prior to plant startup.

Based on this information, the staff concludes that GE adequately addressed this issue for the ABWR design.

20.2.16 Issue 89: Stiff Pipe Clamps

Stiff pipe clamps were installed because of requirements for piping systems to withstand dynamic loads such as SRV discharges to suppression pools, LOCA-induced loads, and seismic loads. A preloading of pipe clamp U-bolts or straps, which imposes a constant compressive load on the piping, is necessary to prevent stiff pipe clamps from lifting under dynamic loading. In addition to the large preloading of the clamps, four other design features were identified as requiring additional analyses because of their differences from conventional pipe clamps. Those were (1) use of high-strength or non-ASME- approved materials; (2) local surface contact on the pipe; (3) uncommonly thick and/or wide design of clamp; and (4) clamp applications to piping components other than straight pipe, such as pipe elbows.

It was found that piping designers often assumed that the clamp effects on piping systems were negligible and did not warrant any explicit consideration. Although this assumption was acceptable for most clamp applications, in some cases, piping systems coupled with specific pipe clamp design requirements could experience interaction effects that need to be evaluated to determine the magnitude of pipe stresses induced.

The ASME Code, Section III, requires that the effects of attachments in producing thermal stresses, stress concentrations, and restraints on pressure-retaining members be taken into consideration in checking for compliance with stress criteria. The review criteria for this issue are that the effects of stiff clamps on piping stresses should be included in the piping system design. That is, in designing a piping system, cumulative stress contribution, such as due to thermal expansion of the pipe and clamp, discontinuity stresses in the pipe, stresses from thermal gradient through the pipe wall, and the external loads from dynamic events such as earthquake should also include the stresses attributed to the stiff clamps.

SSAR Section 19B.2.43 states that a study was performed in 1980 for typical stiff pipe clamps on BWR main steam and recirculation piping systems. For each system, the stiff clamps were installed on straight pipe or on bends with a radius of at least five pipe diameters. The purpose of these analyses was to evaluate the additional stresses at clamp locations resulting from the following:

- Differential thermal expansion of the pipe and clamp
- Discontinuity stresses in the pipe from internal pressure restraint
- Thermal gradient through the pipe wall in the vicinity of the pipe clamp
- External loads resulting from dynamic events such as earthquakes.

The results of these calculations showed that the total primary and secondary stresses, including clamp-induced stresses, were less than 70 percent of code-allowable stresses. GE states that the governing stress locations occurred at pipe branch connections, elbows, and shear lugs, and that they did not occur at stiff clamp locations. GE says this indicates that the stress intensification that occurs at elbows, branch connections, and shear lugs is greater than that occurring at stiff pipe clamps. Based on these calculations GE concluded, and the staff concurred, that explicit consideration of clamp-induced piping stresses is not required when the clamps are installed on straight pipe or on bends with a radius of at least five pipe diameters.

GE states in the SSAR section referred to above that the stiff clamp pipe analysis described above will be extended to the ABWR design and the pipe design specifications on other than NSSS piping will also consider these stress requirements.

Based on this information, the staff concludes that GE adequately addressed this issue for the ABWR design.

20.2.17 Issue 103: Design for Probable Maximum Precipitation

This issue concerns the use of the most recent National Oceanic and Atmospheric Administration (NOAA) procedures for determining probable maximum precipitation (PMP). More specifically, the issue was centered upon the NRC use of NOAA Hydrometeorological Reports (HMRs) Nos. 51 and 52, published in June 1978 and August 1982, respectively. The PMP values are used in estimating design flood levels at reactor sites. Objections were raised against use of these reports because by using HMR-52, higher values of flood levels were obtained than those when earlier reports were used. That would constitute unauthorized backfit under NRC procedures.

The procedures for estimating PMP and, therefore, the probable maximum flood (PMF) acceptable to the NRC are given in Appendices A and B of RG 1.59, "Design Basis Floods for Nuclear Power Plants," (Appendix A has since been superseded by ANSI N170-1976, "Standards for Determining Design Basis Flooding at Power Reactor Sites." ANSI N170-1976 defines the PMF as a hypothetical flood that is considered to be the most severe reasonably possible, based on comprehensive hydrometeorological application of PMP and other hydrologic factors favorable for maximum flood runoff. Thus, PMP is an integral component of PMF determination.

GDC 2 of 10 CFR Part 50, Appendix A, requires that the design bases for floods reflect consideration of the most severe historical flood data, with sufficient margin for the limited accuracy, quantity, and period of time in which the data have been accumulated. Additional guidance for estimating PMP and PMF is contained in SRP Sections 2.4.2, Revision 3, and 2.4.3 Revision 3, and in GL 89-22, "Potential for Increased Roof Loads and Plant Area Flood Runoff Depth at Licensed Nuclear Plants Due to Recent Change in Probable Maximum Precipitation Criteria Developed by the National Weather Service."

SSAR Section 19B.2.44 states that the ABWR design meets the requirements of GDC 2. In SSAR Table 2.0-1, GE provides the envelope of the ABWR site design parameters, among which is the maximum precipitation (for roof design) expressed in terms of a maximum rainfall rate of 49.3 cm/hr (19.4 in./hr), and a maximum snow load of 2.354 kPa (.341 psi). In Section 4.5.2.2 of the EPRI Evolutionary SER (NUREG-1242, "NRC Review of Electric Power Research Institute's Advanced Light Water Reactor Utility Requirements Document"), the staff stated that a 5-minute PMP is reasonable and that the 49.3 cm/hr (19.4 in./hr) PMP, together with 2.6 km^2 (1.0 mi^2), 5-minute PMP of 15.7 cm (6.2 in.) appears to be acceptable to the staff. However, that might exclude a number of sites in the Great Lakes area. GE further states that the ABWR meets the intent of SRP Sections 2.4.2, Revision 3 and 2.4.3, Revision 3 and GL 89-22. Detailed site characteristics based upon historical site environmental data will be provided by the site owner-operator for any specific applications. The staff verified that GE established a COL action item in SSAR Section 2.3 to provide site characteristics information. This approach is acceptable to the staff.

Based on this information, the staff determined that the ABWR is designed in accordance with the requirements of GDC 2 for the most severe environmental conditions, including floods, tornadoes, and hurricanes, that are expected and meets the intent of SRP Section 2.4.2 Revision 3, SRP Section 2.4.3 Revision 3, and GL 89-22. Therefore, the staff concludes that GE adequately addressed this issue for the ABWR design.

20.2.18 Reserved.

20.2.19 Issue 105: Interfacing Systems LOCA at LWRs

To protect against an intersystem LOCA (ISLOCA), designers of future ALWR plants should reduce the possibility of a LOCA outside containment by designing, to the extent practicable, all systems and subsystems connected to the RCS to an ultimate rupture strength (URS) at least equal to the normal RCS operating pressure.

Enhancements of isolation capability or the number of intersystem barriers (e.g., three isolation valves) are not considered to be adequate alternatives in systems that can be practically designed to the URS criteria. For example, piping runs should be designed to meet the URS criteria, as should all associated flanges, connectors, and packing, including valve stem seals, pump seals, heat exchanger tubes, valve bonnets, and RCS drain and vent lines. The designer should also make every effort to minimize the pressure loading experienced by each system and subsystem connected to the RCS should an ISLOCA occur. The staff does recognize, however, that all systems must eventually interface with atmospheric pressure and that it would be difficult or prohibitively expensive to design certain large tanks and heat exchangers to an URS equal to normal RCS operating pressure.

Applicants must provide justification demonstrating that it is not practicable to reduce the pressure challenge any further for each interfacing system and component that does not meet the RCS URS. This justification must be based upon an engineering feasibility analysis and not solely on the ratio of risk to benefit. Applicants must also demonstrate a compensating isolation capability for each interface for which it submits acceptable justification on the impracticability of normal RCS operating pressure capability. This would include a discussion of how the degree and quality of isolation or the reduced severity of the pressure challenges compensate for the low-pressure design of the interfacing system or component. The vendor may also need to consider the adequacy of pressure relief and piping of relief back to primary containment. In SECY-90-016, the staff stated that systems that have not been designed to full RCS pressure must include the following protection measures:

- the capability for leak testing of the pressure isolation valves,
- valve position indication that is available in the control room when isolation valve operators are deenergized, and
- high-pressure alarms to warn control room operators when rising RCS pressure approaches the design pressure of the attached low pressure systems and both isolation valves are not closed.

In the DFSER, the staff reported on its review of GE's preliminary evaluation of the RHR, HPCF, SLC, and RCIC systems and indicated that it would complete its review of Issue 105 upon receiving and reviewing additional information from GE. This was identified in the DFSER as Open Item 20.2-3.

In the original ABWR design, low-pressure piping that indirectly interfaces with the RCS (such as the RHR system suction piping) was designed to 1,277 kPa (200 psig). This design pressure was not considered to be acceptable since the piping was susceptible to pipe break because of ISLOCA. The staff required GE to upgrade the pressure rating of the piping that interfaces with the RCS. GE provided its implementation of the issue resolution in a submittal dated October 8, 1992, from J. Fox to C. Poslusny (NRC), "Proposed Resolution of ISLOCA Issue for ABWR." The staff reviewed the GE submittal and concluded that the design pressure for the low-pressure piping systems that interface with the RCS pressure boundary should be equal to 0.4 times the normal reactor operating pressure of 6,965 kPa (1025 psig), that is, 2,786 kPa (410 psig), and the minimum wall thickness of the low-pressure piping should be no less than that of a standard weight pipe. The staff concluded that this minimum pressure will ensure reasonable protection against burst failure should the low-pressure system be subjected to full pressure.

GE agreed with the staff position and made the necessary design changes as described below. All the low-pressure piping was changed to 2,786 kPa (410 psig). Furthermore, the staff will continue to require periodic surveillance and leak rate testing of the pressure isolation valves per TS requirements as a part of the ISI program. The details of the staff position on this issue is given in Section 3.9.3.1.1 of this report.

The following items form the basis of what constitutes practicality and set forth the test of practicality used to establish the boundary limits of URS for ABWR.

It is impractical to construct large tank structures to the URS design pressure that are vented to the atmosphere and have a low design pressure. Tanks included in this category are:

- Condensate storage tank
- SLCS main tank
- Low-conductivity waste receiving tank
- High-conductivity waste receiving tank
- FPC skimmer surge tank
- FPC spent fuel storage pool and cask pit
- Condensate hotwell.

Also included as impractical to upgrade were the suppression pool and primary containment. The suppression pool provides a low-pressure sink, approximately -96.31 kPa (0.75 psig) for its interfacing systems, and the first closed valve is rated to at least 2,786 kPa (410 psig). The ABWR containment is designed to 209 kPa (45 psig) and is designed to seismic Category I.

GE upgraded the design pressure of the following small tanks as a result of the review:

- SLCS test tank
- RCIC turbine barometric condenser tank.

Based on the staff guidance described above, GE evaluated in SSAR Appendix 3MA the following systems that interface with the RCS to verify that they are designed for an ISLOCA "to the extent practicable":

- RHR system
- HPCF system
- Standby liquid control system
- RCIC
- Control rod drive (CRD) system
- Reactor water cleanup system
- FPC system

- Nuclear boiler system (NBS)
- Reactor recirculation system (RRS)
- Makeup water (condensate) (MUWC) system
- Makeup water (purified) (MUWP) system
- Radwaste system
- Condensate and feedwater (CFS) system
- Sampling (SAM) system

The pressure of each system piping boundary was reviewed to identify where changes were needed to provide the URS protection. Where low-pressure piping interfaces with higher-pressure piping that is connected to the RCS, design pressure values were increased to 2,786 kPa (410 psig). The low-pressure piping boundaries were upgraded to URS pressures and extended to the last closed valve connected to the piping interfacing a low-pressure sink, such as the suppression pool or the condensate storage tank. For some systems, with low-pressure piping and normally open valves, the valves were changed to locked-open valves to ensure a pathway from the last URS boundary to the tank or low-pressure sink. Also, the minimum wall thickness for all the piping was changed to the standard weight pipe.

It is the staff's position that components such as heat exchangers, flanges, and pump seals should also be designed to a pressure of 2,786 kPa (410 psig). These changes have been implemented and are indicated in the SSAR by revised boundary symbols in the P&IDs. A stated parameter (e.g., design pressure) of a boundary symbol on the P&ID applies to all the piping and components on the P&ID that extend away from the boundary symbol, including any branch line, until another boundary symbol occurs on the P&ID.

GE upgraded the design pressure of piping in 14 systems that interface with the RCS and changed the design pressure of two tanks as the result of the ISLOCA review. Based on the above, the staff concludes that using the staff guidance, GE has modified the ABWR systems design "to the extent practicable" and, hence, has adequately addressed this issue for the ABWR design. Therefore, DFSER Open Item 20.2-3 is resolved.

Based on the above discussions, the staff concludes that the ABWR design meets the criteria of SECY-90-016 regarding ISLOCA prevention and mitigation.

20.2.20 Issue 106: Piping and Use of Highly Combustible Gases in Vital Areas

Issue 106 addresses the risk associated with the use of hydrogen and other combustible gases, such as propane and acetylene, during normal plant operation. This issue does not cover the use of large quantities of liquid hydrogen at hydrogen water chemistry (HWC) installations at BWRs or liquified petroleum gases (which are covered under Licensing Issue 136).

The review criteria for this issue are taken from EPRI Report NP-5283-SR-A, "Guidelines for Permanent BWR Hydrogen Chemistry Installations," and SRP Section 9.5.1. The current SRP Section 9.5.1, with the BTP CMEB 9.5-1, Part C.5.d(5), should be modified as follows: "Hydrogen lines in safety-related areas should follow the guidance of RG 1.29, "Seismic Design Classification," Section C.2. The lines should (1) be equipped with an excess flow valve or equivalent protection located outside the building so that in case of a line break, the hydrogen concentration in affected areas does not exceed 2 percent volume or (2) be sleeved with the outer pipe vented directly to the outside." The criteria should be applied to systems that supply hydrogen for cooling of the electric generators.

The system design should comply with the following general guidance:

- Design features and administrative controls should be provided to prevent inadvertent bypass of small or normally isolated hydrogen supplies.

- Flow limiting devices should be used to limit hydrogen releases to a leak or pipe break.

- Equipment and controls to mitigate the consequences of a hydrogen fire or explosion should be accessible and remain functional during an event.

- Design features and administrative controls should be provided to isolate the hydrogen supply if normal building ventilation is lost.

- Backflow to a leak or line break of hydrogen contained in components (e.g., generator) should be considered in evaluating the consequences of leaks or breaks and measures taken to mitigate these consequences.

- Threaded joints in the hydrogen distribution lines within safety-related areas should be back welded.

EPRI Report NP-5283-SR-A provides guidelines for HWC installations. With the exception of information dealing specifically with the HWC application (e.g., certain trips, injection points, and main steam line radiation), most of the EPRI guidelines dealing with hydrogen are applicable to these other uses. The guidelines give a number of system design features and administrative controls that are in addition to, or more restrictive than, those in SRP Section 9.5.1. In addition, safety-related equipment should not be located in the turbine building because of the hazards associated with hydrogen fires or explosions and large oil fires and the large uncertainties in estimating the consequences.

According to SSAR Section 9.3.9.1.2, hydrogen is used in the ABWR design to reduce the dissolved oxygen in the reactor water in order to mitigate the potential for IGSCC of sensitized austenitic stainless steels. The amount of hydrogen required is in the range of 1.0 to 1.5 ppm, but the exact amount, which depends on many factors, including for instance, incore recirculation rates, will be determined by tests performed during initial operation of the plant.

SSAR Section 19B.2.48 states that the ABWR design uses hydrogen for HWC and the main generator bulk hydrogen supply system. These systems are non-nuclear and non-safety-related and are located in the turbine building, which is a non-safety-related structure in a nonvital area.

SSAR Section 10.2.2.1 states that there are no safety-related systems or components located within the turbine building, hence any local failure associated with the turbine-generator (T-G) unit will not affect any safety-related equipment. It further states that failure of T-G equipment cannot preclude safe shutdown of the RCS. SSAR Sections 9.3.9.1.2 and 10.2.2.2 state that the HWC system and the bulk hydrogen system, respectively, utilize the guidelines given in EPRI Report NP-5283-SR-A with respect to those portions of the guidelines involving hydrogen which do not deal specifically with the HWC system. Specifically, the bulk hydrogen system and HWC system piping and components will be located to reduce risk from their failures. The bulk hydrogen storage is located outside but near the turbine building. The arrangement of buildings at the facility and location of building doors and the bulk hydrogen storage tanks will be designed to ensure that damage to buildings containing safety-related equipment due to combustion of hydrogen or an explosion is unlikely.

Hydrogen lines are provided with a pressure-reducing station before the piping enters the turbine building which limits the maximum flow of hydrogen to less than 100 standard cubic meters per minute (3530 scfm). The hydrogen piping inside the turbine building will be designed in accordance with the guidance of RG 1.29, Position C.2, regarding the seismic design of non-safety-related equipment whose failure could affect safety-related equipment. This is in order to comply with the modified BTP CMEB 9.5-1, Part C.5.d(5). Additionally, all threaded joints in the hydrogen distribution piping will be back welded. Equipment and controls used to mitigate the consequences of a hydrogen fire or explosion will be designed to be accessible and remain functional during the postulated post-accident condition. Design features and/or administrative controls will be provided to ensure that the hydrogen supply is isolated when normal building ventilation is lost.

Based on this information, the staff concludes that GE adequately addressed this issue for the ABWR design.

20.2.21 Issue 113: Dynamic Qualification Testing of Large Bore Hydraulic Snubbers

Large bore hydraulic snubbers (LBHSs) have a load rating greater than 22.68 tonnes (50 kips). They are active mechanical devices used to restrain safety-related piping and equipment during seismic or other dynamic events (e.g., high-energy line breaks), yet also allow sufficient piping and component flexibility to accommodate system expansion and contraction resulting from thermal transients, such as normal plant heatups and cool downs. Dynamic qualification testing and periodic functional testing are important to verify that the LBHSs are properly designed and maintained for the life of the plant. Issue 113 addresses the need for requirements for dynamic testing of LBHSs.

SSAR Section 19B.2.64 states that for the ABWR design, LBHSs will only be used for piping systems when dynamic supports are required at locations where large thermal displacements prohibit the use of rigid supports. They will not be used in applications other than piping restraints. Section 19B.2.64 refers the reader to SSAR Section 3.9.3.4.1(3) for information on design, testing, installation, and pre-service examination of mechanical and hydraulic snubbers, including LBHSs. To assure snubber functionality under various normal and abnormal conditions, snubbers are to be designed in accordance with provisions of the ASME Code, Section III, Subsection NF. This design requirement includes analysis for normal, upset, emergency, and faulted loads. These calculated loads are then compared with the allowable loads to verify that the stresses are below the code-allowable limits.

GE described the tests that the snubbers will be subjected to, and they will consist of the following:

- Force or displacement versus time loading at frequencies within the range of significant modes of the piping system.
- Dynamic cyclic tests to determine the operational characteristics of the snubber control valve.

- Displacement tests to determine the specified performance characteristics.

The snubbers will be tested for various abnormal environmental conditions. Upon completion of the abnormal environmental transient test, the snubbers will be tested dynamically at a frequency within a specified frequency range.

Based on this information, the staff concludes that GE provided acceptable commitments to dynamically test LBHSs, and therefore, adequately addressed this issue for the ABWR design.

20.2.22 Issue 118: Tendon Anchorage Failure

An inspection of a PWR prestressed concrete containment structure showed that three lower vertical tendon anchor heads were broken. The failures appeared to be caused by stress corrosion cracking. Quantities of water ranging from a few milliliters (oz.) to about 5.7 liters (1.5 gals.) were found in the grease caps.

A reinforced concrete containment for BWRs will be used. Since a prestressed concrete containment is not specified, no specific requirements for tendon anchorage are provided.

SSAR Sections 3.8.1 and 19B.2.48 state that the primary containment of the ABWR standard plant is designed as a reinforced concrete structure. Since the technique of prestressed concrete design has not been used by GE, the issue of tendon anchorage failure is not applicable for the containment structure design of the ABWR standard plant. Therefore, the staff concludes that GE adequately addressed this issue for the ABWR design.

20.2.23 Issue 120: On-Line Testability of Protection Systems

During its 1985 review of several plant TS, the staff discovered that the design of protection systems of some plants did not provide as complete a degree of on-line, at-power surveillance testing capability as did other plants undergoing staff review and evaluation at that time. This raised questions about the on-line testability of protection systems and the possibility that some nuclear power plants might not provide complete testing capability. Issue 120 was established to examine these questions. Protection systems consist of the reactor protection system (RPS) and the engineered safety features actuation system (ESFAS). The main concern of this issue, however, is the on-line testability of the actuation subgroup (slave) relays in the ESFAS.

The requirements for at power testability of components are included in GDC 21 of 10 CFR Part 50, Appendix A. RGs 1.22, "Periodic Testing of Protection System Actuation Functions," and 1.118, "Periodic Testing of Electric Power and Protection Systems," and IEEE 338-1977, "Criteria for the Periodic Testing of Nuclear Power Generating Station Safety Systems," provide supplemental guidance. This guidance is intended to ensure that protection systems (including logic, actuation devices, and associated actuated equipment) will be designed to permit testing while a plant is operating at power without adversely affecting the plant's operation. The scope of testing covered consists of functional tests, checks, calibration verifications, and time- response measurements. Criteria are provided for determining system operational availability, status, and necessary documentation, and for establishing test intervals and test procedures during operation.

The review criteria applied for this issue included an assessment of the capability for periodic functional testing of the systems. This periodic testing should be manually initiated, but automatically performed once initiated, and should meet the guidance of RGs 1.22 and 1.118 and IEEE 338. Automatic initiation of periodic testing may be provided where the testing does not degrade the system functionality. Built-in, automated test features are expected to be provided for periodic, functional testing, as necessary, to eliminate physical reconfiguration of systems (e.g., adding jumpers, lifting leads, swapping cables) to perform the required tests. The safety-related systems are to have automatic test features that are sufficient to meet TS requirements for periodic surveillance of the system's functionality as defined by RGs 1.22 and 1.118 and IEEE 338.

This issue was resolved during the staff's USI-GSI review for the ABWR without the establishment of new requirements. Nevertheless, the staff evaluated the capability of the ABWR for continuous on-line self-testing of hardware and system integrity as detailed above.

SSAR Sections 7.1.2.1.6 and 19B.2.49 indicate that the ABWR design's RPS and ESFAS can be tested at power during reactor operation by six separate tests. The first five tests are primarily manual tests and although each individually is a partial test, combined with the sixth test they constitute a complete system test. The sixth test is the self-test of the system logic and control that automatically tests the complete system excluding sensors and actuators.

The sixth test is an integrated self-test provision built into the microprocessors within the safety system logic and control (SSLC). It consists of on-line, continuously operating, self-diagnostic, monitoring network and an off-line semi-automatic (operator initiated, but automatic to completion), end-to-end surveillance program. This testing includes the following:

- On-line continuous testing, which is a self-diagnostic program monitoring each signal-processing module from input to output (actuation of the trip functions is not performed during this test)
- Off-line semi-automatic end-to-end testing, which exercises the trip outputs of the SSLC logic processors.

All testing features adhere to the single-failure criterion.

Since the design of the systems permit periodic testing of their functioning when the reactor is in operation, including a capability to test channels independently to determine failures and losses of redundancy that may have occurred, the staff determined that the ABWR meets the requirements of GDC 21 of 10 CFR Part 50, Appendix A. Therefore, the staff concludes that GE adequately addressed this issue for the ABWR design.

20.2.24 Issue 121: Hydrogen Control for Large, Dry PWR Containments

As a result of the TMI-2 accident, the Commission promulgated regulatory requirements on hydrogen control in 10 CFR 50.34(f) and 50.44. 10 CFR 50.34(f) requires a hydrogen control system based on 100-percent, fuel-clad metal-water reaction and hydrogen concentration limit of 10 percent on uniformly distributed hydrogen in the containment or on a post-accident atmosphere that will not support hydrogen combustion.

The review criterion for this issue is that the control of hydrogen generated in the containment in a degraded core accident shall meet the requirements of 10 CFR 50.34(f) on limiting the distributed hydrogen concentration to 10 percent, on limiting the combustible concentration, and on maintaining safe-shutdown equipment and containment integrity.

This issue is relevant to large, dry containments, such as those used in PWRs. Therefore, this issue does not apply to the ABWR design. Additionally, the ABWR primary containment is inerted and is, therefore, protected from hydrogen combustion regardless of the amount or rate of hydrogen generation.

Based on this information, the staff concludes that GE adequately addressed this issue for the ABWR design. See also the staff's evaluation of 10 CFR 50.34(f)(2)(ix) in Section 20.5.21 of this report.

20.2.25 Issue 124: Auxiliary Feedwater System Reliability

In 1985, it was observed by the NRC staff and industry that the auxiliary feedwater systems (AFWSs) continued to fail at a high rate. These studies also indicated that plants with similar AFWS reliabilities (as calculated in accordance with the SRP guidance) did not necessarily exhibit similar AFWS availabilities. Based on these studies and on engineering judgement, the staff concluded that the PWR AFW system reliabilities calculated in accordance with the SRP guidance may have represented the relative reliability of AFW system hardware configurations for various plants, but did not represent the real availability of the system.

A function of the AFWS in most cases is to supply water to the secondary side of the steam generator during system fill, normal plant heatup, normal plant standby, and normal plant cold shutdown. The AFWS also functions following loss of normal feedwater flow, including loss because of offsite power supply failure, and provides emergency feedwater (EFW) following such postulated events as main feedwater line break or main steam line break.

The AFWS reliability criterion has been specified in SRP Section 10.4.9. For the ABWR, SSAR Section 19B.2.51 states that the acceptance criteria for resolution of this issue will be that the AFWS will be designed for a high degree of reliability (that is, using reliability analyses the system will attain 0.0001 to 0.00001 unavailability per demand).

As it has been pointed out above, the function of the AFWS in most cases is to supply water to the secondary side of the steam generator under various conditions. Since this condition does not exist in case of the ABWR, this issue is not applicable to the ABWR.

20.2.26 Issue 128: Electrical Power Reliability

Concerns have been raised regarding the dependence on Class 1E power, especially dc power, of the decay heat removal systems required for long-term heat removal. Failure of one division would generally result in a reactor scram, which would then require removal of decay heat. The frequency of reported failures of single dc divisions gives rise to the concern that the second dc division may not be available. This issue combines three interrelated issues, namely, A-30, "Adequacy of Safety-Related DC Supplies;" 48, "LCOs for Class 1E Vital Instrument Buses in Operating Reactors;" and 49, "Interlocks and LCOs for Redundant Class 1E Tie Breakers."

Issue A-30 involves the power source to the inverters for the vital 120-Vac instrument power supplies that are related to Issues 48 and 49. It is also related to Issue 76, "Instrumentation and Control Power Interactions," because a loss of 120-Vac vital instrument power could challenge emergency safeguards systems and could cause reactor trips, loss of feedwater, loss of emergency core and containment cooling systems, and loss of post-accident monitoring instrumentation.

Issue A-48 concerns the fact that in some plants, there are no administrative controls governing operational restrictions for their Class 1E 120-Vac vital instrument buses (VIBs) and associated inverters. Without such restrictions, these power sources could be out of service indefinitely and might place certain safety systems in a situation where they could not meet the single-failure criterion. This is of particular concern during the period before the start and load of diesel generators following a loss of offsite power. In this condition, some VIBs may be subjected to power failure modes that may not have been considered during the safety analysis of the plant.

Issue A-49 arises from the fact that in some plants there is lack of adequate procedural and administrative controls that are used to monitor and provide assurance that the tie breakers between redundant Class 1E divisions of electrical power and multi-units are always open during plant operation. Such controls are necessary to provide assurance that the Class 1E power buses are not compromised. There is also a related concern, when a tie breaker is closed, involving electrical interlocks to prevent out-of-synchronization interconnections of a diesel generator to either the off-site power source or another diesel generator.

The review criteria for this issue are contained in NUREG/CR-5414, "Technical Findings for Proposed Integrated Resolution of Generic Issue 128, Electric Power Reliability." Generally, this document states that the plant design should provide for the separation and isolation of electrical power systems to preclude interactions that could adversely affect such functions as diesel generator loading and offsite to diesel generator power transfers.

Each division of the ESFs requiring electric power will be provided with an onsite source of ac and dc power. At least two separate and independent connections will be provided to offsite power sources capable of starting and running all Class 1E loads required for safe shutdown. The specified functions of the ESFs will be met by the use of redundant divisions and that the divisions will be totally independent and separated both mechanically and electrically. The DHR systems will be redundant and safety grade.

The plant designs will have three independent divisions for the core coolant inventory control and DHR systems and each division will have its own independent ac and dc power source.

Separation of electrical power systems will be such as to preclude interactions that could adversely affect the functioning of the dc power systems. Specifically, the use of bus ties between safety divisions is prohibited.

Non-safety-related loads will be placed on power supplies that are completely separate from those on which safety-related loads are placed.

The loss of any plant battery or dc bus concurrent with a single independent failure in any other system required for shutdown cooling will not result in a total loss of reactor cooling capability.

Each reactor protection channel will be normally powered from a dedicated Class 1E source that is normally independent of other dc sources.

SSAR Section 19B.2.52 describes the ABWR design features pertinent to the resolution of Issues 128, A-30, 48, and 49.

For Issue A-30, the dc buses for the safety-related dc power system meet the acceptance criteria because of the following:

- The safety-related dc power system does not supply power to any non-Class 1E loads.
- Consists of four separate and independent dc battery systems.
- Does not contain any direct bus tie between dc battery systems. However, it does contain two standby battery charges, each of which is capable of supplying one of the divisional dc systems. Redundant key-locked breakers are provided to prevent manual paralleling between divisions, and no automatic connections are provided between dc divisions.

The ABWR design meets the acceptance criteria for the resolution of Issue 48 by the system design and TS. The ABWR design consists of four separate and independent Class 1E 120-Vac vital instrument buses with their respective inverters. TS contain the appropriate operating restrictions, to assure the onsite Class 1E ac and dc power distribution system availability and thus, an uninterruptable power source for safety-related systems and components. The TS contain specific requirements regarding a periodic evaluation of the onsite power system bus condition which addresses such availability items as correct breaker and bus alignment and bus voltage.

The ABWR meets the acceptance criteria with regards to Issue 49. The ABWR Class 1E system design does not contain bus tie breakers between Class 1E divisions, but it is possible to manually cross-connect the Class 1E diesel buses through the CTG connections, since the power to each diesel bus can be provided from the CTG. Each diesel generator is provided with a synchronizing equipment for paralleling offsite supplies. The normal and alternate offsite feeder breakers to Class 1E buses are interlocked to prevent paralleling offsite circuits.

Based on this information, the staff concludes that GE adequately addressed this issue for the ABWR design.

20.2.27 Issue 130: Essential Service Water Pump Failures at Multiplant Sites

This issue addresses shared essential service water (ESW) systems between PWR units located on the same site. Specifically, this issue deals with multiplant sites that have only two ESW pumps per plant with cross-tie capability and the impact of such sharing on the availability of the ESW pumps. The safety concern is that the needed ESW pumps may not be available during all possible operating conditions for the multiple units, thereby increasing the core melt and radiological risks at the site. Other multiplant and single plant configurations may also contain similar ESW system vulnerabilities. Therefore, the concern of this issue is equally applicable to other multiplant and single plant PWR sites. According to Appendix B to NUREG-0933, Issue 130 is not applicable to BWR vendors.

Issue 130 was resolved with the issuance on September 19, 1991, of GL 91-13, "Request for Information Related to the Resolution of Generic Issue 130, 'Essential Service Water System Failures at Multi-Unit Sites,' Pursuant to 10 CFR 50.54(f)." This letter contained TS and emergency procedures improvements for seven specific multiplant PWR sites.

Although this issue is not applicable to BWR plants, the staff included an evaluation of the concerns as related to the ABWR RSW design in the DFSER. In the DFSER, the staff said that the SSAR stated that the ABWR has been designed as a single unit with no specific consideration of possible shared systems. (Note that in all system analysis reviews, the requirements of GDC 5 have been identified as not being applicable to the ABWR design.) Shared systems are, therefore, not a concern for the ABWR design.

The staff reported in the DFSER that SSAR Appendix 19B, "Resolution of Applicable Unresolved Safety Issues and Generic Safety Issues," Sections 19B.2.10 and 19B.3.5, allowed for the possible modification of the ABWR design by plant-specific applicants. The guidelines provided by GE to assure that the modified design would address the concerns of Issue 130 included reference to two design requirements from the EPRI ALWR Requirements Document. The first requirement was for limiting the number of shared systems to the "test programs which have been made" and further states that these systems "shall incorporate the pertinent results into the design of the ALWR." The second requirement can be interpreted to address shared systems at multiplant sites stated that each division of any ESW system must have two heat exchangers and two pumps sized so that each division can provide the capacity to absorb the system heat loads generated by the plant during all operational (normal and emergency) modes.

The staff stated in the DFSER that the requirements mentioned above did not provide sufficient guidance to an applicant referencing the ABWR design for a multiplant site to design ESW systems for the units that reflect an acceptable resolution of the concern of Issue 130. Specifically, the staff indicated that the first EPRI requirement as identified by GE is confusing. The subject requirement as stated in the Requirements Document (see Volume 2, Chapter 1, Section 6.2.B.1) limits the number of shared systems to auxiliary support systems such as sewer, auxiliary steam, or site security. Further, another EPRI requirement (see Requirements Document, Volume 2, Chapter 8, Item B.1.4.2) clarifies the above EPRI requirement by specifically ruling out sharing of ESW pumps between divisions and between units of a multiplant site and by requiring the ESW system for each unit to be designed to the same requirements as for a single unit. The EPRI requirement as identified by GE, on the other hand, implies that the ESW system can be shared between the units, provided certain conditions are satisfied. However, the staff related that the ABWR SSAR provided little or no guidance regarding whether the ESW system pumps will be shared between the units and if so, how they will be shared, the number of shared as well as non-shared ESW pumps for each unit, the capacity of each ESW pump, and the operational limitations required to minimize system misoperation (human errors associated with the wrong train or the wrong plant identification). In addition, the staff said that GE should address the applicable reliability concerns of Issue 130 for single-plant units.

In the DFSER, the staff required GE to address all the above concerns and modify the two identified interface requirements accordingly to provide sufficient guidance to the applicants referencing the ABWR design. The staff stated that such guidance is needed so that each plant-specific applicant will be able to provide supporting documentation and analysis to justify modifications to the ABWR essential cooling water systems if a multi-unit site is to incorporate shared cooling water systems in the modified designs. The staff indicated that the analysis and documentation should demonstrate that the revised plant design meets the concerns identified in Issue 130, as well as the requirements of GDC 5 with regard to the sharing of structures, systems, and components at multi-unit sites. This was identified in the DFSER as Open Item 20.2-4.

Further, the staff required in the DFSER that the COL applicant address the plant-specific resolution of Issue 130. This was identified as COL Action Item 20.2-3. The staff stated that it would review the plant-specific response to the concerns of Issue 130 and that the review would include, among other things, the available redundancy in ESW pumps within a division to achieve any licensing-basis requirement, available flexibility for needed recovery actions, and specific measures to preclude potential operator errors.

GE revised the SSAR to delete the discussion of Issue 130, because it is not applicable to BWR plants. The SSAR discussions and EPRI requirements that are the subject of DFSER Open Item 20.2-4 no longer exist in SSAR Sections 19B.2.10 and 19B.3.5. SSAR Section 1.1.6 still states that the ABWR has been designed as a single-unit plant, therefore, the staff need not be concerned with specific consideration of possible shared systems and with addressing Issue 130. Because Issue 130 is not applicable to BWR plants and based on the subsequent revisions to the SSAR discussed above, the staff concludes that DFSER Open Item 20.2-4 is resolved.

Because this issue is not applicable to the ABWR and based on the above discussion of the ABWR design as a single-unit plant without any shared systems, the staff concludes that the COL applicant need not prepare a plant-specific response to the issue. On this basis, the staff concludes that DFSER COL Action Item 20.2-3 was not warranted and need not be addressed.

20.2.28 Issue 142: Leakage Through Electrical Isolators in Instrumentation Circuits

This issue addresses electrical isolators used to maintain electrical separation between safety-related and non-safety-related electrical systems in nuclear power plants, preventing malfunctions in the non-safety-related systems from degrading the performance of safety-related circuits. The primary concern is that the amount of energy that could pass through certain types of isolation devices (and be transmitted to safety-related circuitry) during certain electrical transients might damage or seriously degrade the performance of Class 1E components. Or, this energy could cause the isolation devices to give false output, or the electrically generated noise on the circuit might cause the isolation device to give a false output.

The review criteria for this issue are contained in a letter from T. Murley (NRC) to R. Artigas (GE), "Advanced Boiling Water Reactor Licensing Review Bases," dated August 7, 1987. This letter contains the staff's expectations regarding isolation devices. The letter describes the design and environmental qualifications that insulators must satisfy. It requires that description of tests be provided, as well as other requirements, as appropriate. The letter also addresses guidance for fiber-optic cable, and states that the staff is working to develop comprehensive guidance on this subject, and that it will be based on the existing IEEE cable standards, such as IEEE 323 and 384, and applicable ANSI standards.

Therefore, the review criteria for this issue must contain guidance for:

- Inspection and testing of all electrical insulation devices between Class 1E and non-Class 1E systems.

- Replacement or repair of isolators that fail the tests, including description of acceptable hardware fixes to the isolators.

- Implementation of an annual program to inspect and test all electrical isolators between Class 1E and non-Class 1E systems.

SSAR Appendix 7A and Section 19B.2.53 state that the isolating devices used in the ABWR design are similar to those in Group 1 referred to in NUREG/CR-3453, "Electronic Isolators Used in Safety Systems of U.S. Nuclear Power Plants." Review of this reference confirmed that test results demonstrated that only minor amounts of high-frequency energy pass through the barrier during testing. These isolators, though they provide the best isolation and are recommended as the safest units, they are the most susceptible to damage. The report indicates that they are so fragile that some were damaged during the electromagnetic interference (EMI) tests.

GE indicates that the ABWR design will use a fiber-optic system for electrical isolation of logic and analog signals between protection divisions and from protection divisions to non-safety equipment. This selection of isolation devices appears to be acceptable, in view of the statement in the referenced report that "optical-fiber isolators are the newest of the isolators and consist of both analog and digital types."

Another positive feature of the design is the fact that the ABWR RPS and ESF functions are not supplied directly from a commercial power source, but from different plant power sources. Thus, the low voltage dc supplies fed from these sources are regulated and filtered. This is important, because as the reference states, when power is supplied from commercial sources to the input power supply through the common ac power line, as it is in most PWRs, power surge appears to be high.

The advance SER stated that as part of the resolution of this issue, there should be a requirement stated in the acceptance criteria in SSAR Section 19B.2.53 that the following will be implemented:

- Annual inspection and testing of all electrical isolation devices between Class 1E and non-Class 1E systems.
- Repair or replacement of insulators that fail the tests.

These changes were incorporated in SSAR Amendment 34.

In the advance SER, the staff concluded that GE provided an acceptable system of isolators to resolve this issue, assuming that the testing, inspection, and replacement of isolators, when needed, would be incorporated in resolution of this issue. The staff verified that GE established a COL action item in SSAR Section 19B.3.2 and Table 1.9-1 addressing testing, inspection, and replacement of isolators. On the basis of this information, the staff concludes that GE adequately addressed this issue for the ABWR design.

20.2.29 Issue 143: Availability of Chilled Water Systems

This issue concerns partial or complete loss of the HVAC system. Many of these problems exist because of the desire to provide increased fire protection and the need to avoid severe temperature changes in equipment control circuits. The improvements in this area, which started after the Browns Ferry fire, consist mainly of enclosing the affected equipment in small, isolated rooms. This resulted in a significant reduction in room cooling rate. Plant control and safety have improved with the introduction of electronic integrated circuits. However, these circuits are more susceptible to damage from severe changes in temperature caused by the loss of room cooling.

It is believed that failures of air cooling systems for areas housing key components, such as RHR pumps, switchgear, and diesel generators, could contribute significantly to core-melt probability in certain plants. Because corrective measures are often taken at the affected plants once such failures occur, the impact of these failures on the proper functioning of air cooling systems may not have been considered. Thus, plants with similar inherent deficiencies may not be aware of these problems.

Operability of some safety-related components is dependent upon operation of HVAC and chilled water systems to remove heat from the rooms containing the components. If chilled water and HVAC systems are unavailable to remove heat, the functionality of the equipment within the rooms may be destroyed.

The review criterion for this issue is to assure that the equipment can be functional during the period of loss of room cooling.

SSAR Section 19B.2.54 states the following criteria as the acceptable ABWR design:

- An evaluation of the dependencies or nondependencies of safety-related equipment on HVAC cooling will be performed. This evaluation will include assessment of room heat load and heatup rates, and establish equipment operating conditions. Equipment ability to withstand these conditions without loss of function will be established.
- For equipment found to be significantly dependent on the HVAC cooling, an assessment of the HVAC system reliability will be performed. PRA analyses will be carried out to assess plant risk and determine whether any modifications are necessary.
- Corrective design measures will be identified where necessary to reduce plant risk.

SSAR Section 19B.2.54 states that the following features have been incorporated into the ABWR design to satisfy the criteria stated above:

- RCIC pump and turbine are designed to operate for at least 8 hours without room cooling. This system will provide core cooling during a prolonged loss of HVAC cooling.
- Operation of other injection systems (HPCF, LPFL, RHR) is more dependent on the availability of room cooling. However, these systems are designed to operate for at least 10 minutes without room cooling. The equipment in question is designed to be operational at the highest temperature expected during that time.
- Detailed design specifications for ABWR safety-related equipment will specify the room conditions under which equipment must operate without room cooling.

Room heat assessment will be performed to establish environmental conditions for equipment specification.

- Potential modifications including procedure changes or hardware changes will be evaluated through PRA analyses to ensure acceptable risk.

GE also listed the safety-related HVAC systems that will provide room cooling under most circumstances. These include the secondary containment safety-related HVAC, reactor building safety-related electrical equipment HVAC, reactor building safety-related diesel generator HVAC, and HVAC emergency cooling water (three divisions).

The staff concludes that GE adequately addressed this issue for the ABWR design.

20.2.30 Issue 145: Actions to Reduce Common Cause Failures

This issue concerns the fact that common cause failures (CCFs) can be a major cause of a system failure.

Testing of equipment has its limitations; in fact, testing can be an important cause of CCF that occur when the testing does not reflect true demands of the equipment under operating conditions. For example, MOVs may work during a test but not during a true demand when there exists a high differential pressure across them. Much design-basis testing cannot be performed in situ. Prototypical testing, on the other hand, is expensive and the application of it to equipment in plants is sometimes not practical. Effective maintenance is important to ensure that design assumptions and margins in the original design bases are maintained. In the design of nuclear plants, an important safety margin is the redundancy of equipment to perform safety functions. This redundancy, however, can be degraded by CCFs. Thus, measures are needed to identify CCF precursors before they occur so that corrective measures can be taken.

The review criterion for the resolution of Issue 145 is to demonstrate compliance with the maintenance rule, 10 CFR 50.65, which requires that a program of performance and condition monitoring activities be evaluated at least every refueling cycle (provided the interval between evaluations does not exceed 24 months) and that industry-wide experience be incorporated in the program. When monitoring and preventive maintenance activities are performed, an assessment of the total plant equipment that is out of service should be taken into account to determine the overall effect on performance of safety functions.

Implementation of 10 CFR 50.65 is an effective and practical way to prevent or reduce CCFs.

SSAR Section 19B.2.55 states that compliance with 10 CFR 50.65 will be responsibility of the COL applicant. This approach is acceptable to the staff.

In addition, the ABWR capability to respond to system interactions and CCFs is described in the SSAR Section 19.2.3.4. Five factors are considered and incorporated in the analysis of system interactions and CCFs:

- Component commonality at the system level, such as common initiating signal.
- Common divisional services, such as common electric power buses or common service water loops.
- System dependency, such as automatic depressurization system (ADS) dependency on the operability of at least one of the five (two high-pressure and three low-pressure) ECCS pumps.
- Past experience of losing onsite power.
- Human errors.

Actions to reduce CCFs fall into the Owner/Operator's Reliability Assurance Process (O-RAP), described in SSAR Section 17.3.10 and which has been reviewed by the staff and found to be acceptable as discussed in Sections 17.3.9 and 17.3.10 of this report. The COL applicant will specify the policy and implementation procedures for the O-RAP and submit it for staff review. The staff verified that GE established a COL action item in SSAR Section 17.9.13 to make use of information provided by GE to help the owner/operator determine activities that should be included in the O-RAP. This approach is acceptable to the staff.

Based on this information, the staff concludes that GE adequately addressed this issue for the ABWR design. The staff will evaluate compliance with 10 CFR 50.65 as part of its inspection activities during plant operations.

20.2.31 Issue 151: Reliability of Recirculation Pump Trip During an ATWS

This issue concerns reliability of breakers used to trip the recirculation pumps at high pressure or low-water-level signals during ATWS mitigation in BWRs.

If a plant transient requiring a reactor scram occurs and the scram function does not occur, then an ATWS event exists. To lessen the effects of an ATWS event, negative reactivity must be added to the reactor core by tripping the recirculation pumps. Negative reactivity is added as a result of the ensuing steam voiding in the core area as the core flow decreases, thereby decreasing the power generation and limiting the power or pressure disturbance. If the recirculation pump trip (RPT) fails to trip on demand, the reactor could experience continued power generation resulting in a high suppression pool temperature.

Plants equipped with GE AKF-25 circuit breakers have experienced failures of the field breakers in the RPT system that were caused by binding of the trip latch mechanism and misadjustment of the breakers' mechanical linkage.

The review criterion for this issue is determining the use of reactor RPT hardware or a RPT method that is more reliable than the previously used AKF-2-25 breaker hardware or RPT method.

Since the design for the ABWR reactor recirculation system and RPT method is completely different from the previously designed BWR reactor recirculation systems and RPT trip methods, resolution of this issue was evaluated on the basis of the ABWR RPT information about the new design for the recirculation pump system provided by GE in SSAR Sections 7.7.1.3(7) and 7.7.1.3(8). A summary of the important design changes is provided in the following paragraphs.

GE states that the ABWR reactor recirculation system and the RPT design is completely different from the previously designed BWR reactor recirculation systems and RPT methods. They state that it is more diverse and more redundantly reliable. The ABWR uses 10 pumps and multiple pump and RPT trip logic, circuits and hardware, rather than only two recirculation pumps, as has been generally used in BWRs. The recirculation flow control (RFC) system consists of three redundant process controllers, adjustable-speed drives (ASDs), switches, sensors, and alarm devices provided for operational manipulation of the 10 RIPs and the surveillance of the associated equipment. RFC is achieved either by manual operation or by automatic operation if the power level is above 70 percent of rated power.

In the event of (a) a turbine trip or generator load rejection when reactor power is above a predetermined level, (b) the reactor pressure exceeds the high dome pressure set point, or (c) the reactor water level drops below the Level 3 set point, the RPT logic will automatically trip off a group of four RIPs. If the reactor water level continues to drop and reaches Level 2 after the first group of four RIPs have been tripped, the remaining six RIPs will be tripped. The implementation of the second RPT function is similar to that of the first RPTs, using 2-out-of-4 confirmation logic.

It is known that plants with GE AKF-25 circuit breakers have experienced failures of the field breakers in the RPT system. In the ABWR design, instead of using AKF-25 breaker switching hardware to provide an RPT, RFC controller switching and ADS gate inverter turn-off circuit hardware provide the RPT.

The staff determined that the system described above appears to be more reliable than the previously used AKF-2-25 breaker hardware and method and, therefore, provides reasonable assurance that in the event of an ATWS, the RIPs will be tripped, thus lessening the effect of the ATWS. Based on this determination, the staff concludes that GE adequately addressed this issue for the ABWR design.

20.2.32 Issue 153: Loss of Essential Service Water in LWRs

This issue concerns reliability of the ESW supply that is critical in the transfer of heat from various safety-related and non-safety-related systems and equipment to the UHS. The ESW is needed in every phase of plant operations and, under accident conditions, supplies adequate cooling water to systems and components that are important to safe plant shutdown or to mitigate consequences of an accident. Under normal operating conditions, the ESW provides component and room cooling (mainly via the component cooling water system). During shutdowns, it also ensures that the residual heat is removed from the reactor core, cooling towers, and water treatment systems at a plant. A complete loss of the ESW system could lead to a core-melt accident, posing a significant risk to the public.

Loss of ESW can be caused by a number of reasons: various fouling mechanisms (sediment decomposition, biofouling, corrosion and erosion, foreign materials, and debris intrusion), ice effects, single failures and other design deficiencies, flooding, multiple equipment failures, and personnel and procedural errors. Additionally, the design and operational characteristics of the ESW system differ significantly from plant to plant within each reactor type. For these reasons, it is practically impossible to formulate generic ESW design criteria that would be universally applicable. The design bases of the system will conform to the requirements of GDC 2 relating to the ability of structures, systems, and mechanisms to withstand the effects of earthquakes; GDC 4 regarding the protection of safety-related equipment from the effects of internally generated missiles, pipe whip, and environmental conditions resulting from high and moderate energy line breaks and the dynamic effects associated with flow instabilities and loads (e.g., water hammer); GDC 5 relating to shared systems and components; and GDC 44 as is relates to transferring heat from structures, systems, and components important to safety, to an ultimate heat sink. More specific review criteria are contained in SRP Section 9.2.1.

The ABWR RSW system removes heat from the reactor building cooling water (RCW) and transfers it to the UHS. The RSW system is provided in three divisions. Each division has two pumps that send cooling water to three heat exchangers. Normally, one pump and two heat exchangers are operating in each division. The remaining pump and the third heat exchanger are automatically put in operation when heat-removal requirements increase.

In case of failure of any of the three RSW divisions, the remaining two divisions are sufficient to meet safe shut down requirements for the plant. The ABWR RSW system is protected from common-cause effects by the fact that the three divisions are separated both physically and electrically from each other.

Degradation of the RSW system is prevented by periodic inspection and testing to ensure integrity and functional capability. All three divisions of the RSW system are designed to allow periodic inservice inspection of all the system components. This testing capability consists of structural and leak-tightness visual inspection, entire system operability, and system component operability and performance.

SSAR Section 19B.2.57 lists the design features for the portions of system that are not within the ABWR standard plant scope that will be provided by the COL applicant.

Based on the above, the staff concludes that GE adequately addressed this issue for the ABWR design.

20.2.33 Issue 155.1: More Realistic Source Term Assumptions

This issue is a result of the study conducted by the TMI-2 Safety Advisory Board, and is one of the seven recommendations forwarded to the NRC. The subject issue deals with the fact that during the TMI-2 accident, fission products did not behave as predicted by the analytical methods and assumptions used in the licensing process at the time and delineated in RG 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Boiling Water Reactors," RG 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors," and TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites." Contrary to the original predictions that during the TMI-2 accident major core damage had occurred, NRC determined that approximately 50 percent of the core was in a molten state, and only about 55 percent of the highly volatile fission products and noble gases were released from the reactor vessel with a major portion retained in the reactor building. There is also evidence that less than 5 percent of the medium- and low-volatile fission products were released from the reactor vessel.

The review of this issue consists of determining that the plant is designed to ensure that the dose commitment to the public, in the event of a licensing design-basis accident, will be within those limits prescribed by existing regulations based upon the guidelines of 10 CFR Part 100.

SSAR Section 19B.2.58 states that in view of lack of adequate guidance as to acceptance methods and conditions, i.e., revised RGs and SRP, it is premature for the ABWR to use the revised source terms. It also states that the ABWR design has been analyzed in accordance with the current RGs, SRP, and the GDC, all of which are based on TID-14844. The staff evaluated offsite radiological consequences using the TID-14844 source term procedures that are consistent with the guidelines provided in the applicable SRP sections and RGs. (See Section 15.4.3 of this report.) Two deviations from the current staff position were noted: (1) credit was given for radioactive iodine removal in the main steam lines and in the main condensers by holdup for decay and deposition and (2) the staff accepted the ABWR design without an MSIV leakage control system.

The advance SER indicated that in response to the staff's concern, the BWR Owners' Group (BWROG) performed further studies to determine the consequences, if any, of the ABWR design regarding radioactive offsite releases. The staff subsequently evaluated the results of the BWROG and GE proposals and found them to be acceptable. In particular, in SSAR Chapter 15, GE performed radiological consequence assessments of certain DBAs and concluded that the ABWR design, using TID-14844 source terms with the two deviations stated above, will meet the dose reference values established in 10 CFR Part 100 and the dose limits given in GDC 19 of 10 CFR 50, Appendix A. To verify GE's conclusion, the staff independently assessed the radiological consequences resulting from DBAs, also using TID-14844 and the deviations discussed earlier, and found GE's assessments to be acceptable. The staff assessments and conclusion are discussed further in Section 15.4 of this report.

Based on this information, the staff concludes that GE adequately addressed this issue for the ABWR design.

20.3 Human Factors Issues

This section addresses staff evaluation of GSIs categorized as "human factors issues" in NUREG-0933. All the following issues are relevant to the ABWR design.

20.3.1 Issue HF 1.1: Staffing and Qualifications - Shift Staffing

The purpose of this issue is to assure that the number and capabilities of the staff at nuclear power plants are adequate to provide safe operation. To meet this goal, consideration should be given to: (1) the number and functions of the staff needed to safely perform all required plant operations, maintenance, and technical support for each operational mode; (2) the minimum qualifications of plant personnel in terms of education, skill, knowledge, training experience, and fitness for duty; and (3) appropriate limits and conditions for shift work including overtime, shift duration, and shift rotation. More specifically, this issue refers to determination of the minimum appropriate shift crew staffing composition.

The review criteria for this issue are contained in the 10 CFR 50.54, SRP Section 13.1.2-13.1.3, and RG 1.114, "Guidance on Being Operator at the Controls of a Nuclear Power Plant."

This issue is beyond the scope of the ABWR design certification review and the COL applicant will be responsible for addressing it. The staff verified that GE established a general COL action item (Item 18.1.1 in SSAR Section 18.8) to conduct the detailed HFE design according to design and implementation. The staff considers this to include the resolution of Issue HF 1.1. This approach is acceptable to the staff as discussed in Section 18.7.2.2 of this report.

20.3.2 Issue HF 4.4: Procedures - Guidelines for Upgrading Other Procedures

The objective of this issue is to provide assurance that plant procedures are adequate and can be used effectively and to guide operators in maintaining plants in a safe state under all operating conditions. This latter includes the ability to control upset conditions without first having to diagnose the specific initiating event. This objective is to be met by: (1) developing guidelines for preparing and criteria for evaluating emergency operating procedures (EOPs), normal operating procedures, and other procedures that affect plant safety and (2) upgrading the procedures, training the operators in their use, and implementing the upgraded procedures.

The review criteria for this issue are contained in SRP Sections 13.5.1 and 13.5.2 and NRC Information Notice 86-64, "Deficiencies in Upgrade Programs for Plant Emergency Operating Procedures."

The development of detailed procedures and training materials is beyond the scope of the ABWR design certification review and the COL applicant will be responsible for addressing this generic issue. The staff verified that GE established a COL action item in SSAR Section 13.5.3 for procedure development. SSAR Section 13.5.3.1 indicates that the methods and criteria for the development, verification and validation (V&V), implementation, maintenance, and revision of procedures will include considerations of Issues I.C.1, I.C.5, and I.C.9. The staff considers this to also include the resolution of Issue HF 4.4. This approach is acceptable to the staff as discussed in Section 13.5 of this report.

20.3.3 Issue HF 5.1: Man-Machine Interface - Local Control Stations

The objective of this issue is to ensure that the man-machine interface is adequate for the safe operation and maintenance of nuclear power plants. The regulatory guidance has been limited to the control room and the remote shutdown panel. Further guidance is necessary regarding local control stations and auxiliary operator interfaces. To accomplish this task, analyses of control room crew and local control activities should be conducted to establish and describe communication and control links between the control room and the auxiliary control stations.

The review criteria for this issue are contained in SRP Section 18.2, Appendix A.

This issue is beyond scope of the ABWR design certification review and the COL applicant will be responsible for addressing it. The staff verified that GE established a COL action item (Item 18.8.11 in SSAR Section 18.8) to analyze this issue. This approach is acceptable to the staff as discussed in Section 18.7.2.2 of this report.

20.3.4 Issue HF 5.2: Man-Machine Interface - Review Criteria for Human Factors Aspects of Advanced Controls and Instrumentation

With the outcome of advanced technologies utilizing improved annunciator systems, guidelines for evaluation of these longer-term annunciator improvements are necessary. These guidelines will be based upon evaluations of results from advanced concept activities being performed by the

Government and commercially sponsored research programs.

The existing HFE guidelines for nuclear power plant control rooms primarily address the control, display, and information concepts and technologies that are now being used in process control systems. While these guidelines were adequate in the past, they may not be sufficient for advanced and developing technologies that are being introduced in more advanced designs.

The review criteria for this issue are contained in SRP Section 18.2, Appendix A.

In SSAR Section 18.3, GE described an integrated design implementation process to incorporate HFE principles. The process includes an integrated design of control and instrumentation systems and HSI of the ABWR. The design implementation process facilitates selection of design features that satisfy the top level requirements and goals of individual systems and the overall plant.

In its review of GE's detailed control room design process, the staff found that this issue is beyond the scope of the ABWR design certification review and that it is to be addressed by the COL applicant as part of the detailed design implementation. The staff verified that GE established a general COL action item in SSAR Section 18.8 (Item 18.8.1) to conduct the detailed HFE design according to the design and implementation as defined by the HFE ITAAC and DAC described in ABWR CDM Table 3.1 and SSAR Appendix 18E. The staff considers this to include the analysis of this generic issue. This approach is acceptable to the staff as discussed in Section 18.7.2.2 of this report.

20.4 TMI Action Plan Items

This section addresses staff evaluation of GSIs that are categorized as "TMI Action Plan items" in NUREG-0933. Except as noted, all the following issues are relevant to the ABWR design.

20.4.1 Issue I.A.1.1: Operating Personnel -- Operating Personnel and Staffing - Shift Technical Advisor

This issue requires the provision of an on-shift technical advisor, with specific education and training, to the shift supervisor.

SSAR Section 19B.3.1 states that the COL applicant is responsible for providing resolutions of issues identified as "COL Applicant" in the "Safety Issues Index" of SSAR Appendix 19B. The SSAR states that issues are identified for COL action because they pertain to operating personnel issues, operating procedures, and other topics beyond the scope of the ABWR design certification review. SSAR Section 19B.1.1 lists specific documentation the COL applicant is to provide for resolution of such issues.

GE identified this issue for COL applicant action in the "Safety Issues Index." The staff verified that GE established a COL action item (Item 19-28) in SSAR Table 1.9-1 to address unresolved generic and TMI safety issues. This approach is acceptable to the staff. The staff will review the COL applicant's proposed resolution of this issue on a case-by-case basis.

20.4.2 Issue I.A.1.2: Operating Personnel -- Operating Personnel and Staffing - Shift Supervisor Administrative Duties

This issue requires review of the administrative duties of the shift supervisor by the senior officer responsible for plant operations. It also requires that when administrative functions detract from or are subordinate to the management responsibility for assuring the safe operation of the plant, they are to be delegated to other operations personnel not on duty in the control room.

SSAR Section 19B.3.1 indicates that the COL applicant is responsible for providing resolutions of issues identified as "COL Applicant" in the "Safety Issues Index" of SSAR Appendix 19B. The SSAR indicates that issues are identified for COL action because they pertain to operating personnel issues, operating procedures, and other topics beyond the scope of the ABWR design certification review. SSAR Section 19B.1.1 lists specific documentation the COL applicant is to provide for resolution of such issues.

GE identified this issue for COL applicant action in the "Safety Issues Index." The staff verified that GE established a COL action item (Item 19-28) in SSAR Table 1.9-1 to address unresolved generic and TMI safety issues. This approach is acceptable to the staff. The staff will review the COL applicant's proposed resolution of this issue on a case-by-case basis.

20.4.3 Issue I.A.1.3: Operating Personnel -- Operating Personnel and Staffing - Shift Manning

This issue requires adherence to the shift manning and overtime requirements for normal plant operation established by the NRC.

SSAR Section 19B.3.1 indicates that the COL applicant is responsible for providing resolutions of issues identified as "COL Applicant" in the "Safety Issues Index" of SSAR

Appendix 19B. The SSAR indicates that issues are identified for COL action because they pertain to operating personnel issues, operating procedures, and other topics beyond the scope of the ABWR design certification review. SSAR Section 19B.1.1 lists specific documentation the COL applicant is to provide for resolution of such issues.

GE identified this issue for COL applicant action in the "Safety Issues Index." The staff verified that GE established a COL action item (Item 19-28) in SSAR Table 1.9-1 to address unresolved generic and TMI safety issues. This approach is acceptable to the staff. The staff will review the COL applicant's proposed resolution of this issue on a case-by-case basis.

20.4.4 Issue I.A.1.4: Operating Personnel -- Operating Personnel and Staffing - Long-Term Upgrading

The purpose of this issue was to develop changes to 10 CFR 50.54 concerning shift staffing with licensed operators and their working hours.

The resolution of this issue is beyond the scope of the ABWR design certification review and the COL applicant will be responsible for addressing it. This was identified in the DFSER as COL Action Item 18.7.2.2-8. The staff verified that GE established a general COL action item (Item 18.1.1 in SSAR Section 18.8) to conduct the detailed HFE design during design and design implementation. The staff considers this to include the resolution of Issue I.A.1.4. This approach adequately addresses DFSER COL Action Item 18.7.2.2-8 as discussed in Section 18.7.2.2 of this report.

20.4.5 Issue I.A.2.1: Operating Personnel -- Training and Qualification of Operating Personnel - Immediate Upgrading of Operator and Senior Operator Training and Qualifications

This issue requires that, effective December 1, 1980, all senior reactor operator (SRO) applicants must have been a licensed operator for at least 1 year.

SSAR Section 19B.3.1 states that the COL applicant is responsible for providing resolutions of issues identified as "COL Applicant" in the "Safety Issues Index" of SSAR Appendix 19B. The SSAR indicates that issues are identified for COL action because they pertain to operating personnel issues, operating procedures, and other topics beyond the scope of the ABWR design certification review. SSAR Section 19B.1.1 lists specific documentation the COL applicant is to provide for resolution of such issues.

GE identified this issue for COL applicant action in the "Safety Issues Index." The staff verified that GE established a COL action item (Item 19-28) in SSAR Table 1.9-1 to address unresolved generic and TMI safety issues. This approach is acceptable to the staff. The staff will review the COL applicant's proposed resolution of this issue on a case-by-case basis.

20.4.6 Issue I.A.2.3: Operating Personnel -- Training and Qualifications of Operating Personnel - Administration of Training Programs

This issue requires that, subject to the accreditation of training institutions, licensees and applicants assure that training center and facility instructors who teach systems, integrated responses, transients, and simulator courses demonstrate SRO qualifications and be enrolled in appropriate requalification programs.

SSAR Section 19B.3.1 states that the COL applicant is responsible for providing resolutions of issues identified as "COL Applicant" in the "Safety Issues Index" of SSAR Appendix 19B. The SSAR indicates that issues are identified for COL action because they pertain to operating personnel issues, operating procedures, and other topics beyond the scope of the ABWR design certification review. SSAR Section 19B.1.1 lists specific documentation the COL applicant is to provide for resolution of such issues.

GE identified this issue for COL applicant action in the "Safety Issues Index." The staff verified that GE established a COL action item (Item 19-28) in SSAR Table 1.9-1 to address unresolved generic and TMI safety issues. This approach is acceptable to the staff. The staff will review the COL applicant's proposed resolution of this issue on a case-by-case basis.

20.4.7 Issue I.A.2.6(1): Operating Personnel -- Training and Qualifications of Operating Personnel - Long-Term Upgrading of Training and Qualifications; Revise Regulatory Guide 1.8

This issue required NRC development of a revised RG 1.8 to incorporate recommendations on upgrading personnel training and qualifications.

SSAR Section 19B.3.1 states that the COL applicant is responsible for providing resolutions of issues identified as "COL Applicant" in the "Safety Issues Index" of SSAR Appendix 19B. The SSAR indicates that issues are identified for COL action because they pertain to operating personnel issues, operating procedures, and other topics beyond the scope of the ABWR design certification review.

SSAR Section 19B.1.1 lists specific documentation the COL applicant is to provide for resolution of such issues.

GE identified this issue for COL applicant action in the "Safety Issues Index." The staff verified that GE established a COL action item (Item 19-28) in SSAR Table 1.9-1 to address unresolved generic and TMI safety issues. This approach is acceptable to the staff. The staff will review the COL applicant's proposed resolution of this issue on a case-by-case basis, evaluating compliance with the version of RG 1.8 current at the time of the review.

20.4.8 Issue I.A.3.1: Operating Personnel -- Licensing and Requalification of Operating Personnel - Revise Scope of Criteria for Licensing Examinations

This issue requires the inclusion of simulator examinations as part of the licensing examinations.

SSAR Section 19B.3.1 states that the COL applicant is responsible for providing resolutions of issues identified as "COL Applicant" in the "Safety Issues Index" of SSAR Appendix 19B. The SSAR states that issues are identified for COL action because they pertain to operating personnel issues, operating procedures, and other topics beyond the scope of the ABWR design certification review. SSAR Section 19B.1.1 lists specific documentation the COL applicant is to provide for resolution of such issues.

GE identified this issue for COL applicant action in the "Safety Issues Index." The staff verified that GE established a COL action item (Item 19-28) in SSAR Table 1.9-1 to address unresolved generic and TMI safety issues. This approach is acceptable to the staff. The staff will review the COL applicant's proposed resolution of this issue on a case-by-case basis.

20.4.9 Issue I.A.4.1(2): Operating Personnel -- Simulator Use and Development - Initial Simulator Improvement; Interim Changes in Training Simulators

This issue requires the following capabilities for simulators: modeling saturation conditions; providing multiple-failure accident training, including incorrect instrument responses; providing training for both active and passive failure of ESF components; providing training on natural circulation operation under solid water conditions; and other simulator weaknesses that may have been identified under I.A.2.6 and I.A.4.2.

SSAR Section 19B.3.1 states that the COL applicant is responsible for providing resolutions of issues identified as "COL Applicant" in the "Safety Issues Index" of SSAR Appendix 19B. The SSAR states that issues are identified for COL action because they pertain to operating personnel issues, operating procedures, and other topics beyond the scope of the ABWR design certification review. SSAR Section 19B.1.1 lists specific documentation the COL applicant is to provide for resolution of such issues.

GE identified this issue for COL applicant action in the "Safety Issues Index." The staff verified that GE established a COL action item (Item 19-28) in SSAR Table 1.9-1 to address unresolved generic and TMI safety issues. This approach is acceptable to the staff. The staff will review the COL applicant's proposed resolution of this issue on a case-by-case basis.

20.4.10 Issue I.A.4.2: Operating Personnel -- Simulator Use and Development - Long-Term Training Upgrade

Refer to the evaluation of 10 CFR 50.34(f)(2)(i) in Section 20.5.13 of this report.

20.4.11 Issue I.C.1: Operating Procedures - Short-Term Accident Analysis and Procedures Revision

The objective of this issue was to improve the analysis of design-basis and off-normal transients and accidents and the procedures for handling them. Actions to address this issue include the performance of analyses of small-break LOCAs, inadequate core cooling, transients, and accidents; preparation of emergency procedure guidelines (EPGs); implementation of appropriate emergency procedures; and training of operators.

The development of detailed procedures and training materials is beyond the scope of the ABWR design certification review and the COL applicant will be responsible for addressing this TMI item. The staff verified that GE established a COL action item in SSAR Section 13.5.3 for procedure development. SSAR Section 13.5.3.1 states that the methods and criteria for the development, V&V, implementation, maintenance, and revision of procedures will include considerations of I.C.1. This approach is acceptable to the staff as discussed in Section 13.5 of this report.

20.4.12 Issue I.C.2: Operating Procedures - Shift Relief and Turnover Procedures

This issue requires that plant procedures include provisions to assure that shift and relief turnover is adequately prescribed to ensure that each oncoming shift is aware of critical plant status information and system availability prior to assuming duties.

SSAR Section 19B.3.1 states that the COL applicant is responsible for providing resolutions of issues identified as "COL Applicant" in the "Safety Issues Index" of SSAR Appendix 19B. The SSAR states that issues are identified for COL action because they pertain to operating personnel issues, operating procedures, and other topics beyond the scope of the ABWR design certification review. SSAR Section 19B.1.1 lists specific documentation the COL applicant is to provide for resolution of such issues.

GE identified this issue for COL applicant action in the "Safety Issues Index." The staff verified that GE established a COL action item (Item 19-28) in SSAR Table 1.9-1 to address unresolved generic and TMI safety issues. This approach is acceptable to the staff. The staff will review the COL applicant's proposed resolution of this issue on a case-by-case basis.

20.4.13 Issue I.C.3: Operating Procedures - Shift Supervisor Responsibilities

This issue requires review and revision of plant procedures and directives to assure that duties, responsibilities, and authority are properly defined to establish a definite line of command and clear delineation of the command decision authority of the supervisor in the control room relative to other plant management personnel. It also requires training programs for shift supervisors to emphasize and reinforce the responsibility for safe operation and the management function of the shift supervisor to assure safe operation of the plant.

SSAR Section 19B.3.1 states that the COL applicant is responsible for providing resolutions of issues identified as "COL Applicant" in the "Safety Issues Index" of SSAR Appendix 19B. The SSAR states that issues are identified for COL action because they pertain to operating personnel issues, operating procedures, and other topics which are beyond the scope of the ABWR design certification review. SSAR Section 19B.1.1 lists specific documentation the COL applicant is to provide for resolution of such issues.

GE identified this issue for COL applicant action in the "Safety Issues Index." The staff verified that GE established a COL action item (Item 19-28) in SSAR Table 1.9-1 to address unresolved generic and TMI safety issues. This approach is acceptable to the staff. The staff will review the COL applicant's proposed resolution of this issue on a case-by-case basis.

20.4.14 Issue I.C.4: Operating Procedures - Control Room Access

This issue requires that the authority and responsibilities of the person in charge of control room access and clear lines of authority and responsibility in the control room in the event of an emergency be established in conformance with item 2.2.2.a of NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations," dated July 1979.

SSAR Section 19B.3.1 states that the COL applicant is responsible for providing resolutions of issues identified as "COL Applicant" in the "Safety Issues Index" of SSAR Appendix 19B. The SSAR states that issues are identified for COL action because they pertain to operating personnel issues, operating procedures, and other topics beyond the scope of the ABWR design certification review. SSAR Section 19B.1.1 lists specific documentation the COL applicant is to provide for resolution of such issues.

GE identified this issue for COL applicant action in the "Safety Issues Index." The staff verified that GE established a COL action item (Item 19-28) in SSAR Table 1.9-1 to address unresolved generic and TMI safety issues. This approach is acceptable to the staff. The staff will review the COL applicant's proposed resolution of this issue on a case-by-case basis.

20.4.15 Issue I.C.5: Operating Procedures - Procedures for Feedback of Operating Experience to Plant Staff

Refer to the evaluation of 10 CFR 50.34(f)(3)(i) in Section 20.5.41 of this report.

20.4.16 Issue I.C.6: Operating Procedures - Procedures for Verification of Correct Performance of Operating Activities

This issue requires review and revision, as necessary, of procedures to assure that an effective system of verifying the correct performance of operating activities is provided as a means of reducing human errors and improving the quality of normal operations. Such a verification system may include automatic system status monitoring and human verification of operations and maintenance activities independent of the people performing the activity.

SSAR Section 19B.3.1 states that the COL applicant is responsible for providing resolutions of issues identified as "COL Applicant" in the "Safety Issues Index" of SSAR Appendix 19B. The SSAR states that issues are identified for COL action because they pertain to operating personnel issues, operating procedures, and other topics beyond the scope of the ABWR design certification review. SSAR Section 19B.1.1 lists specific documentation the COL applicant is to provide for resolution of such issues.

GE identified this issue for COL applicant action in the "Safety Issues Index." The staff verified that GE established a COL action item (Item 19-28) in SSAR Table 1.9-1 to address unresolved generic and TMI safety issues. This approach is acceptable to the staff. The staff will review the COL applicant's proposed resolution of this issue on a case-by-case basis.

20.4.17 Issue I.C.7: Operating Procedures - NSSS Vendor Review of Procedures

This issue requires that applicants for near-term operating licenses (NTOLs) obtain the NSSS vendor's review of the low-power and power-ascension test and emergency procedures to further verify the adequacy of the procedures.

SSAR Section 19B.3.1 states that the COL applicant is responsible for providing resolutions of issues identified as "COL Applicant" in the "Safety Issues Index" of SSAR Appendix 19B. The SSAR states that issues are identified for COL action because they pertain to operating personnel issues, operating procedures, and other topics beyond the scope of the ABWR design certification review. SSAR Section 19B.1.1 lists specific documentation the COL applicant is to provide for resolution of such issues.

GE identified this issue for COL applicant action in the "Safety Issues Index." The staff verified that GE established a COL action item (Item 19-28) in SSAR Table 1.9-1 to address unresolved generic and TMI safety issues. This approach is acceptable to the staff. The staff will review the COL applicant's proposed resolution of this issue on a case-by-case basis.

20.4.18 Issue I.C.8: Operating Procedures - Pilot Monitoring of Selected Emergency Procedures for NTOL Applicants

This issue requires an interdisciplinary and interoffice NRC task force review of emergency procedures received from NTOL applicants and of the training related to the symptoms of the postulated transients.

SSAR Section 19B.3.1 states that the COL applicant is responsible for providing resolutions of issues identified as "COL Applicant" in the "Safety Issues Index" of SSAR Appendix 19B. The SSAR indicates that issues are identified for COL action because they pertain to operating personnel issues, operating procedures, and other topics beyond the scope of the ABWR design certification review. SSAR Section 19B.1.1 lists specific documentation the COL applicant is to provide for resolution of such issues.

GE identified this issue for COL applicant action in the "Safety Issues Index." The staff verified that GE established a COL action item (Item 19-28) in SSAR Table 1.9-1 to address unresolved generic and TMI safety issues. This approach is acceptable to the staff. However, the COL applicant's responsibility for resolution of this issue extends only to providing the necessary procedures to the task force, accommodating the task force review, and addressing the review findings to the satisfaction of the NRC.

20.4.19 Issue I.C.9: Operating Procedures - Long-Term Program Plan Procedures for Upgrading of Procedures

Refer to the evaluation of 10 CFR 50.34(f)(2)(ii) in Section 20.5.14 of this report.

20.4.20 Issue I.D.1: Control Room Design - Control Room Design Reviews

Refer to the evaluation of 10 CFR 50.34(f)(2)(iii) in Section 20.5.15 of this report.

20.4.21 Issue I.D.2: Control Room Design - Plant Safety Parameter Display Console

Refer to the evaluation of 10 CFR 50.34(f)(2)(iv) in Section 20.5.16 of this report.

20.4.22 Issue I.D.3: Control Room Design - Safety System Status Monitoring

Refer to the evaluation of 10 CFR 50.34(f)(2)(v) in Section 20.5.17 of this report.

20.4.23 Issue I.D.5(2): Control Room Design - Plant Status and Post-Accident Monitoring

The objective of this issue is to improve the ability of nuclear power plant control room operators to prevent, diagnose, and properly respond to accidents and concentrates on the operator's information needs.

The review criteria for this issue are contained in RG 1.97 (Rev. 3). This document provides guidance for the design of instrumentation to help the operators (1) to determine the nature of an accident and whether the reactor trip and engineered safety features are functioning properly, (2) to provide information regarding the potential for breaching the barriers to radioactivity release, and (3) furnish data for deciding on the need to take manual action if an engineered safety feature malfunctions.

SSAR Section 19B.2.65 states that the ABWR information system provides information for manual initiation and control of safety systems. These systems provide information sufficient for the operators to take an appropriate action when needed. Section 19B.2.65 refers to SSAR Section 7.5, "Information Systems Important to Safety," which describes safety-related display systems that provide information for the safe operation of the plant during normal operation, anticipated operational occurrences, and accidents. SSAR Section 7.5.2.1, "Post Accident Monitoring System," describes Type A Variables, which are plant-specific parameters needed to alert the control room operators to take actions manually, initiating a system or function that otherwise would not be automatically initiated in the course of an event. In conformance with the guidelines provided in RG 1.97, SSAR Table 7.5-2 lists post-accident monitoring variables which are common to BWR designs. The staff concludes that the features discussed above adequately address this issue for the ABWR design.

20.4.24 Issue I.D.5(3): Control Room Design - On-Line Reactor Surveillance System

This issue addresses noise surveillance and diagnostic techniques associated with the on-line reactor surveillance system. More specifically, it focuses on neutron noise monitoring in BWRs to detect the impact of instrument tubes against fuel channel boxes or detect other loose internal reactor parts.

The review criteria for this issue are addressed in RG 1.133, "Loose Parts Detection Program for the Primary System of Light-Water Cooled Reactor." SSAR Section 4.4.4, "Loose-Parts Monitoring System," describes the ABWR design features to provide detection of loose metallic parts within the RPV. The loose-parts monitoring system (LPMS) is designed to provide detection and operator warning of loose parts in the RPV to avoid or mitigate damage to or malfunction of reactor components. Additional design considerations provide for the inclusion of electronic features to minimize operator interfacing requirements during normal LPMS operation. These electronic features improve the LPMS capability when operator action is required. GE provided a general description of the LPMS, including the design bases, system description, system operation, safety evaluation, test, inspection, and application. The LPMS includes sensors (accelerometers located at neutral loose parts collection regions, e.g., steam outlet nozzle, feedwater inlet nozzle, control rod drive housings), signal conditioning, signal analysis, alarms, and calibration. The staff reviewed the LPMS description and concluded in Section 4.4.4.2 of this report that it conforms with RG 1.133 and, therefore, concludes that GE adequately addressed this issue for the ABWR design.

20.4.25 Issue I.F.1: Quality Assurance (QA) - Expand QA List

Refer to the evaluation of 10 CFR 50.34(f)(3)(ii) in Section 20.5.42 of this report.

20.4.26 Issue I.F.2: Quality Assurance - Develop More Detailed QA Criteria

Refer to the evaluation of 10 CFR 50.34(f)(3)(iii) in Section 20.5.43 of this report.

20.4.27 Issue I.G.1: Preoperational and Low-Power Testing - Training Requirements

The objective of this issue is to increase the capability of shift crews to operate facilities in a safe and competent manner by assuring that training for plant changes and off-normal events is conducted.

The review criterion for this issue is the definition of training plans prior to fuel loading and the conduct of training prior to full-power operation for each operating shift. The resolution of this issue is beyond the scope of the ABWR design certification review, and the COL applicant will be responsible for addressing it. SSAR Section 13.2 discusses training requirements for reactor operators. The staff verified that GE established a COL action item in SSAR Section 13.2.3.2 to include training requirements for preoperational and low-power testing activities. This approach is acceptable to the staff as discussed in Section 13.2 of this report.

20.4.28 Issue I.G.2: Preoperational and Low-Power Testing - Scope of Test Program

The objective of this issue is to review the comprehensiveness of test programs to identify anomalies in a plant's response to transients.

The review criteria for resolution of this issue are specified in SRP Chapter 14, "Initial Test Program - Final Safety Analysis Report," and RG 1.68, "Initial Test Programs for Water-Cooled Reactor Power Plants." SRP Section 14.2 sets forth the acceptable test procedures to establish the degree of conformance with the applicable tests identified in RG 1.68. RG 1.68, in turn, describes the general scope and depth of initial test programs acceptable to the NRC staff for light-water cooled reactors. RG 1.68, Appendix A provides a representative listing of plant structures, systems, and components design features and performance capability tests that should be demonstrated during the initial test program.

The staff reviewed the test program proposed by GE for the ABWR and concludes in Section 14.2 of this report that it conforms to SRP Chapter 14 and RG 1.68. Therefore, the staff concludes that GE adequately addressed this issue for the ABWR.

20.4.29 Issue II.B.1: Consideration of Degraded or Melted Cores in Safety Review - Reactor Coolant System Vents

Refer to the evaluation of 10 CFR 50.34(f)(2)(vi) in Section 20.5.18 of this report.

20.4.30 Issue II.B.2: Consideration of Degraded or Melted Cores in Safety Review - Plant Shielding to Provide Access to Vital Areas and Protect Safety Equipment for Post-Accident Operation

Refer to the evaluation of 10 CFR 50.34(f)(2)(vii) in Section 20.5.19 of this report.

20.4.31 Issue II.B.3: Consideration of Degraded or Melted Cores in Safety Review - Post-Accident Sampling

Refer to the evaluation of 10 CFR 50.34(f)(2)(viii) in Section 20.5.20 of this report.

20.4.32 Issue II.B.4: Consideration of Degraded or Melted Cores in Safety Review - Training for Mitigating Core Damage

This issue requires development and implementation of a training program to teach the use of installed equipment and systems to control or mitigate accidents in which the core is severely damaged.

SSAR Section 19B.3.1 indicates that the COL applicant is responsible for providing resolutions of issues identified as "COL Applicant" in the "Safety Issues Index" of SSAR Appendix 19B. The SSAR states that issues are identified for COL action because they pertain to operating personnel issues, operating procedures, and other topics which are beyond the scope of the ABWR design certification review. SSAR Section 19B.1.1 lists specific documentation the COL applicant is to provide for resolution of such issues.

GE identified this issue for COL applicant action in the "Safety Issues Index." The staff verified that GE established a COL action item (Item 19-28) in SSAR Table 1.9-1 to address unresolved generic and TMI safety issues. This approach is acceptable to the staff. The staff will review the COL applicant's proposed resolution of this issue on a case-by-case basis.

20.4.33 Issue II.B.8: Consideration of Degraded or Melted Cores in Safety Review - Rulemaking Proceeding on Degraded Core Accidents

Refer to the evaluations of 10 CFR 50.34(f)(1)(i), 50.34(f)(2)(ix), 50.34(f)(3)(iv), and 50.34(f)(3)(v) in Sections 20.5.1, 20.5.21, 20.5.44, and 20.5.45, respectively, of this report.

20.4.34 Issue II.D.1: Reactor Coolant System Relief and Safety Valves - Testing Requirements

Refer to the evaluation of 10 CFR 50.34(f)(2)(x) in Section 20.5.22 of this report.

20.4.35 Issue II.D.3: Reactor Coolant System Relief and Safety Valves - Relief and Safety Valve Position Indication

Refer to the evaluation of 10 CFR 50.34(f)(2)(xi) in Section 20.5.23 of this report.

20.4.36 Issue II.E.1.1: System Design -- Auxiliary Feedwater System - Auxiliary Feedwater System Evaluation

This issue is not applicable to the ABWR as discussed in the evaluation of 10 CFR 50.34(f)(1)(ii) in Section 20.5.2 of this report.

20.4.37 Issue II.E.1.2: System Design -- Auxiliary Feedwater System - Auxiliary Feedwater System Automatic Initiation and Flow Indication

This issue is not applicable to the ABWR as discussed in the evaluation of 10 CFR 50.34(f)(2)(xii) in Section 20.5.24 of this report.

20.4.38 Issue II.E.1.3: System Design -- Auxiliary Feedwater System - Update the Standard Review Plan and Develop Regulatory Guidance

This issue requires the NRC to update SRP Section 10.4.9 and issue a regulatory guide on AFWSs.

The staff determined that this issue is not applicable to the ABWR, since a BWR plant does not incorporate an AFWS. Therefore, this issue is not technically relevant to the ABWR design and does not need to be addressed.

20.4.39 Issue II.E.3.1: System Design -- Decay Heat Removal - Reliability of Power Supplies for Natural Circulation

This issue is not applicable to the ABWR as discussed in the evaluation of 10 CFR 50.34(f)(2)(xiii) in Section 20.5.25 of this report.

20.4.40 Issue II.E.4.1: System Design -- Containment Design - Dedicated Penetrations

Refer to the evaluation of 10 CFR 50.34(f)(3)(vi) in Section 20.5.46 of this report.

20.4.41 Issue II.E.4.2: System Design -- Containment Design - Isolation Dependability

Refer to the evaluation of 10 CFR 50.34(f)(2)(xiv) in Section 20.5.26 of this report.

20.4.42 Issue II.E.4.4: System Design -- Containment Design - Purging

Refer to the evaluation of 10 CFR 50.34(f)(2)(xv) in Section 20.5.27 of this report.

20.4.43 Issue II.E.5.1: System Design -- Design Sensitivity of Babcock & Wilcox (B&W) Reactors - Design Evaluation

This issue is not applicable to the ABWR as discussed in the evaluation of 10 CFR 50.34(f)(2)(xvi) in Section 20.5.28 of this report.

20.4.44 Issue II.E.6.1: System Design -- In-Situ Testing of Valves - Test Adequacy Study

The objective of this issue is to establish the adequacy of current requirements for safety-related valve testing. It recommends a study that would result in recommendations for alternate means of verifying performance requirements. This issue was divided into four parts during its resolution: (1) pressure isolation valves (PIVs), (2) check valves, (3) reevaluation of thermal-overload protection provisions of RG 1.106, "Thermal Overload Protection for Electric Motors on Motor-Operated Valves," for MOVs, and (4) in-situ testing of MOVs.

Relative to in-situ testing of PIVs, SSAR Section 3.9.6 requires that all the PIVs listed in Table 3.9-9 be leak tested in accordance with the ABWR TS. This approach is an acceptable way of addressing this part of the issue.

Regarding in-situ testing of check valves, SSAR Section 3.9.6.2.1 indicates that the COL applicant is responsible for performing in-situ full-flow testing of check valves, in addition to the ASME Section XI inservice testing requirements. Advanced, nonintrusive techniques will be used to assess degradation and performance characteristics of the check valves. In addition, the COL applicant is to develop a program to establish the frequency and extent of disassembly and inspection of check valves. This approach is responsive to the applicable guidelines of SECY-90-016 regarding inservice testing of pumps and valves and is an acceptable means of addressing this part of the issue.

For reevaluation of MOV thermal-overload protection, SSAR Tables 1.8-20 and 1.8-22 state that the ABWR design complies with RG 1.106 and NUREG-1296, "Thermal Overload Protection for Electric Motors on Motor-Operated Valves - Generic Issue II.E.6.1," respectively. In addition, SSAR Section 3.9.6.2.2 states that the guidelines of GL 89-10 will be implemented. Since the staff determined that the guidelines of RG 1.106 adequately address thermal-overload protection and GL 89-10 addresses control switch settings, which include thermal overload, the staff concludes that GE's approach to this part of the issue is acceptable.

With respect to in-situ testing of MOVs, SSAR Section 3.9.6.2.2 discusses implementation of the guidelines of GL 89-10 in sufficient detail for the staff to conclude in Section 3.9.6.2 of this report, that the staff's positions in SECY-90-016 as an applicable regulation regarding inservice testing of pumps and valves are addressed and that GE took an acceptable approach to this part of the issue.

20.4.45 Reserved.

20.4.46 Issue II.F.1: Instrumentation and Controls - Additional Accident Monitoring Instrumentation

Refer to the evaluation of 10 CFR 50.34(f)(2)(xvii) in Section 20.5.29 of this report.

20.4.47 Issue II.F.2: Instrumentation and Controls - Identification of and Recovery from Conditions Leading to Inadequate Core Cooling

Refer to the evaluation of 10 CFR 50.34(f)(2)(xviii) in Section 20.5.30 of this report.

20.4.48 Issue II.F.3: Instrumentation and Controls - Instruments for Monitoring Accident Conditions

Refer to the evaluation of 10 CFR 50.34(f)(2)(xix) in Section 20.5.31 of this report.

20.4.49 Issue II.G.1: Electrical Power - Power Supplies for Pressurizer Relief Valves, Block Valves, and Level Indicators

This issue is not applicable to the ABWR as discussed in the evaluation of 10 CFR 50.34(f)(2)(xx) in Section 20.5.32 of this report.

20.4.50 Issue II.J.3.1: General Implications of TMI for Design and Construction Activities -- Management for Design and Construction - Organization and Staffing to Oversee Design and Construction

Refer to the evaluation of 10 CFR 50.34(f)(3)(vii) in Section 20.5.47 of this report.

20.4.51 Issue II.J.4.1: General Implications of TMI for Design and Construction Activities - Revise Deficiency Reporting Requirements

This issue requires the NRC to improve the event-reporting requirements of 10 CFR 50.55(e) and 10 CFR Part 21 to ensure that all reportable items are reported promptly and that information submitted is complete.

SSAR Section 19B.3.1 states that the COL applicant is responsible for providing resolutions of issues identified as "COL Applicant" in the "Safety Issues Index" of SSAR Appendix 19B. The SSAR states that issues are identified for COL action because they pertain to operating personnel issues, operating procedures, and other topics beyond the scope of the ABWR design certification review. SSAR Section 19B.1.1 lists specific documentation the COL applicant is to provide for resolution of such issues.

GE identified this issue for COL applicant action in the "Safety Issues Index." The staff verified that GE established a COL action item (Item 19-28) in SSAR Table 1.9-1 to address unresolved generic and TMI safety issues. This approach is acceptable to the staff. However, the COL applicant's responsibility for resolution of this issue extends only to complying with the current regulations. The staff will review the COL applicant's proposed resolution of this issue on a case-by-case basis, evaluating compliance with the versions of 10 CFR 50.55(e) and 10 CFR Part 21 current at the time of the review.

20.4.52 Issue II.K.1(5): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- IE Bulletins - Safety-Related Valve Position Description

The objective of this issue is to have plants (1) review all valve positions and positioning requirements and positive controls, along with all related test and maintenance procedures, to assure proper ESF functioning, if required, and (2) verify that AFW valves are in the open position.

The verification that AFW valves are in the open position is applicable to PWRs only, since a BWR design does not include an AFWS. Therefore, this portion of the issue it is not technically relevant to the ABWR design and does not need to be addressed.

The review of valve positions and positioning requirements and positive controls is beyond the scope of the ABWR design certification review, and the COL applicant will be responsible for addressing this portion of the TMI item. This was identified in the DFSER as COL Action Item 18.7.2.2-7. The staff verified that GE established a COL action item (Item 18.8.7) in SSAR Section 18.8 for an evaluation of the indication of local valve position. This approach adequately addresses DFSER COL Action Item 18.7.2.2-7 as discussed in Section 18.7.2.2 of this report.

20.4.53 Issue II.K.1(10): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- IE Bulletins - Review and Modify Procedures for Removing Safety-Related Systems from Service

Between April 1, 1979, and July 26, 1979, the former NRC Office of Inspection and Enforcement (IE) issued nine bulletins to various operating plant licensees. This issue requires compliance with the requirements of the IE Bulletins related to operability determination and criteria needed to be met before removing safety-related equipment from service.

The DFSER reported that the SSAR stated that the COL applicant will review all maintenance and test procedures during the preoperational test phase. It also stated that the COL applicant will ensure that the maintenance and test procedures require verification of operability of redundant safety-related systems before removing the safety system from service. The SSAR also stated that the COL applicant will verify the operability of safety-related systems after performing maintenance or tests as part of the test to restore a system to service. The staff concluded that these requirements satisfied this TMI item, but stated that compliance with the TMI item was DFSER COL Action Item 20.3.1-2.

The staff verified that GE established a COL action item in SSAR Section 1A.3.2 for reviewing and modifying, as required, the procedures for removing safety-related systems from and restoring them to service to assure that their operability status is known. (The staff will verify that the procedures satisfy these requirements, which correspond to Item 8 of IE Bulletin 79-08, while reviewing the preoperational testing.) The staff concludes that this approach adequately addresses DFSER COL Action Item 20.3.1-2 and GE adequately addressed this TMI item for the ABWR design.

20.4.54 Issue II.K.1(13): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- IE Bulletins - Proposed Technical Specification Changes Reflecting Implementation of All Bulletin Items

Between April 1, 1979, and July 26, 1979, IE issued nine bulletins to various operating plant licensees. This issue requires operating plants to propose TS changes reflecting implementation of all bulletin items, as required.

Since 1969, there has been a trend to include in TS not only those requirements derived from safety analyses and evaluations, but also many other Commission requirements governing the operation of nuclear power reactors. Therefore, to make TS more operator oriented and to focus on the more important requirements, in its interim "Commission Policy Statement on Technical Specifications Improvements for Nuclear Power Plants," (52 FR 3788 dated February 6, 1987), the Commission established criteria to determine which requirements should remain in TS and which requirements could be relocated to licensee-controlled documents. Based on these criteria and with industry input, the staff developed improved standard technical specifications (STS). The bulletin items covered by this TMI item were considered in the development of the improved STS.

Future TS, including those for advanced reactors such as the ABWR, are to be based on the improved STS. Consequently, this approach supersedes the need to specifically address the bulletin items covered by this issue for the ABWR. Section 16 of this report discusses the development and acceptability of the ABWR TS.

20.4.55 Issue II.K.1.(22): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- IE Bulletins - Describe Automatic and Manual Actions for Proper Functioning of Auxiliary Heat Removal Systems When Feedwater System Not Operable

Refer to the evaluation of 10 CFR 50.34(f)(2)(xxi) in Section 20.5.33 of this report.

20.4.56 Issue II.K.1(23): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- IE Bulletins - Describe Uses and Types of RV Level Indication for Automatic and Manual Initiation Safety Systems

Between April 1, 1979, and July 26, 1979, IE issued nine bulletins to various operating plant licensees. This issue requires the performance of systems reliability analyses and changes in EOPs and operator training to improve the capability of plants to mitigate the consequences of the SBLOCAs and loss-of-feedwater events.

The review criterion for this issue is that the reactor water level must be known to the operators under all normal and abnormal conditions. The instrumentation that serves this purpose must be functional under all conditions, and must provide the operators all the information necessary to assess the state of the plant and what corrective action to be taken when needed.

SSAR Section 1A.2.21, describes the instrumentation that give the operator the information necessary to assess plant status. It provides information for the following conditions:

- Shutdown water level range - used to monitor the reactor water level during shutdown conditions when the reactor is flooded for maintenance and head removal.
- Narrow water level range - RPV taps at the elevation above the main steam outlet nozzle and a tap at an elevation near the bottom of the dryer skirt. This range is used for the water-level control and indication inputs of the feedwater control system.

- Wide water level range - RPV taps at the elevation above the main steam outlet nozzle and taps at the elevation near the top of the active fuel. These instruments provide inputs to various safety systems and ESFs.
- Fuel zone, water level range - RPV taps at the elevation above the main steam outlet nozzle and taps just above reactor internal pump deck. These instruments provide input to water-level indication only.

The instrumentation described above will improve the capability of the plant to mitigate the consequences of the SBLOCAs and loss-of-feedwater events, which is the objective of this issue. Therefore, the staff concludes that GE adequately addressed the requirements of this TMI item for the ABWR design.

20.4.57 Issue II.K.2(9): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Commission Orders on B&W Plants - Analysis and Upgrading of Integrated Control System

This issue is not applicable to the ABWR as discussed in the evaluation of 10 CFR 50.34(f)(2)(xxii) in Section 20.5.34 of this report.

20.4.58 Issue II.K.2(10): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Commission Orders on B&W Plants - Hard-Wired Safety-Grade Anticipatory Reactor Trips

This issue is not applicable to the ABWR as discussed in the evaluation of 10 CFR 50.34(f)(2)(xxiii) in Section 20.5.35 of this report.

20.4.59 Issue II.K.2(16): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Commission Orders on B&W Plants - Impact of RCP Seal Damage Following SBLOCA With Loss of Offsite Power

This issue is not applicable to the ABWR as discussed in the evaluation of 10 CFR 50.34(f)(1)(iii) in Section 20.5.3 of this report.

20.4.60 Issue II.K.3(2): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Report on Overall Safety Effect of Power-Operated Relief Valve (PORV) Isolation

This issue is not applicable to the ABWR as discussed in the evaluation of 10 CFR 50.34(f)(1)(iv) in Section 20.5.4 of this report.

20.4.61 Issue II.K.3(3): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Report Safety and Relief Valve Failures Promptly and Challenges Annually

This issue requires all operating plants and operating license applicants to report safety and relief valve failures promptly and challenges annually.

In the DFSER, the staff reported that SSAR Section 1.9 committed the COL applicant to report the failures of safety and relief valves in the annual report to the NRC in accordance with the requirement of this TMI item and that this approach was acceptable. COL applicant compliance with the requirement was identified as DFSER COL Action Item 20.3.1-3. The staff verified that GE established a COL action item in SSAR Section 1A.3.4. This approach adequately addresses DFSER COL Action Item 20.3.1-3.

20.4.62 Issue II.K.3(11): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Control Use of PORV Supplied by Control Components, Inc., Until Further Review Complete

This issue requires all plants to justify the use of PORVs supplied by Control Components, Inc., that had failed during testing.

SSAR Chapter 15 demonstrates the ABWR's capability to respond to the full spectrum of line breaks and loss-of-feedwater accidents without loss of containment or significant core damage. SSAR Section 5.2 describes the overpressure protection provided by the SRVs performing an overpressure relief valve function, an overpressure safety valve function, or an ADS function. SSAR Section 19B.2.70 states that the SRV for the ABWR is not a PORV by Control Components, Inc. For the safety valve function, the ABWR will use a spring-loaded safety valve with a pneumatic cylinder or piston for power operation in the ADS and relief function. Further, SSAR Section 3.9.3.2.4.2 describes the qualification by type test of the SRVs to IEEE 344, "Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations," for operability during a dynamic event.

The staff concludes that since the ABWR design does not use Control Components, Inc. PORVs and that the safety valve for the ABWR design will be appropriately qualified, this issue is adequately addressed for the ABWR design.

20.4.63 Issue II.K.3(13): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Separation of HPCI and RCIC System Initiation Levels

Refer to the evaluation of 10 CFR 50.34(f)(1)(v) in Section 20.5.5 of this report.

20.4.64 Issue II.K.3(15): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Modify Break Detection Logic to Prevent Spurious Isolation of HPCI and RCIC Systems

The HPCI and RCIC systems use differential pressure sensors on elbow taps in the steam lines to their turbine drives to detect and isolate pipe breaks in the systems. In NUREG-0737, the staff stated that the pipe-break-detection circuitry has resulted in spurious isolation of the HPCI and RCIC systems because of the pressure spike that accompanies the actuation of the systems. This TMI item requires applicants to modify the pipe-break-detection circuitry so that pressure spikes resulting from HPCI and RCIC system initiation will not cause inadvertent system isolation.

SSAR Section 1A.2.23 states that the ABWR design will maintain the high-pressure inventory using the motor-driven HPCF system rather than the turbine-driven HPCI system. Therefore, this TMI item only applies to the turbine-driven RCIC system of the ABWR.

SSAR Section 1A.2.23 states that the ABWR high leak detection and isolation system processes the differential pressure signals that isolate the RCIC turbine. Spurious trips are avoided because the RCIC has a bypass startup system controlled by valves F037 and F045. Upon receiving RCIC start signals, bypass valve F045 opens to pressurize the line downstream and accelerate the turbine. The bypass line through F045 is small (diameter of 1 in.) and naturally limits the initial flow surge to prevent a differential pressure spike in the upstream pipe.

After approximately 5 to 10 seconds, steam supply valve F037 opens to admit full steam flow to the turbine. At this stage, the line downstream is already pressurized. This design feature will reduce the possibility that a pressure spike would occur during any phase of the normal startup process. In the DFSER, the staff concluded that the ABWR design adequately addresses the requirements of this TMI item. However, the staff stated that the COL applicant should test the RCIC bypass startup system during plant startup and designated this as DFSER [COL] Action Item 20.3.1-4. In the advance SER, the staff concluded that since GE had not yet included a COL action item in the SSAR addressing this test, DFSER COL Action Item 20.3.1-4 would remain open until GE had done so. In Amendment 34, GE provided revised SSAR Section 1A.2.23 and provided a new Section 1A.3.8 that establish a COL action item for the COL applicant to test the RCIC bypass startup system during plant startup. This approach adequately addresses DFSER COL Action Item 20.3.1-4.

20.4.65 Issue II.K.3(16): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Reduction of Challenges and Failures of Relief Valves; Feasibility Study and System Modification

Refer to the evaluation of 10 CFR 50.34(f)(1)(vi) in Section 20.5.6 of this report.

20.4.66 Issue II.K.3(17): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Report on Outage of ECC Systems; Licensee Report and Technical Specification Changes

This TMI item required all GE plants to review data on ECC system outages to determine if cumulative outage time limitations should be incorporated in TS. It also required submittal of a report detailing outage dates, lengths of outages, and causes of the outages for all ECCSs.

The DFSER reported that the STS permit several components of the ECCS to have substantial outage times (e.g., 72 hours for one diesel generator; 14 days for the HPCF system). The ABWR TS contain limits on allowable outage times for ECCSs and ECC components but do not specify cumulative outage time limitations for the ECCSs. This was identified in the DFSER as TS Item 20.3.1. The advance SER stated that cumulative outage times were not required to be in the ABWR TS, but would be implemented in plant administrative procedures as discussed in Chapter 16 of the SER. The DFSER, also reported that SSAR Section 1.9 established an action item for the COL applicant to report ECCS outages in annual summary reports to the NRC. The staff also reported in the DFSER that it would review compliance with this requirement during the COL review. This was identified in the DFSER as COL Action Item 20.3.1-5.

The staff verified that GE established a COL action item in SSAR Table 1.9-1 (Item 1.9) to prepare and submit annual reports on ECCS unavailability that also include information on outage dates, lengths, and causes; ECCSs or ECC components involved; and any corrective action taken. SSAR Section 1A.2.25 also states that operating license applicants will establish a plan to meet these reporting requirements. This is approach adequately addresses DFSER COL Action Item 20.3.1-5.

20.4.67 Issue II.K.3(18): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Modification of ADS Logic; Feasibility Study and Modification for Increased Diversity for Some Event Sequences

Refer to the evaluation of 10 CFR 50.34(f)(1)(vii) in Section 20.5.7 of this report.

20.4.68 Issue II.K.3(21): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Restart of Core Spray and LPCI Systems on Low Level; Design and Modification

Refer to the evaluation of 10 CFR 50.34(f)(1)(viii) in Section 20.5.8 of this report.

20.4.69 Issue II.K.3(22): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Automatic Switchover of RCIC System Suction; Verify Procedures and Modify Design

This TMI item required that until the automatic switchover of RCIC system suction from the condensate storage tank to the suppression pool when the condensate storage tank level is low was implemented in BWRs, licensees and applicants would need to verify that clear and cogent procedures existed for the manual switchover of the RCIC system suction.

The RCIC system in the ABWR design includes an automatic switchover feature to change the pump suction source from the RCIC condensate storage tank to the suppression pool. The safety-grade switchover will automatically occur when the RCIC system receives a low-level signal from the condensate storage tank or a high-level signal from the suppression pool. The staff concludes that since the ABWR design incorporates this automatic switchover, there is no need for verification of the manual switchover procedures and that GE's approach adequately addresses the requirements of this TMI item for the ABWR design.

20.4.70 Issue II.K.3(23): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Central Water Level Recording

Refer to the evaluation of 10 CFR 50.34(f)(2)(xxiv) in Section 20.5.36 of this report.

20.4.71 Issue II.K.3(24): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Confirm Adequacy of Space Cooling for HPCI and RCIC Systems

Refer to the evaluation of 10 CFR 50.34(f)(1)(ix) in Section 20.5.9 of this report.

20.4.72 Issue II.K.3(25): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Effect of Loss of ac Power on Pump Seals

Refer to the evaluation of 10 CFR 50.34(f)(1)(iii) in Section 20.5.3 of this report.

20.4.73 Issue II.K.3(27): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Provide Common Reference Level for Vessel Level Instrumentation

This issue required all reactor vessel level instruments on GE plants to be referenced to the same point to avoid potential confusion of operators if different reference points were used for various reactor vessel water level instruments. It recommended the use of either the bottom of the vessel or the active fuel as reasonable common reference points.

The review criterion for this issue is to confirm that the ABWR design has a common zero reference for all water level indications. SSAR Section 19B.2.71 states a common reference for the reactor vessel water level has been set at the top of the active fuel level as described in SSAR Section 7.7. The staff confirmed that SSAR Section 7.7.1.1 (6)(c) indicates that the zero of the reactor vessel water level instruments has been set at the top of the active fuel and the instruments are calibrated to be accurate at the normal power operating point. Therefore, the staff concludes that the ABWR design adequately addresses the requirements of this TMI item.

20.4.74 Issue II.K.3(28): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Study and Verify Qualification of Accumulators on ADS Valves

Refer to the evaluation of 10 CFR 50.34(f)(1)(x) in Section 20.5.10 of this report.

20.4.75 Issue II.K.3(30): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Revised SBLOCA Methods to Show Compliance with 10 CFR Part 50, Appendix A

This issue requires all licensees and applicants to revise, document, and submit for NRC approval the analyses used by NSSS vendors and/or fuel suppliers for SBLOCA analysis in compliance with 10 CFR Part 50, Appendix K. The revised analyses were to account for comparisons with experimental data, including data from the loss-of-fluid test (LOFT) and Semiscale test facilities.

In response to this issue, the BWROG conducted a study that was later endorsed by GE as being applicable to the ABWR within this area. In a letter from R.H. Buchholz (GE) to D.G. Eisenhut (NRC) dated June 26, 1981, GE submitted information on the results of its study and NRC's concerns with the small-break model. The GE information consisted of modeling in SAFE, treatment of pressure variation, and overall model assessment. The staff reviewed this information and concluded that it was acceptable. The staff concluded that the SBLOCA model need not be changed because the test data comparisons and other information submitted by GE acceptably demonstrate that its small-break model complies with the analysis requirements in 10 CFR Part 50, Appendix K.

The staff also reviewed the applicability of the BWROG evaluation to the ABWR design. It concurs that no model changes are required for the ABWR, because it is similar in design to the current BWRs and thus will respond similarly to SBLOCAs. Therefore, the staff concludes that the requirements of this TMI item have been adequately addressed for the ABWR SBLOCA model.

20.4.76 Issue II.K.3(31): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Plant-Specific Calculations to Show Compliance with 10 CFR 50.46

This issue requires licensees and applicants to submit for NRC approval plant-specific calculations, using NRC-approved models for SBLOCAs, to show compliance with 10 CFR 50.46.

The ABWR-specific SBLOCA calculations in SSAR Section 6.3 show compliance with 10 CFR 50.46, as discussed in Section 6.3 of this report. Therefore, the staff concludes that GE has adequately addressed the requirements of this TMI item for the ABWR design.

20.4.77 Issue II.K.3(44): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Evaluation of Anticipated Transients with Single Failure to Verify no Significant Fuel Failure

This issue requires licensees and applicants of GE plants to demonstrate that the core remains covered or provide analysis to show that no significant fuel damage results from core uncovery for anticipated transients combined with the worst single failure and proper operator actions. This category includes transients that result from a stuck-open relief valve.

GE has endorsed the results of the BWROG study (NEDO-24708, "Additional Information Required for NRC Staff Generic Report on Boiling Water Reactors," which had been accepted by the staff) as applicable to ABWR in this area. In a letter dated December 29, 1980, from D.B. Walters (BWROG) to NRC, the BWROG enclosed an evaluation (NEDO-24708) in which it stated that the worst-case transient with single failure combination for BWR/2-6 plants is the loss-of-feedwater event with a failure of the high-pressure ECCS. However, since the ABWR design includes three high-pressure core injection systems, the probability of a loss of all high-pressure ECCS is low. GE also considered an event with a stuck-open relief valve, and a high-pressure ECCS failure, and concluded that the core remained covered throughout the transient either because the RCIC system operated or because the RCS was depressurized by automatic or manual means, permitting low-pressure inventory makeup. GE also assumed the operator would manually depressurize the vessel to permit low-pressure injection.

The staff has reviewed the results of the BWROG's study and its applicability to ABWR and finds that GE has shown that the ABWR design can keep the core covered and have no fuel damage from core uncovery for transients combined with the worst single failure. Therefore, GE adequately addressed the requirements of this TMI item for the ABWR design.

20.4.78 Issue II.K.3(45): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Evaluate Depressurization with Other Than Full ADS

Refer to the evaluation of 10 CFR 50.34(f)(1)(xi) in Section 20.5.11 of this report.

20.4.79 Issue II.K.3(46): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Response to List of Concerns from ACRS Consultant

This TMI item includes 16 questions, developed by Mr. C. Michaelson of the ACRS staff, most of which pertain to PWRs, to which all licensees and applicants were required to respond. GE responded to Mr. Michaelson's concerns as they related to BWRs in a letter dated February 21, 1980.

The staff required GE to review each of Mr. Michaelson's questions and verify that the responses given for BWRs in its February 21, 1980, letter were valid for the ABWR design. In SSAR Table 1A-1, GE responds to all 16 questions. One question pertains to the adequacy of the net positive suction head (NPSH) since the ABWR RCIC and HPCF systems share a common suction line from the condensate storage tank. It is an ABWR design requirement that adequate NPSH be available to the RCIC and HPCF pumps for simultaneous operating modes of these systems. Other questions pertain to the isolation of small breaks, the adequacy of auxiliary feedwater, the recirculation mode of HPCI pumps at high pressure, and the simultaneous operation of HPCI and RHR pumps.

The staff reviewed the responses in SSAR Table 1A-1 and found them to be similar to GE's responses to Mr. Michaelson's questions for operating BWRs that it previously accepted. This approach is acceptable for the ABWR design. Therefore, the staff concludes that GE adequately addressed the requirements of this TMI item for the ABWR design.

20.4.80 Issue III.A.1.1(1): Emergency Preparedness and Radiation Effects -- Improve Licensee Emergency Preparedness - Short Term; Upgrade Emergency Preparedness, Implement Action Plan Requirements for Promptly Improving Licensee Emergency Preparedness

This issue requires approval of the overall state of preparedness, primarily with respect to the capability of offsite agencies to take appropriate emergency actions in the event of nuclear power plant accidents.

SSAR Section 19B.3.1 states that the COL applicant is responsible for providing resolutions of issues identified as "COL Applicant" in the "Safety Issues Index" of SSAR Appendix 19B. The SSAR states that issues are identified for COL action because they pertain to operating personnel issues, operating procedures, and other topics beyond the scope of the ABWR design certification review. SSAR Section 19B.1.1 lists specific documentation the COL applicant is to provide for resolution of such issues.

GE identified this issue for COL applicant action in the "Safety Issues Index." The staff verified that GE established a COL action item (Item 19-28) in SSAR Table 1.9-1 to address unresolved generic and TMI safety issues. This approach is acceptable to the staff. However, the COL applicant's responsibility for resolution of this issue extends only to providing support to the review and approval process and addressing the review findings to the satisfaction of the NRC.

20.4.81 Issue III.A.1.2: Emergency Preparedness and Radiation Effects -- Improve Licensee Emergency Preparedness - Short Term; Upgrade Emergency Preparedness

Refer to the evaluation of 10 CFR 50.34(f)(2)(xxv) in Section 20.5.37 of this report.

20.4.82 Issue III.A.2.1: Emergency Preparedness and Radiation Effects -- Improve Licensee Emergency Preparedness - Long Term; Amend 10 CFR Part 50 and 10 CFR Part 50, Appendix E

This issue required the NRC to revise 10 CFR Part 50, as appropriate, to upgrade the emergency preparedness of nuclear power plants and to revise the inspection program to cover the upgraded requirements.

SSAR Section 19B.3.1 states that the COL applicant is responsible for providing resolutions of issues identified as "COL Applicant" in the "Safety Issues Index" of SSAR Appendix 19B. The SSAR states that issues are identified for COL action because they pertain to operating personnel issues, operating procedures, and other topics beyond the scope of the ABWR design certification review. SSAR Section 19B.1.1 lists specific documentation the COL applicant is to provide for resolution of such issues.

GE identified this issue for COL applicant action in the "Safety Issues Index." The staff verified that GE established a COL action item (Item 19-28) in SSAR Table 1.9-1 to address unresolved generic and TMI safety issues. This approach is acceptable to the staff. However, the COL applicant's responsibility for resolution of this issue extends only to complying with the current regulations. The staff will review the COL applicant's proposed resolution of this issue on a case-by-case basis, evaluating compliance with the version of the emergency preparedness requirements of 10 CFR Part 50, as well as 10 CFR Part 50, Appendix E, current at the time of the review.

20.4.83 Issue III.A.2.2: Emergency Preparedness and Radiation Effects -- Improve Licensee Emergency Preparedness - Long Term; Development of Guidance and Criteria

This issue requires the emergency plans to include information on meteorological criteria, means for promptly notifying the population, and emergency response facilities as detailed in Revision 1 to NUREG-0654 (FEMA-REP-1), "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants."

SSAR Section 19B.3.1 states that the COL applicant is responsible for providing resolutions of issues identified as "COL Applicant" in the "Safety Issues Index" of SSAR Appendix 19B. The SSAR states that issues are identified for COL action because they pertain to operating personnel issues, operating procedures, and other topics beyond the scope of the ABWR design certification review. SSAR Section 19B.1.1 lists specific documentation the COL applicant is to provide for resolution of such issues.

GE identified this issue for COL applicant action in the "Safety Issues Index." The staff verified that GE established a COL action item (Item 19-28) in SSAR Table 1.9-1 to address unresolved generic and TMI safety issues. This approach is acceptable to the staff. The staff will review the COL applicant's proposed resolution of this issue on a case-by-case basis.

20.4.84 Issue III.A.3.3: Emergency Preparedness and Radiation Effects -- Improving NRC Emergency Preparedness - Communications

This issue requires the availability of communication means that will enable the NRC, in the event of a nuclear accident, to (1) monitor and evaluate the situation and (2) potentially advise the plant operating staff, as needed, and (3) in extreme cases, be able to issue orders governing such operations.

SSAR Section 19B.3.1 states that the COL applicant is responsible for providing resolutions of issues identified as

"COL Applicant" in the "Safety Issues Index" of SSAR Appendix 19B. The SSAR states that issues are identified for COL action because they pertain to operating personnel issues, operating procedures, and other topics beyond the scope of the ABWR design certification review. SSAR Section 19B.1.1 lists specific documentation the COL applicant is to provide for resolution of such issues.

GE identified this issue for COL applicant action in the "Safety Issues Index." The staff verified that GE established a COL action item (Item 19-28) in SSAR Table 1.9-1 to address unresolved generic and TMI safety issues. This approach is acceptable to the staff. The staff will review the COL applicant's proposed resolution of this issue on a case-by-case basis.

20.4.85 Issue III.D.1.1: Radiation Protection -- Radiation Source Control - Primary Coolant Sources Outside the Containment Structure

Refer to the evaluation of 10 CFR 50.34(f)(2)(xxvi) in Section 20.5.38 of this report.

20.4.86 Issue III.D.3.3: Radiation Protection -- Worker Radiation Protection Improvement - Inplant Radiation Monitoring

Refer to the evaluation of 10 CFR 50.34(f)(2)(xxvii) in Section 20.5.39 of this report.

20.4.87 Issue III.D.3.4: Radiation Protection -- Worker Radiation Protection Improvement - Control Room Habitability

Refer to the evaluation of 10 CFR 50.34(f)(2)(xxviii) in Section 20.5.40 of this report.

20.5 10 CFR 50.34(f), Additional TMI Requirements

This section addresses staff evaluation of paragraphs (1)(i) through (3)(vii) of 10 CFR 50.34(f).

20.5.1 10 CFR 50.34(f)(1)(i): Consideration of Degraded or Melted Cores in Safety Review - Rulemaking Proceeding on Degraded Core Accidents (TMI Item II.B.8), "Design Alternatives from PRA"

Paragraph (1)(i) of 10 CFR 50.34(f) requires the applicant to "perform a plant/site specific probabilistic risk assessment, the aim of which is to seek such improvements in the reliability of core and containment heat removal systems as are significant and practical and do not impact excessively on the plant."

20.5.1.1 Introduction

GE has made extensive use of the results of the PRA to arrive at a final ABWR design. As a result, the estimated core damage frequency and risk calculated for the ABWR is very low both relative to operating plants and in absolute terms. The low core damage frequency and risk for the ABWR is a reflection of GE's efforts to systematically minimize the effect of initiators or sequences that have been important contributors to core damage frequency in previous BWR PRAs. This has been done largely through the incorporation of a number of hardware improvements in the ABWR design. These include the provision of: three separated divisions of ECCSs, a diverse and independent combustion gas turbine capable of providing ac power to any of the three divisions, an ac-independent water addition system, and an FMCRD system as a backup to the hydraulic drive system. Several improvements have also been incorporated in the ABWR design to mitigate the consequences of a core damage event, including inerting of the containment atmosphere, inclusion of a lower drywell flooder system and a containment overpressure protection (vent) system, the use of basaltic concrete in the lower drywell, and an increased ultimate pressure capacity. These and additional ABWR design features which contribute to low core damage frequency and risk for the ABWR are discussed further in FSER Section 19.1.

In response to 10 CFR 50.34(f)(1)(i), GE provided an initial evaluation of further ABWR design improvements in a February 25, 1992 submittal. This submittal was based on the original PRA results, and included consideration of risk from both internally and seismically initiated events. Based on this evaluation, GE concluded that none of the design improvements considered were cost beneficial.

The initial evaluation was subsequently revised to reflect the results of the updated Level 1 PRA and containment analyses, and was resubmitted on June 30, 1992. The revised assessment was based on the risk reduction potential for internal events only, in contrast to the original evaluation which considered both internally and seismically initiated events. This more limited scope was a consequence of GE's change in methodology from a quantitative treatment of seismic risk to a qualitative, margins-type analysis of seismic events. The net result of the new analysis was an order of magnitude reduction of estimated risk from severe accidents -- from 0.047 person-Sv to 0.0048 person-Sv (4.7 person-rem to 0.48 person-rem) over a 60-year plant life -- largely due to removal of seismic events from the risk profile. The reduced risk in the revised analysis further strengthened GE's original conclusion that none of the design improvements, beyond those already incorporated in the ABWR design, were cost beneficial.

In response to staff comments, GE further modified their evaluation of design improvements to include additional discussion of selected design alternatives, further clarification of the basis for risk-reduction estimates, and additional factors in estimating costs for the modifications. The risk estimates were also adjusted to reflect the results of the final Level 2 analysis and the updated offsite consequence calculations performed as part of the Level 3 portion of the PRA. The final evaluation was submitted as SSAR Appendix 19P.

GE's evaluation of potential design improvements was submitted in response to the requirements of 10 CFR 50.34(f)(1)(i). The staff's review of GE's final evaluation of potential ABWR design improvements is presented below.

20.5.1.2 Estimate of Risk for ABWR

20.5.1.2.1 GE Risk Estimates

GE estimated offsite consequences at five different sites, each representing a different geographic region of the U.S. Offsite consequences were calculated for each release class or case from the Level 2 analysis using the CRAC2 code. The meteorological and population data were obtained from previously developed information contained in NUREG/CR-2239, "Technical Guidance for Siting Criteria Development." The source terms were determined using the MAAP code for each of the release categories as discussed in Section 19.1 of the FSER. The results of the five sets of consequence calculations were averaged together, to represent a typical site in the U.S.

GE's estimate of the cumulative offsite risk to the population within 80 km (50 miles) of the site is provided in SSAR Table 19P-1. The total cumulative exposure calculated by GE is about 0.003 person-Sv (0.3 person-rem), assuming a 60-year plant life. The extremely small level of risk calculated by GE is primarily due to the low estimated core damage frequency for the ABWR (1.6E-7 per reactor year). As a case in point, even if all core damage accidents resulted in the worst release, based on GE's core damage frequency estimates for internal events, the total exposure would only be about 0.3 person-Sv (30 person-rem).

As a result of the low estimated core damage frequency and associated risk levels for the ABWR, any potential modifications which cost more than a few dollars would not be cost-effective, even if the design modification were to totally eliminate the severe accidents or their consequences.

The staff notes that the frequencies of core damage accidents and release bins on which GE based its evaluation of design alternatives are slightly different than those reported in SSAR Section 19D.5. However, these differences are minor and would not alter the essential conclusions of the analysis.

20.5.1.2.2 Staff Review of GE's Risk Estimates

The staff independently estimated the risk associated with severe accidents in the ABWR. A comparison of GE and staff estimates of the person-Sv (person-rem) exposure for each of GE's release classes is provided in FSER Table 20.5.1-1 for internally initiated events. GE's estimates are based on the use of the MAAP and CRAC2 computer codes, and meteorology for five different sites, as described previously. The staff estimates of person-Sv (person-rem) are based on use of the 50th percentile source terms developed during the initial staff review of the ABWR (FSER Table 20.5.1-2), the MACCS offsite consequence code, and meteorology for the Zion site. The staff estimates of the frequency of occurrence of each release class are as reported in FSER Section 19.1.

The GE and staff estimates of person-Sv (person-rem) exposure per event are generally consistent for the large release classes (Cases 7, 8, and 9). The staff's dose estimate is significantly higher for vented scenarios (Case 1) due to the higher fission product release fractions used in the staff's assessment. Similarly, the staff's estimate is much lower for sequences with normal containment leakage (NCL) due to a significantly smaller staff source term for this case. The differences between staff and GE estimates for both these release classes are insignificant, however, since these release classes do not contribute appreciably to total risk.

The estimated total risk over a 60-year reactor operating lifetime is extremely small in both the GE and staff assessment. GE's analysis indicates a total dose of about 0.003 person-Sv (0.3 person-rem) over the 60-year period. The staff's estimate is about 0.01 person-Sv (1 person-rem). The difference is due largely to an increased frequency of early releases in the staff assessment to account for: (1) the contribution from unisolated LOCAs outside containment, and (2) an increased probability of early containment failure from direct containment heating. It can be noted that total risk is dominated by events which lead to early containment failure, and containment bypass. This is consistent with the results from PRAs for operating plants.

Table 20.5.1-1a Comparison of GE and staff adjusted offsite consequences (person-Sv)

Case*	GE Estimates 1/				Staff-Adjusted Estimates			
	Frequency	Person-Sv	Person-Sv per 60 y	Fraction	Frequency 2/	Person-Sv 3/	Person-Sv per 60 y	Fraction
NCL	1.3E-7	96	.00075	.28	1.34E-7	.1	.000008	<.01
Case 1	2.1E-8	1.4E2	.00018	.07	2.08E-8	1.4E3	.0017	.13
Case 7	3.9E-10	2.7E4	.00063	.23	3.6E-10	1.6E4	.00035	.03
Case 8	4.1E-10	3.2E4	.00079	.29	3.6E-9 4/	4.5E4	.0097	.77
Case 9	1.7E-10	3.3E4	.00034	.13	3.3E-10	4.5E4 5/	.00089	.07
Total	1.6E-7		.00269	1.0	1.6E-7		.0126	1.0

Table 20.5.1-1b Comparison of GE and staff adjusted offsite consequences (person-rem)

Case*	GE Estimates 1/				Staff-Adjusted Estimates			
	Frequency	Person-Rem	Person-Rem per 60 y	Fraction	Frequency 2/	Person-Rem 3/	Person-Rem per 60 y	Fraction
NCL	1.3E-7	9600	.075	.28	1.34E-7	100	.0008	<.01
Case 1	2.1E-8	1.4E4	.018	.07	2.08E-8	1.4E5	.17	.13
Case 7	3.9E-10	2.7E6	.063	.23	3.6E-10	1.6E6	.035	.03
Case 8	4.1E-10	3.2E6	.079	.29	3.6E-9 4/	4.5E6	.97	.77
Case 9	1.7E-10	3.3E6	.034	.13	3.3E-10	4.5E6 5/	.089	.07
Total	1.6E-7		0.269	1.0	1.6E-7		1.26	1.0

Notes:

* For case description, refer to Table 20.5.1-2.

1/ Based on information reported in SSAR Table 19P-1.

2/ Based on GE's containment event tree end state frequencies with staff corrections.

3/ Based on staff 50th percentile source terms (see Table 20.5.1-2) and use of MACCS consequence code with Zion site meteorology.

4/ Staff frequency estimate includes: (1) contribution from unisolated LOCAs outside containment, and (2) increased probability of early containment failure from direct containment heating.

5/ Based on the staff's source term estimate for Case 8.

Table 20.5.1-2 Cesium and Iodine release fractions, as estimated by the staff and GE

	GE Case Identifier	Staff's Estimate 5th	Staff's Estimate 50th	Staff's Estimate 95th	GE's Estimate
Cesium	NCL	$2.1x10^{-11}$	$3.2x10^{-9}$	$3.9x10^{-8}$	$5.1x10^{-5}$
	Case 1	$2.7x10^{-6}$	$2.7x10^{-4}$	$5.3x10^{-2}$	$1.3x10^{-5}$
	Case 7	$6.6x10^{-6}$	$2.4x10^{-3}$	$1.4x10^{-1}$	$9.9x10^{-2}$
	Case 8	0.002	0.06	0.75	0.25
	Case 9	$8.3x10^{-4}$	$7.8x10^{-3}$	$1.7x10^{-1}$	0.36
Iodine	NCL	$3.4x10^{-11}$	$2.3x10^{-6}$	$3.8x10^{-4}$	$3.8x10^{-6}$
	Case 1	$8.5x10^{-6}$	$9.2x10^{-4}$	$6.1x10^{-2}$	$1.5x10^{-7}$
	Case 7	$4.7x10^{-4}$	$9.7x10^{-2}$	$3.2x10^{-1}$	$8.9x10^{-2}$
	Case 8	0.007	0.19	0.69	0.19
	Case 9	0.002	0.10	0.16	0.37

Case	Description
NCL	Normal containment leakage, no containment failure.
Case 1	Fission products scrubbed by suppression pool before release, includes the "venting" sequences.
Case 7	Late containment failure due to overpressurization, spray available, no suppression pool scrubbing.
Case 8	Early containment failure, no suppression pool scrubbing.
Case 9	Late containment failure due to overpressurization, no spray, no suppression pool scrubbing.

As discussed below, the staff based its assessment of risk-reduction potential for the various ABWR design improvements on the staff estimate of risk for internally initiated events, that is, 0.01 person-Sv (1 person-rem). However, the validity of the conclusions of this analysis were tested by considering the uncertainties in core damage frequency estimates, as well as the potential frequency of core damage due to external events.

20.5.1.3 Identification of Potential Design Improvements

20.5.1.3.1 List of Potential Design Improvements

GE identified a set of potential design improvements for the ABWR based on a survey of previous industry- and NRC-sponsored studies of preventative and mitigative features which address severe accidents. Through this effort GE developed a composite list of 68 potential design improvements, organized into 14 general categories. These categories and many of the design improvements are the same as considered for the General Electric Standard Safety Application Report (GESSAR) II design. The resulting list of potential design improvements for the ABWR is presented in SSAR Table 19P-3.

GE eliminated certain design improvements from further consideration on the basis that they are either already incorporated into the ABWR design, or not applicable to the ABWR design. Examples of design improvements already included in the design are: improved low-pressure injection system (fire pump), RWCU decay heat removal, low flow unfiltered vent, and combustible gas control (inerted containment). On the basis of this screening, 21 potential design improvements covering 12 of the 14 general categories were retained for further consideration. The set of design improvements selected for further evaluation is listed in Table 20.5.1-3, and summarized in Section 20.5.1.3.2 of this report.

The staff and its contractor, Brookhaven National Laboratory (BNL), have reviewed the set of potential design improvements identified by GE (SSAR Table 19P-3) and find it to be comprehensive. The list includes all improvements identified as part of the GESSAR II review, and the NRC Containment Performance Improvement (CPI) program. The staff notes that the set of design improvements is not all-inclusive, in that additional, perhaps less expensive design improvements can always be postulated. However, the staff concludes that the benefits offered by any additional modifications would not likely exceed those for the modifications evaluated. The staff also concludes that the costs of alternative improvements would not likely be less than that of the lowest cost improvements evaluated, when the subsidiary costs associated with maintenance, procedures, and training are considered. On this basis, the staff concludes that the set of potential design improvements identified by GE is acceptable.

The set of design improvements selected for further evaluation also appears to be reasonable. The staff notes that the improvements considered include a filtered containment vent, and flooded rubble bed core retention device, which are two improvements specifically cited in NUREG-0660 for evaluation as part of TMI Item II.B.8. A modification intended to delay the time of reactor vessel failure through the use of alternative materials for the bottom head penetration piping was also considered by GE. This modification was instigated by the results of recent analyses of reactor vessel bottom head failure as documented in draft NUREG/CR-5642.

Finally, it should be noted that certain features of several of the improvements selected for further evaluation have been or will be incorporated as part of the ABWR design, independent of this evaluation of design improvements. For example, severe accident EPGs or accident management guidelines (AMGs) will be implemented by the COL applicant as part of its accident management program, as discussed in FSER Section 19.2, and much of the benefits of improved maintenance procedures or manuals will be achieved through the COL applicant's reliability assurance program, as discussed in Section 19.1.3.7 of this report.

20.5.1.3.2 Description of Design Improvements

A description of the design improvements selected by GE for cost-benefit evaluation is provided in SSAR Sections 19P.3 and 19P.4, and summarized below.

- Severe accident EPGs or AMGs - extend the EPGs and EOPs to address arrest of a core melt, emergency planning, radiological release assessment and other areas related to severe accidents. This modification would lead to increased reliability of manual actions in response to core-damage events.

- Computer-aided instrumentation - provide artificial intelligence-based improvements to plant status monitoring, including human-engineered displays of important variables in the EPGs and AMGs, and procedural options for the operator to evaluate during severe accidents. This modification would lead to increased reliability of manual actions to prevent core damage.

Table 20.5.1-3 Potential Design Improvements and Associated Costs (Provided by GE)

	Modification	Estimated Cost ($M)	Person-Sv (Person-Rem) Averted		Cost($M)/Person-Sv (Person-Rem) Averted	
1.	**Accident Management**					
1a.	Severe accident EPGs	0.60	0.00015	(0.015)	4,000	(40)
1b.	Computer-aided instrumentation	0.60	0.00010	(0.01)	>4,000	(>40)
1c.	Improved maintenance procedures/manuals	0.30	0.00016	(0.016)	1,880	(18.8)
2.	**Decay Heat Removal**					
2a.	Passive high pressure system	1.75	0.00138	(0.138)	1,270	(12.7)
2b.	Improved depressurization	0.60	0.00042	(0.042)	1,430	(14.3)
2c.	Suppression pool jockey pump	0.12	0.00002	(0.002)	>4,000	(>40)
2d.	Safety-related condensate storage tank	1.0	0.00010	(0.01)	>4,000	(>40)
3.	**Containment Capability**					
3a.	Larger volume containment	8.0	0.00150	(0.15)	>4,000	(>40)
3b.	Increased containment pressure capacity	12.0	0.00020	(0.02)	>4,000	(>40)
3c.	Improved vacuum breakers	0.10	0.0000003	(0.00003)	>4,000	(>40)
3d.	Improved bottom head penetration design	0.75	0.00057	(0.057)	1,320	(13.2)
4.	**Containment Heat Removal**					
4a.	Larger volume suppression pool	8.0	0.000002	(0.0002)	>4,000	(>40)
5.	**Containment Atmosphere Mass Removal**					
5.a	Low-flow filtered vent	3.0	0.00014	(0.014)	>4,000	(>40)
7.	**Containment Spray Systems**					
7a.	Drywell head flooding	0.10	0.00060	(0.06)	1,700	(1.7)
8.	**Prevention Concepts**					
8a.	Additional service water loop	6.0	0.00016	(0.016)	>4,000	(>40)
9.	**AC Power Supplies**					
9a.	Steam driven turbine generator	6.0	0.00052	(0.052)	>4,000	(>40)
9b.	Alternate pump power source	1.2	0.00069	(0.069)	1,740	(17.4)
10.	**DC Power Supplies**					
10a.	Dedicated RHR dc Power Supply	3.0	0.00069	(0.069)	>4,000	(>40)
11.	**ATWS Capability**					
11a.	ATWS-sized vent	0.30	0.00030	(0.03)	1,000	(10)
13.	**System Simplification**					
13a.	Reactor building sprays	0.10	0.00017	(0.017)	5,900	(5.9)
14.	**Core Retention Devices**					
14a.	Flooded rubble bed	18.8	0.00001	(0.001)	>4,000	(>40)

- Improved maintenance procedures or manuals - provide improved maintenance manuals and additional information on the components important to the risk of the plant within the GE scope of supply. These manuals and information would lead to increased reliability of important equipment.

- Passive high pressure system - add an isolation condenser-type high pressure system for removing decay heat from both the core and containment. The benefit of this system would be equivalent to an additional RCIC system and containment heat-removal system.

- Improved depressurization - provide manually controlled, seismically protected air operators to permit manual depressurization in the event of loss of dc control power or control air events. Improved depressurization would reduce the threat of containment failure due to high pressure melt ejection, and allow more reliable access to low-pressure systems.

- Suppression pool jockey pump - add a small, ac-independent makeup pump to provide low-pressure decay heat removal from the reactor pressure vessel using suppression pool water as the source. This modification would have a benefit similar to that provided by the ac-independent water addition mode of RHR (fire water), but without the associated long-term containment inventory concerns.

- Safety-related condensate storage tank - upgrade the structure of the condensate storage tank such that it would be available to provide makeup to the reactor following a large seismic event. This would enhance core injection capabilities in seismic events, by providing an alternative to the suppression pool as a source of water for injection.

- Larger volume containment - increase the volume of containment by a factor of two. This would reduce the peak pressures associated with energetic events, thereby reducing the potential for drywell head failure, and would also reduce the rate of long-term containment pressurization, thereby delaying the time of fission product release.

- Increased containment pressure capacity - increase the ultimate pressure capacity of containment (including seals) to a level at which all release modes except normal containment leakage are eliminated.

- Improved vacuum breakers - add a second vacuum breaker valve in each of the eight drywell-to-wetwell vacuum breaker lines to make these valves redundant. This modification would reduce the potential for suppression pool bypass due to stuck-open or leaking vacuum breaker valves.

- Improved bottom head penetration design - change the transition piece (used to connect the stainless steel RPV drain line to the RPV) from carbon steel to a material with a higher melting point, such as inconel. Also, establish external welds or restraints on the CRDs external to the vessel so that the drives would not be ejected following failure of the internal welds. This modification would delay the time of reactor vessel failure by several hours, thereby increasing the potential to arrest core damage in the vessel, but may also increase the potential for gross failure of the lower head.

- Larger volume suppression pool - increase the size of the suppression pool to provide reduced pool heatup rates. This modification would reduce the frequency of core melt from Class II sequences (loss of containment heat removal), and ATWS sequences by providing additional time for operator actions and recovery of heat removal systems.

- Low-flow filtered vent - add a filter system external to the containment to further reduce the magnitude of radioactive releases via containment venting. The system would be similar to the multi-venturi scrubbing systems implemented in some plants in Europe. The system would provide fission product scrubbing beyond that presently offered by the suppression pool, but would not affect releases due to drywell head failure and containment bypass sequences.

- Drywell head flooding - provide an additional line to permit intentional flooding of the upper drywell head using the existing fire water additional system. Drywell head flooding would cool the drywell head seal and provide fission product scrubbing in the event of drywell head leakage. Instrumentation and controls to permit manual control from the control room were considered part of this modification.

- Additional service water loop - provide an additional service water cooling loop (pump and heat exchanger) to improve the overall reliability of the service water network. This cooling loop would be capable of removing heat from any one of the three divisions. This would reduce the frequency of sequences involving failure of injection due to loss of component cooling.

- Steam-driven turbine generator - add a steam-driven turbine generator that uses reactor steam and exhausts to the suppression pool. The benefits of this modification would be a further reduction in the frequency of station blackout sequences, similar to that which might be obtained by adding another gas turbine generator.

- Alternate pump power source - provide a separate diesel generator and supporting auxiliaries to power the feedwater or condensate pumps. This modification would remove the reliance of these pumps on offsite power, and permit them to be used as a backup to HPCF and low-pressure core flooder.

- Dedicated dc power supply - provide a separate, diverse dc power source (fuel cell or separate battery) to supply a dc motor-pump combination for RPV and containment cooling. This modification would further reduce the risk from loss of offsite power and station blackout.

- ATWS-sized vent - provide a wetwell vent line capable of passing the steam flow associated with ATWS. The system would be significantly larger than the existing containment overpressure protection system (COPS) design, and manually initiated from the control room. This system would prevent containment overpressure failure in ATWS events, and thereby prevent core damage.

- Reactor building sprays - modify the fire water spray system in the reactor building to spray in areas vulnerable to release. This modification would reduce the risk associated with releases into the reactor building, such as drywell head failures and containment bypass events, but would not impact releases via COPS.

- Flooded rubble bed - provide a bed of refractory pebbles that would be flooded with water. The rubble bed would impede the flow of molten corium to the concrete drywell structures, and increase the available heat transfer area, thereby enhancing debris coolability. This modification would further reduce the potential for core concrete interactions in the ABWR. A major drawback of the modification is that additional experimental testing would be necessary to validate the concept for the ABWR application.

20.5.1.4 Risk Reduction Potential of Design Improvements

20.5.1.4.1 GE Evaluation of Risk Reduction Potential

GE used the reduction in cumulative risk of accidents occurring during the life of the plant as the basis for estimating the benefit that could be derived from plant improvements. Estimates of risk reduction were developed by determining the approximate effect of each modification on the frequency of the various release classes in the PRA. GE's basis for estimating the risk reduction for each design improvement is provided in SSAR Section 19P.4, and summarized in Table 20.5.1-4 of this report.

The staff reviewed GE's bases for estimating the risk reduction associated with the various design improvements. The staff notes that considerable judgement was exercised in estimating the risk reduction potential, but that in general, the rationale and assumptions on which the risk reduction estimates are based (center column of Table 20.5.1-4) are reasonable, and in many cases conservative. However, this is not to say that the estimates of person-Sv (person-rem) averted are conservative, since the staff is not in complete agreement with GE's characterization of baseline risk. For example, the risk-reduction potential for improved vacuum breakers appears to be underestimated in GE's analysis. GE estimates that improved vacuum breakers (addition of a second vacuum breaker valve in series with each of the existing valves) would reduce risk by about 3E-7 person-Sv (0.00003 person-rem). This value is in large part due to significant credit for fission product removal by wetwell sprays (when available), and the failure to account for the impact of the design improvement on bypass scenarios in which sprays are not available. GE's risk-reduction estimate for this improvement would increase by at least three orders of magnitude if just the latter factor was accounted for. Nevertheless, the risk reduction would remain small since the probability of the events involved is on the order of 1E-10 per reactor year.

20.5.1.4.2 Staff Evaluation of Risk-Reduction Potential

In view of the extremely small residual risk for the ABWR, rather than perform an independent assessment of the risk-reduction potential of each ABWR design improvement, the staff used a bounding assumption that each improvement would eliminate all of the risk for the ABWR (0.01 person-Sv (1 person-rem) for the 60-year plant life). This approach tends to overestimate the benefits because the ABWR risk profile reflects contributions from several unique types of sequences (e.g., station blackout, containment bypass, and LOCAs). An individual design improvement would generally reduce or eliminate some of these contributors but would not be effective on others. Moreover, there are numerous and diverse modes of containment failure which must be dealt with to ensure containment integrity in a severe accident. Thus, a carefully selected set of plant improvements would generally be needed, each one acting on particular components of risk, to effectively and significantly reduce total risk.

Table 20.5.1-4 Summary of GE's assessment of risk reduction for candidate design improvements

Potential ABWR design modification	GE's basis for estimating risk reduction	Person-SV (person-rem) averted
Accident Management		
Severe accident EPGs/AMGs	10% reduction in failure rates for manually initiated mitigative actions	0.00015 (0.015)
Computer-aided instrumentation	10% reduction in failure rates for manually initiated preventive actions	0.00010 (0.01)
Improved maintenance procedures/manuals	10% improvement in reliability of HPCF, RCIC, RHR, LPCF	0.00016 (0.016)
Decay Heat Removal		
Passive high pressure system	Equivalent to adding a diverse RCIC and RHR system with 10% unavailability	0.00138 (0.138)
Improved depressurization system	Factor of 2 reduction in depressurization failure rates	0.00042 (0.042)
Suppression pool jockey pump	10% improvement in reliability of low pressure makeup (resulting in 2% reduction in core damage frequency from low pressure sequences	0.00002 (0.002)
Safety-related condensate storage tank	Engineering judgement	0.00010 (0.01)
Containment Capability		
Larger volume containment	Elimination of all containment release modes involving drywell head failure (Cases 3, 6, 7, 8, 9)	0.00150 (0.15)
Increased containment pressure capacity	Elimination of all containment release modes except normal containment leakage	0.00020 (0.02)
Improved vacuum breakers	Elimination of releases from Release Class 2	.0000003 (0.00003)
Improved bottom head penetration design	Factor of 2 increase in the probability of arresting core damage in-vessel	0.00057 (0.057)

Table 20.5.1-4 Summary of GE's assessment of risk reduction for candidate design improvements (continued)

Potential ABWR design modification	GE's basis for estimating risk reduction	Person-SV (person-rem) averted
Containment Heat Removal		
Large volume suppression pool	Elimination of Class II Sequences	.000002 (0.0002)
Containment Mass Removal		
Low-flow filtered vent	Elimination of the risk associated with releases via COPS	0.00014 (0.014)
Containment Spray Systems		
Drywell head flooding	Elimination of drywell head over-temperature failures and reduction in releases from drywell head over-pressure failures	0.00060 (0.06)
Prevention Concepts		
Additional service water Loop	10% increase in reliability of HPCF, RCIC, RHR, LPCF	0.00016 (0.016)
AC Power Supplies		
Steam-driven turbine generator	80% reduction in the diesel generator common mode failure rate	0.00052 (0.052)
Alternate pump power source	Equivalent to adding a diverse RCIC system	0.00069 (0.069)
DC Power Supplies		
Dedicated dc power supply	Factor of 10 increase in RCIC availability in LOOP and SBO sequences	0.00069 (0.069)
ATWS Capability		
ATWS-sized vent	Elimination of risk from ATWS (Case 9)	0.00030 (0.03)
System Simplification		
Reactor building sprays	10% reduction in risk from releases through the reactor building	0.00017 (0.017)
Core Retention Devices		
Flooded rubble bed	Elimination of sequences with core concrete interactions, except those with failure of containment heat removal (1% of Cases 1, 6, and 7)	0.000010 (0.001)

20.5.1.5 Cost Impacts of Candidate Design Improvements

GE determined the approximate costs for each design improvement. The costing methodology and assumptions are described in SSAR Section 19P.1.3. The cost basis for each plant improvement is provided in SSAR Section 19P.5 on an item-by-item basis.

GE stated in the SSAR that the cost estimates represent the incremental costs that would be incurred in a new plant, rather than costs that would apply on a backfit basis. GE also stated that the costs were intentionally biased on the low side, but that all known or reasonably expected costs were accounted for so that a reasonable assessment of the minimum cost would be obtained.

For modifications which reduce core damage frequency, GE reduced the costs of the design improvements by an amount proportional to the reduction in the present worth of the risk of averted onsite costs. Onsite costs considered include replacement power at $.013/kwh differential cost, direct accident costs including onsite cleanup at $2 billion, and the economic loss of the facility at $1.4 billion. The resulting costs for each of the design improvements are provided in Table 20.5.1-3.

The staff reviewed the bases for GE's cost estimates and finds them to be reasonable. For certain improvements, the staff also compared GE's cost estimates with estimates developed elsewhere for similar improvements, even though the bases for some of these cost estimates were different. The staff considered the cost estimates developed as part of: the evaluation of design improvements for GESSAR II (NUREG-0979, Supplement 4), and the review of SAMDAs for Limerick and Comanche Peak (NUREG-0974 and -0775, respectively).

The staff noted a number of inconsistencies in the cost estimates. For example, GE's cost estimates for certain improvements such as improved vacuum breakers ($100,000), modified reactor building sprays ($100,000), and ATWS-sized vent ($300,000) were are considerably less than expected. The costs for certain other improvements was much higher than expected, such as improved bottom head penetration design ($750,000) and flooded rubble bed (approximately $19 million).

It should be noted that only rough approximations of the costs of specific improvements are possible at this time. Large uncertainties exist because detailed designs are not available and because experience with construction and licensing problems that could surface with this type of work is limited. Nevertheless, the staff views GE's approximate cost estimates as adequate, given the uncertainties surrounding the underlying cost estimates, and the level of precision necessary given the greater uncertainty inherent on the benefit side, with which these costs were compared.

20.5.1.6 Cost-Benefit Comparison

A cost-benefit comparison was performed to determine whether any of the potential severe accident design features could be justified. GE's estimates of the cost per person-Sv (person-rem) averted for the various design improvements are presented in Table 20.5.1-3. The GE values are based on the risk- reduction estimates reported in Tables 20.5.1-3, and 20.5.1-4 of this report. The staff analysis is based on the conservative assumption that each design improvement would eliminate all of the residual risk (0.01 person-Sv (1 person-rem) over the 60-year plant life).

Consistent with current NRC practice (NUREG-3568), GE used a screening criterion of $100,000 per person-Sv ($1000 per person-rem) averted to identify whether any of the design improvements could be cost-effective. As shown in Table 20.5.1-3, the potential cost per averted person-rem ranges from about $1.7 million to far-in-excess of $40 million for the various suggested modifications according to the GE evaluation. Thus, this far exceeds the $100,000 per person-Sv ($1000 per person-rem) criterion. On this basis, GE concluded that no additional modifications to the ABWR design are warranted.

The staff's assessment similarly indicates that none of the design improvements approach a level where they could be considered cost-effective, in spite of the significant conservatisms in assessing risk-reduction potential in the staff's analysis. The staff notes that the lowest cost modifications were estimated to cost about $100,000, and realistically would only partially reduce the residual risk for the ABWR. Even though the cost of implementing design improvements in the ABWR may be less than for an existing plant, given that the ABWR has not yet been constructed, relatively large costs are still to be anticipated for many of the design improvements because they would involve first-of-a-kind engineering, and would need to be integrated within the existing design. In addition, the introduction of a new system will trigger a series of related requirements such as incremental training, procedural changes, and possible licensing requirements. These are all legitimate costs that require consideration in a comprehensive cost estimate. The staff concludes that none of the modifications evaluated would be cost-effective given the low residual risk for the ABWR, and the $1000 per person-rem criterion.

The staff has considered the robustness of this conclusion relative to a number of critical assumptions in the analysis as described below. These involve: the effect of uncertainties in estimating core damage frequency, the use of alternative cost-benefit criteria, and the inclusion of external events within the scope of the analysis.

Based on uncertainty analyses performed by GE for the Level 1 portion of the PRA (see FSER Section 19.1.3.2.6), the 95th percentile core damage frequency is 4.5E-7 per reactor year. This is a factor of three higher than the mean value on which the cost-benefit analysis is based, but still very low both compared to operating plants and in absolute terms. Even if the benefits of the various design improvements were re-quantified on the basis of this upper-bound value, none of the improvements would become cost beneficial. This would remain the case even if the cost-benefit criterion was also increased by a factor of 10 to $1 million per person-Sv ($10,000 per person-rem) averted.

If external events are included, the estimate of ABWR risk could be one or possibly two orders of magnitude higher than considered in this analysis. For example, based on the BNL review of GE's original seismic PRA, as documented in the DSER (SECY-91-309), the total risk from internal and seismic events for the 60-year plant life would range from about 0.4 to 2 person-Sv (40 to 200 person-rem) depending on the site population. However, the value for the final ABWR design would be somewhat less since these estimates do not account for plant improvements incorporated in the design subsequent to the original PRA analysis, including upgrading the seismic capability of the diesel-driven fire water pump.

Even assuming the higher of these two risk estimates and complete elimination of all risk, any design modifications or combinations which cost more than $200,000 would not be cost-effective. This would eliminate most of the candidate design modifications from further consideration. Based on the GE analysis, those modifications which were estimated to cost less than $200,000 have a relatively low risk-reduction potential, and would generally eliminate only about 10 percent of the residual risk from internal events. The improvements are also not expected to be effective in eliminating most of the added risk from seismic events. Since the minimum cost of these systems would be about $100,000, none of these improvements are expected to be cost-effective when their actual effectiveness in reducing risk is taken into account.

The staff concludes that with the significant margins in the results of the cost-benefit analysis, the findings of the analysis would be unchanged even considering the above factors.

20.5.1.7 Conclusions

As discussed in Chapter 19.1.3.2.2 of this report, GE has made extensive use of the results of the PRA to arrive at a final ABWR design. As a result, the estimated core damage frequency and risk calculated for the ABWR is very low both relative to operating plants and in absolute terms. The low core damage frequency and risk for the ABWR is a reflection of GE's efforts to systematically minimize the effect of initiators or sequences that have been important contributors to core damage frequency in previous BWR PRAs. This has been done largely through the incorporation of a number of hardware improvements in the ABWR design. These include providing three separated divisions of ECCSs, a diverse and independent combustion gas turbine capable of providing ac power to any of the three divisions, an ac-independent water addition system, and an FMCRD system as a backup to the hydraulic drive system. Several improvements have also been incorporated in the ABWR design to mitigate the consequences of a core damage event, including inerting of the containment atmosphere, inclusion of a lower drywell flooder system and a containment overpressure protection (vent) system, the use of basaltic concrete in the lower drywell, and an increased ultimate pressure capacity. These and additional ABWR design features which contribute to low core damage frequency and risk for the ABWR are discussed further in Section 19.1.3.2.3 of this report.

Because the ABWR design already includes numerous plant features oriented towards reducing core damage frequency and risk, the benefits and risk- reduction potential of additional plant improvements is significantly reduced. This is true for both internally and externally initiated events. For example, the ABWR seismic design basis (0.3g safe shutdown earthquake) has been shown to result in significant ability to withstand earthquakes well beyond the design basis, as characterized by a high confidence with low probability of failure (HCLPF) value of 0.6g. Moreover, with the features already incorporated in the ABWR design, the ability to estimate core damage frequency and risk approaches the limitations of probabilistic techniques. Specifically, when core damage frequencies of one in a hundred thousand or a million years are estimated in a PRA, it is the areas of the PRA where modelling is least complete, or supporting data is sparse or even nonexistent that could actually be the more important contributors to risk. Areas not modelled or incompletely modelled include human reliability, sabotage, rare initiating events, construction or design errors, and systems interactions. Although improvements in the modelling of these areas may introduce additional contributors to core damage frequency and risk, the staff does not expect that they would be significant in absolute terms.

10 CFR 50.34(f)(1)(i) requires the applicant to perform a plant- or site- specific probabilistic risk assessment, the aim of which is to seek such improvements in the reliability of core and containment heat removal systems as are significant and practical and do not impact excessively on the plant. The staff concludes that the ABWR PRA, and GE's use of the insights of this study to improve the design of the ABWR meets this requirement. The staff concurs with the GE conclusion that none of the potential design modifications evaluated are justified based on cost-benefit considerations. The staff also concludes that it is unlikely that any other design changes would be justified on the basis of person-Sv (person-rem) exposure considerations, because the estimated core damage frequencies would remain very low on an absolute scale.

20.5.2 10 CFR 50.34(f)(1)(ii): System Design -- Auxiliary Feedwater System - Auxiliary Feedwater System Evaluation (TMI Item II.E.1.1)

Paragraph (1)(ii) of 10 CFR 50.34(f) requires the performance of an evaluation of the proposed AFWS of PWR plants to include a simplified AFWS reliability analysis using event-tree and fault-tree logic techniques, design review of AFWS, and an evaluation of AFWS flow design bases and criteria.

This requirement is applicable to PWRs only, since a BWR design does not include an AFWS. Therefore, it is not technically relevant to the ABWR design and does not need to be addressed.

20.5.3 10 CFR 50.34(f)(1)(iii):

Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Commission Orders on B&W Plants - Impact of RCP Seal Damage Following SBLOCA With Loss of Offsite Power (TMI Item II.K.2(16)) and

Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Effect of Loss of ac Power on Pump Seals (TMI Item II.K.3(25))

TMI Item II.K.2(16) is applicable to PWRs only and does not need to be addressed for the ABWR, but II.K.3(25) applies to BWRs. Paragraph (1)(iii) of 10 CFR 50.34(f) requires analyses or experiments to determine the consequences at each plant of a loss of cooling water to the reactor recirculation pump seal coolers. Pump seals should be designed to withstand a complete loss of ac power for at least two hours and the adequacy of the seal design should also be demonstrated. The design should prevent an excessive loss of reactor coolant inventory after an anticipated operational occurrence. It is assumed that the loss of ac power constitutes a loss of offsite power.

SSAR Sections 5.4.1 and 1A.2.30 state that the ABWR design features reactor internal pumps (RIPs) that do not require pump shaft seals. During a loss of ac power, the RIPs are shut down automatically, but there are no shaft seals that require restoration of cooling. During its review of this issue, the staff required GE to confirm that the failure of the following systems would not cause a LOCA

- recirculation motor cooling system
- recirculation motor seal purge system
- recirculation motor inflatable shaft seal subsystem.

GE confirmed in SSAR Section 1A.2.30 that an ac failure would temporarily disrupt the operation of these systems, but their failure would not generate a LOCA. Therefore, the staff concludes that GE adequately addressed the requirements of this TMI item for the ABWR design.

20.5.4 10 CFR 50.34(f)(1)(iv): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Report on Overall Safety Effect of PORV Isolation (TMI Item II.K.3(2))

Paragraph (1)(ii) of 10 CFR 50.34(f) requires performance of an analysis of the probability of a SBLOCA caused by a stuck-open PORV of PWR plants. If the probability is a significant contributor to the probability of SBLOCAs from all causes, it also requires a description and evaluation of the effect on SBLOCA probability of an automatic PORV isolation system that would operate when the RCS pressure falls after the PORV has opened.

This requirement is applicable to PWRs only, therefore, it is not technically relevant to the ABWR design and does not need to be addressed.

20.5.5 10 CFR 50.34(f)(1)(v): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Separation of HPCI and RCIC System Initiation Levels (TMI Item II.K.3(13))

Paragraph (1)(v) of 10 CFR 50.34(f) requires an evaluation of the safety effectiveness of providing for separation of the HPCI and RCIC system initiation levels so that the RCIC system initiates at a higher water level than the HPCI system, and of providing that both systems restart on low water level. For plants with high pressure core spray (HPCS) systems in lieu of HPCI systems, HPCS system is to be evaluated in lieu of the HPCI system.

In NUREG-0737, the staff stated that the initiation levels of the HPCI and RCIC systems should be separated so that the RCIC system initiates at a higher water level than the HPCI system. Further, the initiation logic of the RCIC system should be modified so that the RCIC system will restart upon receiving a low water level signal. These changes could reduce the number of challenges to the HPCI system and could result in less stress on the reactor vessel from cold water injection. Applicants were required to submit the analyses of these changes to the NRC staff and implement the changes if justified by the analyses.

The ABWR design is consistent with this position. The ABWR incorporates an HPCF system that initiates at reactor vessel level 1.5 and an RCIC system which initiates at level 2. At level 8, the injection valves for the HPCF and the RCIC steam supply and injection valves automatically close to prevent water from entering the main steam lines.

If the RPV again reaches a low level, the RCIC steam supply and injection valves automatically reopen to allow the RCIC to continue flooding the vessel. The HPCF injection valves will also automatically reopen unless the operator previously closed them manually.

The ABWR has three high-pressure makeup systems, two HPCFs and one RCIC. Current BWRs have the HPCS or HPCI, and RCIC. Thus, the water level set points for ABWR must meet the following requirements:

- During anticipated abnormal transients, including the loss of all feedwater flow event, the RCIC system will prevent the minimum water level from dropping below vessel level 1.5. This requirement is to minimize the challenge to the HPCFs.
- The ADS will not likely be actuated during any abnormal transient with a failure of RCIC. Therefore, the reactor isolation and two HPCFs initiated at vessel level 1.5 will prevent the water level from dropping below level 1 during a loss-of-all-feedwater-flow transient with RCIC failure.

The staff found that GE provided different initiation levels for HPCF and RCIC so that the RCIC system initiates at a higher water level than the HPCF system, as well as the automatic restart of RCIC on low level. GE's proposed response adequately addresses the requirements of this TMI item for the ABWR design. An in-depth discussion of the RCIC system is provided in Section 5.4.6 of this report and of RCIC and HPCF in Section 6.3.1 of this report.

20.5.6 10 CFR 50.34(f)(1)(vi): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Reduction of Challenges and Failures of Relief Valves; Feasibility Study and System Modification (TMI Item II.K.3(16))

Paragraph (1)(vi) of 10 CFR 50.34(f) requires the performance of a study to identify practicable system modifications that would reduce challenges and failures of relief valves in BWRs, without compromising the performance of the valves or other systems.

In 1980, the staff determined that in the prior three years of operation of all BWR plants, there were approximately 30 failures of relief valves to close in 73 reactor-years (0.41 failures per reactor-year). This demonstrated that the failure of a relief valve to close was the most likely cause of an SBLOCA. The high failure rate is the result of a high relief-valve challenge rate and a relatively high probability of failure for each challenge (0.16 failures per challenge). Typically, five valves are challenged per event. This results in an equivalent failure rate for each challenge of 0.03. In NUREG-0737, the staff stated that the challenge and failure rates can be reduced in many ways.

To resolve this issue, the staff required GE to investigate the feasibility of reducing challenges to relief valves and implement those changes shown to reduce challenges to the safety and relief valves without compromising the performance of the relief valves or other systems. Challenges to the relief valves were expected to be reduced substantially (by an order of magnitude).

GE evaluated the possible ways for reducing the challenges and failure rate of SRVs, and reduced the MSIV isolation set points from RPV level 2 to level 1.5. The staff concludes that this modification will reduce SRV challenge rates and concludes that GE's response adequately addresses the requirements of this TMI item for the ABWR design. Further discussion of SRVs is provided in Section 5.2.2 of this report.

20.5.7 10 CFR 50.34(f)(1)(vii): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Modification of ADS Logic; Feasibility Study and Modification for Increased Diversity for Some Event Sequences (TMI Item II.K.3(18))

Paragraph (1)(vii) of 10 CFR 50.34(f) requires performance of a feasibility and risk assessment study to determine the optimum ADS design and modifications that would eliminate the need for manual activation to ensure adequate core cooling for BWRs.

In NUREG-0737, the staff stated that the ADS actuation logic should be modified to eliminate the need for manual actuation to ensure adequate core cooling. The applicant was required to conduct a feasibility and risk assessment study to determine the best approach. For example, the applicant was to consider actuating the ADS on a low reactor vessel water level signal if the instruments indicate no HPCI or HPCS flow and a low-pressure ECCS is running. This logic would complement, not replace, the existing ADS actuation logic. The ADS must be manually actuated to cool the core adequately for transient and accident events that do not directly produce a high drywell pressure signal (e.g., stuck-open relief valve or steam line break outside containment), and are degraded by the loss of high-pressure ECCSs. This TMI item requires that the ADS logic be modified to eliminate operator action.

In the DFSER, the staff reported that GE proposed to modify the ABWR design in a manner consistent with option 4 of the BWROG response to TMI Item II.K.3(18). GE discussed this proposal in a letter dated October 28, 1982, from T.J. Dente (GE) to D.G. Eisenhut (NRC), "NUREG-0737 Item II.K.3.18 Modification of Automatic Depressurization System Logic." This modification involves a manual inhibit switch and a timer that bypasses the high drywell pressure permissive if the reactor water level is low for a sustained period. GE added an 8-minute high drywell pressure bypass timer to the ABWR ADS initiation logic. This timer will initiate on a low water level-1 signal. When it times out, it bypasses the need for a high drywell pressure signal to initiate the standard ADS initiation logic.

The DFSER reported that the bypass timer would be tested periodically as required by the STS and stated that this was identified in Chapter 6 of the DFSER as TS Item 6.3.3-1. It also indicated that GE documented the results of its evaluation of the adequacy of the 8-minute bypass timer in the SSAR (see GE letter to NRC, MFN No. 038-92, dated February 14, 1992). The staff agreed with the analysis and results and concluded that the modifications to the ADS design were acceptable. In response to TS Item 6.3.3-1, GE modified the ABWR TS to require periodic testing of the ADS bypass timer. The staff concludes that GE has adequately responded to this item as discussed further in Section 6.3.3 of this report.

On the basis of its review of the ADS design and resolution of the DFSER TS item, the staff concludes that GE's response adequately addresses this TMI requirement.

20.5.8 10 CFR 50.34(f)(1)(viii): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Restart of Core Spray and LPCI Systems on Low Level; Design and Modification (TMI Item II.K.3(21))

Paragraph (1)(viii) of 10 CFR 50.34(f) requires performance of a study of the effect on all core-cooling modes of BWRs under accident conditions of designing the core spray and low-pressure coolant injection systems to ensure that the systems will automatically restart on loss of water level, after having been manually stopped, if an initiation signal is still present.

In NUREG-0737, the staff identified a concern that the operator may stop the required flow of the core spray and LPCI system. These systems will not restart automatically on a loss-of-water level signal if an initiation signal is still present. The staff indicated that the vendor should modify the core spray and LPCI system logic so that these systems will restart, if required, to ensure adequate core cooling. Before modifying the design, the staff also stated that the vendor should submit a preliminary design to the staff for approval because it affects several core-cooling modes under accident conditions.

In the DFSER, the staff reported that for the ABWR, GE endorses the conclusions of the study performed by the BWROG which was forwarded to the staff in a December 29, 1980, letter from D.B. Waters (BWROG) to D.G. Eisenhut (NRC). The BWROG concluded that the current BWR ECCS design is adequate and that the proposed changes would decrease the overall safety of the plant. For example, the BWROG stated that the modification would significantly escalate the control system complexity, restrict the operator's flexibility when dealing with anticipated events, and reduce system reliability. The BWROG concluded that the current ECCS design is adequate because the BWR operator training is comprehensive and thoroughly addresses reactor-water-level control, the EOPs address this issue, the operator has sufficient time to correct errors, and the low reactor-water-level conditions are clearly displayed and alarmed in the control room.

The staff reported that it reviewed the results of the BWROG study and considered the emphasis placed on water-level control in BWR operator training. The staff found GE's response acceptable and agrees that the ABWR design need not be modified to provide an automatic restart of the low-pressure ECCS. Therefore, the staff concludes that the ABWR design meets the requirements of this TMI item.

20.5.9 10 CFR 50.34(f)(1)(ix): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Confirm Adequacy of Space Cooling for HPCI and RCIC Systems (TMI Item II.K.3(24))

Paragraph (1)(ix) of 10 CFR 50.34(f) requires a determination of the adequacy of space cooling for the high pressure coolant injection systems at BWRs. Long-term operation of these systems during a complete loss of offsite power, for up to two hours, may require space cooling to maintain the HPCI pump rooms within allowable temperature limits. The HPCI systems and their respective support systems should be designed to withstand the consequences of a complete loss of offsite ac power for 2 hours. For plants with HPCS systems in lieu of HPCI systems, the HPCS system is to be evaluated in lieu of the HPCI system.

The ABWR design contains two systems to provide room cooling to the HPCF and RCIC systems. During normal operation, the two HPCF pump rooms and the single RCIC pump room are cooled by the non-safety-related secondary containment HVAC system. During accident conditions, the room-cooling function is transferred to the safety-grade essential equipment HVAC system and the secondary containment HVAC system is isolated. The essential equipment HVAC system consists of a fan coil unit in each of the pump rooms. Cooling for each fan coil is provided by the safety-related portion of the applicable train of the RCW system. The fan coil unit in the applicable pump room, the HPCF subsystem or the RCIC system which the fan coil unit is serving, and the associated RCW train are all on the same essential electrical power division, which includes the divisional onsite ac power source, that is, emergency onsite diesel generator. The fan coil unit is necessary to keep the temperature of the associated pump room within its design limits. It is automatically initiated upon startup of the respective HPCF or RCIC pump. Based on the above, the staff concludes that space cooling for the HPCF and the RCIC systems will be available as required by this TMI item following a complete loss of offsite ac power to the plant for at least 2 hours.

In the DFSER, the staff noted that the design characteristics for the fan coil units were not specifically identified in SSAR Section 9.4.5. The staff was not able to confirm the ability of the fan coil units to remove sufficient heat to maintain the pump room temperatures within design limits. Also, SSAR Tables 8.3-1 and 8.3-2, which list diesel generator loads, did not include the fan coil units. These concerns were identified in the DFSER as Open Item 20.3-1. Subsequently, in Amendment 32 of the SSAR, GE stated in Section 9.4.5.2.2.1 that the fan coil units are sized to maintain the rooms within 40 °C (104 °F) operational temperature, which is well below the temperature limits specified in SSAR Appendix 3I regarding environmental qualification. Based on this information, the staff determined that there is sufficient margin to provide adequate space cooling during a LOCA. Further, GE added the fan coil units and their capacities to SSAR Table 9.4-4e, and their associated loads in SSAR Table 8.3-1 (the operational loads are one-half of the connected loads). Based on this additional information, the staff concludes that the fan coil units receive emergency power from the diesel generators and are sized to remove the worst-case heat load from the rooms. This additional information adequately resolved the concerns of DFSER Open Item 20.3-1.

The staff concludes that the HPCF and RCIC room cooling units are adequately sized and powered to maintain the rooms within design environmental conditions following a complete loss of offsite ac power to the plant for at least 2 hours, and therefore, the ABWR design adequately addresses the requirements of this TMI item.

20.5.10 10 CFR 50.34(f)(1)(x): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Study and Verify Qualification of Accumulators on ADS Valves (TMI Item II.K.3(28))

Paragraph (1)(x) of 10 CFR 50.34(f) requires the performance of a study to ensure that the ADS valves, accumulators, and associated equipment and instrumentation of BWRs will be capable of performing their intended functions during and following an accident situation, taking no credit for non-safety-related equipment or instrumentation, and accounting for normal expected air (or nitrogen) leakage through valves.

The ABWR has 18 quality Group A, seismic Category I SRVs, eight of which can also perform the ADS function. One 189-liter (50-gallon) capacity nitrogen accumulator with a design pressure of 1379 kPa (200 psig) is provided for each ADS SRV to support its ADS function during and following an accident situation. The accumulator supplies compressed nitrogen gas to the valve for its actuations. The staff's evaluation is based on SSAR Sections 1A.2.31 and 6.7 and is limited to the adequacy of the accumulators and associated equipment to perform their intended functions during and following an accident situation. The staff's evaluation regarding the system's compliance with other regulatory requirements, such as the adequacy of the ADS SRV valves and their associated instrumentation and controls to perform their intended functions during and following an accident situation, is provided in Sections 5.2.2, "Overpressure Protection," 6.3, "Emergency Core Cooling Systems," and 7.3, "Engineered Safety Features Systems," of this report.

Each accumulator is sized to provide sufficient nitrogen gas for one associated ADS valve actuation at drywell design pressure or five actuations at normal drywell pressure with nominal pneumatic supply. Makeup supply for the accumulators is provided by the safety-related portions of the nitrogen gas supply system, which includes two redundant safety-related nitrogen gas supply trains. Each train consists of 10 high-pressure nitrogen gas bottles and associated piping, valves, and instrumentation. The nitrogen bottles in each train have sufficient stored compressed nitrogen gas to supply makeup nitrogen gas to the accumulators associated with four ADS valves to compensate for an expected nitrogen leakage of 28 liters per hour (L/hr) (1 standard cubic foot per hour (scfh)) for each valve for a 7-day period. Additionally, the accumulators can be refilled after the system is operating using the non-safety-related portion of the nitrogen gas supply system in conjunction with the non-safety-related atmospheric control system (ACS) described in SSAR Section 6.2.5. GE determined that a maximum of three ADS valves will be needed to meet short-term demands and one ADS valve will be needed to meet long-term needs and so, one safety-related train of the nitrogen gas supply system will be sufficient to provide the needed nitrogen makeup to the associated ADS accumulators. The system has alarm provisions to indicate failure of any redundant nitrogen supply train due to a loss of nitrogen supply pressure to the ADS valve accumulator.

The accumulators, including the associated equipment to the ADS valves, and the safety-related portions of the nitrogen gas supply system are designed to seismic Category I, quality Group B or C, as appropriate, and quality assurance B requirements. The accumulators, including the associated equipment to the ADS valves, are designed to operate in the environmental conditions to be found in the drywell after a design-basis accident (see SSAR Section 7.3.1.1.1.2 for specific additional information and Section 3.11 of this report for general information on environmental qualification design criteria for equipment important to safety). SSAR Section 7.3.1.1.10 states that the safety-related equipment in the nitrogen gas supply system are selected to accommodate the hostile environment to which they may be exposed during an accident situation. The nitrogen supply system is designed to 1379 kPa (200 psig) and 66 °C (151 °F). In the DFSER, the staff expressed concern regarding the design temperature. The 66 °C (151 °F) design temperature is significantly lower than the temperature to which some portions of the system (e.g., inside the drywell and the reactor building) may be exposed during an accident situation. The staff also expressed concern that SSAR Section 6.7.4 did not explicitly specify a requirement for periodically testing the leakage through each valve to verify that such leakage is within the assumed value of 28 L/hr (1 scfh). These concerns were identified in the DFSER as Open Item 20.3-2.

Subsequently, in Amendment 31, GE modified SSAR Sections 6.7.4 and Figure 6.7-1. The figure now indicates that the high pressure nitrogen (HPIN) system piping from inside primary containment to the outboard containment isolation valve is designed to 171 °C (340 °F), while the remainder of the system outside primary containment is designed to 66 °C (151 °F). These specifications ensure that the system can withstand the worst postulated environmental conditions. SSAR Section 6.7.4 now states that periodic testing will be performed on system components to ensure that leakage will not exceed 28 L/hr (1 scfh). Based on this additional information, the staff concludes that the system is designed to ensure proper ADS SRV valve operation. This additional information adequately resolved the concerns of DFSER Open Item 20.3-2.

Based on its evaluation, the staff concludes that (1) GE's criteria for sizing each nitrogen gas accumulator and the provision for supply of makeup nitrogen gas to the accumulator is appropriate, and (2) the nitrogen accumulators and the associated equipment, including the safety-related portions of the nitrogen gas supply system, will perform their intended functions during and following an accident situation. Therefore, the ABWR design adequately addresses the requirements of this TMI item.

20.5.11 10 CFR 50.34(f)(1)(xi): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Evaluate Depressurization with Other Than Full ADS (TMI Item II.K.3(45))

Paragraph (1)(xi) of 10 CFR 50.34(f) requires provision of an evaluation of depressurization modes, other than full actuation of the ADS, such as early depressurization with one or two SRVs, that would reduce the possibility of exceeding vessel integrity limits during rapid cooldown of BWRs. Slower depressurization would reduce the possibility of exceeding vessel integrity limits by rapidly cooling down.

Since the ABWR design provides three high pressure core injection systems, the probability of ADS actuation is lower than for current BWRs. In the report dated September 27, 1982 (memorandum from T. Speis to G. Lainas), the staff evaluation which addressed this subject concurred with the BWROG study in this area. In the analyses conducted for the study, it was assumed that all high-pressure injection systems would fail, but that all low-pressure systems would operate. The time at which the operator is assumed to actuate the ADS varied. The effects of depressurization over a 10-minute interval and a 20-minute interval were compared to the case in which the RCS is completely depressurized in 3.3 minutes. Vessel fatigue usage was the key parameter studied regarding vessel integrity. The BWROG analyzed the extent to which a longer depressurization period would reduce fatigue usage and then considered the effect of this reduced fatigue usage on the core cooling capability. The BWROG concluded that

- vessel integrity limits are not exceeded for full depressurization using the ADS

- for slower depressurization rates (longer than the approximate 3.3 minute interval for the normal depressurization rate), the usage assignable to the full depressurization using the ADS would not significantly affect vessel fatigue usage relative to that usage assignable to the full ADS blowdown

- slower depressurization rates reduce the core cooling capability, except when the operator begins to depressurize the RCS very early in the accident.

The results also stated that core cooling capability could be improved using a 10-minute depressurization period if the operator actuated the ADS within 1 to 6 minutes after the accident begins. However, it was considered more prudent to activate the high-pressure injection systems during this initial period to avoid using the ADS.

GE performed the analysis for a standard BWR, which has an RPV and ADS similar to the ABWR. Thus, the RPV cooldown rate and material design of the RPV vessel for the ABWR are similar to those features in current designs. Hence, the results should be the same for the ABWR. Based on its review, the staff concludes that the current mode of depressurization using the ADS in the same manner as current operating BWRs for the ABWR is satisfactory and adequately addresses the requirements of this TMI item.

20.5.12 10 CFR 50.34(f)(1)(xii): Evaluation of Alternative Hydrogen Control Systems

Paragraph (1)(xii) of 10 CFR 50.34(f) requires performance of an evaluation of alternative hydrogen control systems that would satisfy the requirement of paragraph (2)(ix) of 50.34(f) (see Section 20.5.21 below). As a minimum, the evaluation must consider a hydrogen ignition and post-accident inerting system and include:

- A comparison of costs and benefits of the alternative systems considered.

- For the selected system, analyses and test data to verify compliance with the requirements of paragraph (2)(ix) of 50.34(f).

- For the selected system, preliminary design descriptions of equipment, function, and layout.

In the DFSER, the staff required GE to provide this information since it had not yet done so. This was identified in the DFSER as Open Item 20.3-3. Subsequently, GE modified SSAR Section 19A.2.12 to state that Section 6.2.7.1 contained COL license information requiring the applicant referencing the ABWR design to provide this information, if appropriate. The staff agrees that evaluating an alternative hydrogen control system is beyond the scope of the ABWR design certification review, and should a COL applicant wish to provide an alternative system, it should provide the supporting information for staff review. Based on the clarifying information provided in the SSAR, the staff concludes that GE has adequately addressed the requirements of 10 CFR 50.34 (f)(1)(xii). Therefore, DFSER Open Item 20.3-3 is resolved.

20.5.13 10 CFR 50.34(f)(2)(i): Operating Personnel -- Simulator Use and Development - Long-Term Training Upgrade (TMI Item I.A.4.2)

Paragraph (2)(i) of 10 CFR 50.34(f) requires the provision of simulator capability that correctly models the control room and includes the capability to simulate SBLOCAs.

GE stated that simulator facilities for use in performing operator training are beyond the scope of the ABWR design certification review and the COL applicant will be responsible for addressing this requirement. This is consistent with the treatment of training in SSAR Chapter 13 and is acceptable because training will be addressed by the COL applicant. This was identified in the DFSER as COL Action Item 18.7.2.2-1. The staff verified that GE established a COL action item (Item 18.8.8 in SSAR Section 18.8) for an operator training program to meet 10 CFR Part 50. This approach adequately addresses DFSER COL Action Item 18.7.2.2-1 as discussed in Sections 13.2 and 18.7.2.2 of this report.

20.5.14 10 CFR 50.34(f)(2)(ii): Operating Procedures - Long-Term Program Plan Procedures for Upgrading of Procedures (TMI Item I.C.9)

Paragraph (2)(ii) of 10 CFR 50.34(f) requires establishment of a program, to begin during construction and follow into operation, for integrating and expanding current efforts to improve plant procedures. The scope of the program is required to include emergency procedures, reliability analyses, HFE, crisis management, operator training, and coordination with INPO and other industry efforts.

The development of detailed procedures is beyond the scope of the ABWR design certification review and the COL applicant will be responsible for addressing this TMI item in the detailed design implementation. This was identified in the DFSER as COL Action Item 18.7.2.2-2. The staff verified that GE established a COL action item in SSAR Section 13.5.3 for procedure development. Additionally, SSAR Section 13.5.3.1 states that the methods and criteria for the development, V&V, implementation, maintenance, and revision of procedures will include considerations of I.C.9. This approach is adequately addresses DFSER COL Action Item 18.7.2.2-2 as discussed in Section 13.5 of this report.

20.5.15 10 CFR 50.34(f)(2)(iii): Control Room Design - Control Room Design Reviews (TMI Item I.D.1)

Paragraph (2)(iii) of 10 CFR 50.34(f) requires the provision, for NRC review, of a control room design that reflects state-of-the-art human factor principles prior to committing to fabrication or revision of fabricated control room panels and layouts.

GE stated that this requirement is beyond the scope of the ABWR design certification review and the COL applicant will be responsible for addressing it in the detailed design implementation. This was identified in the DFSER as COL Action Item 18.7.2.2-3. The staff verified that GE addressed this requirement further in the HFE design (SSAR Section 18.8), included detailed CR development in ABWR CDM Table 3.1 ITAAC and in SSAR Section 18E, and established a COL action item (Item 18.8.1) in SSAR Section 18.8. This approach adequately addresses DFSER COL Action Item 18.7.2.2-3 as discussed in Section 18.7.2.2 of this report.

20.5.16 10 CFR 50.34(f)(2)(iv): Control Room Design - Plant Safety Parameter Display Console (TMI Item I.D.2)

Paragraph (2)(iv) of 10 CFR 50.34(f) requires provision of a plant safety parameter display console that will display to operators a minimum set of parameters defining the safety status of the plant, capable of displaying a full range of important plant parameters and data trends on demand, and capable of indicating when process limits are being approached or exceeded.

GE addresses this requirement in SSAR Section 18.4.2.11 (evaluated in Section 18.3 of this report). This requirement is beyond the scope of the ABWR design certification review and the COL applicant will be responsible for addressing it in the SPDS design. This was identified in the DFSER as COL Action Item 18.7.2.2-4. The staff verified that GE established a COL action item in SSAR Section 18.8 (Item 18.8.4) to evaluate the SPDS design against the applicable provisions of NUREG-0737, Supplement 1. This approach adequately addresses DFSER COL Action Item 18.7.2.2-4 as discussed in Section 18.7.2.2 of this report.

It should be noted that Section 18.7.2.2 of this report states that the ABWR design will not provide a separate SPDS console, but rather, the functions of the SPDS will be integrated into the overall control room display capabilities. The staff concludes that an exemption from the requirements of 10 CFR 50.34(f)(2)(iv) for a separate SPDS console are appropriate for the ABWR for the reasons set forth in Sections 18.3 and 18.7.2.2 of this report.

20.5.17 10 CFR 50.34(f)(2)(v): Control Room Design - Safety System Status Monitoring (TMI Item I.D.3)

Paragraph (2)(v) of 10 CFR 50.34(f) requires provisions for automatic indication of the bypassed and inoperable status of safety systems.

SSAR Section 7.1.2.10.2 states that the ABWR standard plant design complies with the requirements of IEEE 279, paragraph 4.13; RG 1.47; and BTP ICSB 21. IEEE 279, paragraph 4.13 and RG 1.47 state that if the protective action of some part of the I&C system has been bypassed or deliberately rendered inoperative for any purpose, this fact will be continuously indicated in the control room. The ABWR design also provides for automatic indication at the system level when the system loses power or when it is out of service. In addition, a switch will be provided for manual initiation of bypass indication for out-of-service conditions under limited circumstances when bypass is not automatically annunciated. On this basis, the staff concludes that the ABWR standard design meets the stated guidance of RG 1.47, and therefore, meets the requirements of 50.34(f)(2)(v) with respect to the I&C design of safety system status monitoring. See also Section 7.2.3 of this report.

The human factors details of this requirement are beyond the scope of the ABWR design certification review and GE stated that they will be addressed by the COL applicant in the detailed design implementation. This was identified in the DFSER as COL Action Item 18.7.2.2-5. The staff verified that GE established a COL action item in SSAR Section 18.8 (Item 18.8.1) to conduct the detailed HFE design according to design and implementation as defined by the ABWR CDM Table 3.1 ITAAC and SSAR Appendix 18E. The staff considers this to include the resolution of the human factors aspects of this requirement. This approach adequately addresses DFSER COL Action Item 18.7.2.2-5 as discussed in Section 18.7.2.2 of this report.

20.5.18 10 CFR 50.34(f)(2)(vi): Consideration of Degraded or Melted Cores in Safety Review - Reactor Coolant System Vents (TMI Item II.B.1)

Paragraph (2)(vi) of 10 CFR 50.34(f) requires that each applicant and licensee install high-point vents for the RCS and the reactor vessel head. These vents will be remotely operated from the control room. Although the purpose of the system is to vent noncondensable gases from the RCS which may inhibit core cooling during natural circulation, the vents must not lead to an unacceptable increase in the probability of a LOCA or a challenge to containment integrity. These vents are part of the reactor coolant pressure boundary and thus, shall conform to the regulatory requirements of 10 CFR Part 50. The vent system shall be designed with sufficient redundancy to ensure a low probability of inadvertent or irreversible actuation.

This TMI item also requires the submittal of the following information on the design and operation of the high-point vent system:

- A description of the design, location, size, and power supply for the vent system, along with results of analyses for LOCAs initiated by a break in the vent pipe. The results of the analyses should demonstrate compliance with the acceptance criteria of 10 CFR 50.46.

- Procedures and supporting analysis for the operator to use in operating the vents. This information should include the information available to the operator for beginning or ending the use of the vents.

In the ABWR design, the 18 power-operated safety and relief valves are the primary venting capability. Each SRV is seismically qualified. The high-pressure nitrogen gas supply to the eight SRVs which comprise the ADS is also seismically qualified. To vent the RCS, each SRV can be operated in the power-actuated mode by remote manual controls in the main control room. The discharge line for each SRV will include linear variable differential transformers (LVDTs) and a thermocouple to monitor the position and leakage of the SRV. Each SRV discharges to the suppression pool.

The RCS can also be vented through the RCIC system, which directs steam from one of the main steam lines to a turbine-driven pump, from which the steam is exhausted to the suppression pool. The RCIC system can vent the RCS during hot standby mode or during reactor isolation.

The top head vent line of the RPV can also direct steam and noncondensable gases from the reactor's upper dome. This line is used principally to vent the reactor during the final stages of normal shutdown from power operation. A reactor head vent line is a continuous vent which is normally open to discharge to a main steam line.

The COL applicant will develop plant-specific procedures to govern the operator's use of the relief mode for venting the reactor. This was identified in the DFSER as COL Action Item 20.3-1. In the advance SER, the staff concluded that since GE had not yet included a COL action item in the SSAR addressing the development of these procedures, DFSER COL Action Item 20.3-1 would remain open until GE had done so. In Amendment 33, GE revised SSAR Sections 1A.2.5 and 1A.3.6 to establish a COL action item for the COL applicant to develop the indicated plant procedures. This approach adequately addresses DFSER COL Action Item 20.3-1.

GE has submitted no additional accident analyses to address a break in any of the vent lines because the plant's design basis includes a complete steam line break, which is more bounding.

The staff concurs with the applicant's assessment because it includes adequate capacity, operation, and procedural provisions of the ABWR vent system. The staff concludes that GE has adequately addressed the requirements of this TMI item for the ABWR design.

20.5.19 10 CFR 50.34(f)(2)(vii): Consideration of Degraded or Melted Cores in Safety Review - Plant Shielding to Provide Access to Vital Areas and Protect Safety Equipment for Post-Accident Operation (TMI Item II.B.2)

Paragraph (2)(vii) of 10 CFR 50.34(f) requires radiation and shielding design reviews of spaces around systems that may, as a result of an accident, contain source term radioactive materials, and design as necessary to permit adequate access to important areas and to protect safety equipment from the radiation environment. GE's response adequately addresses the requirements of this TMI item for the ABWR design as discussed in Sections 12.3.6 and 13.6.3.5 of this report.

20.5.20 10 CFR 50.34(f)(2)(viii): Consideration of Degraded or Melted Cores in Safety Review - Post-Accident Sampling (TMI Item II.B.3)

Paragraph (2)(viii) of 10 CFR 50.34(f) requires the capability to promptly obtain and analyze samples from the reactor coolant system and containment that may contain TID-14844 source term radioactive materials without radiation exposure to any individual exceeding 0.05 Sv (5 rem) to the whole-body or 0.50 Sv (50 rem) to the extremities.

GE's response adequately addresses the requirements of this TMI item for the ABWR design as discussed in Section 9.3.2.2 of this report.

20.5.21 10 CFR 50.34(f)(2)(ix): Consideration of Degraded or Melted Cores in Safety Review - Rulemaking Proceeding on Degraded Core Accidents (TMI Item II.B.8), "Hydrogen Control System"

Paragraph (2)(ix) of 10 CFR 50.34(f) requires a system for hydrogen control that can safely accommodate hydrogen generated by the equivalent of a 100-percent, fuel clad metal-water reaction. The hydrogen control system and associated systems shall provide with reasonable assurance that:

(A) Uniformly distributed hydrogen concentrations in the containment do not exceed 10 percent during and following an accident that releases an equivalent amount of hydrogen as would be generated from a 100-percent, fuel clad metal-water reaction, or that the post-accident atmosphere will not support hydrogen combustion.

(B) Combustible concentrations of hydrogen will not collect in areas where unintended combustion or detonation could cause loss of containment integrity or loss of appropriate mitigating features.

(C) Equipment necessary for achieving and maintaining safe shutdown of the plant and maintaining containment integrity will perform its safety function during and after being exposed to the environmental conditions attendant with the release of hydrogen generated by the equivalent of a 100-percent, fuel clad metal-water reaction including the environmental conditions created by activation of the hydrogen control system.

(D) If the method chosen for hydrogen control is a post-accident inerting system, inadvertent actuation of the system can be safely accommodated during plant operation.

SSAR Section 6.2.5 discusses the provisions for combustible gas control within the ABWR containment, which include an inerted containment and inclusion of thermal hydrogen recombiners. In Sections 6.2.5 and 19.2.3.3.1 of this report, the staff concludes that these provisions for hydrogen gas control within the ABWR containment, for both design-basis accidents and severe accidents, are acceptable.

Criteria (A) and (B) are met through an inerted containment which ensures that the post-accident containment atmosphere will not support hydrogen combustion. Criterion (C) is met as discussed in Section 19.2.3.3.7 of this report. Criterion (D) is not applicable to the ABWR design, as inerting is accomplished prior to the onset of the accident. The staff concludes that the ABWR design meets the requirements of 10 CFR 50.34(f)(2)(ix).

20.5.22 10 CFR 50.34(f)(2)(x): Reactor Coolant System Relief and Safety Valves - Testing Requirements (TMI Item II.D.1)

Paragraph (2)(x) of 10 CFR 50.34(f) requires that the licensee provide a test program and associated model development and conduct tests to qualify reactor coolant system relief and safety valves for all fluid conditions expected under operating conditions, transients, and accidents. The staff concludes that GE's approach adequately addresses the requirements of this TMI item for the ABWR design as discussed in Section 3.9.3.2 of this report.

20.5.23 10 CFR 50.34(f)(2)(xi): Reactor Coolant System Relief and Safety Valves - Relief and Safety Valve Position Indication (TMI Item II.D.3)

Paragraph (2)(xi) of 10 CFR 50.34(f) requires that the relief and safety valves for the RCS shall include a positive indication in the control room derived from a reliable valve-position detection device or a reliable indication of flow in the discharge pipe.

The ABWR SRVs are equipped with position sensors that will be qualified as Class 1E components. All SRV positions will be indicated in control room. The ABWR design also includes a backup method using thermocouples. The pipe downstream of the SRV is equipped with thermocouples which signal the annunciator and the plant process computer when the temperature in the tail pipe exceeds the predetermined set point.

The staff has reviewed the design, compared it with the position and clarification contained in NUREG-0737, "Clarification of TMI Action Plan Requirements," and determined that GE's response adequately addresses the requirements of this TMI item for the ABWR design as discussed in Section 5.2.2 of this report.

20.5.24 10 CFR 50.34(f)(2)(xii): System Design -- Auxiliary Feedwater System - Auxiliary Feedwater System Automatic Initiation and Flow Indication (TMI Item II.E.1.2)

Paragraph (2)(xii) of 10 CFR 50.34(f) requires the provision of automatic and manual AFWS initiation and the provision of AFWS flow indication in the control room of PWR plants.

This requirement is applicable to PWRs only, since a BWR design does not include an AFWS. Therefore, it is not technically relevant to the ABWR design and does not need to be addressed.

20.5.25 10 CFR 50.34(f)(2)(xiii): System Design -- Decay Heat Removal - Reliability of Power Supplies for Natural Circulation (TMI Item II.E.3.1)

Paragraph (2)(xiii) of 10 CFR 50.34(f) requires provision of a pressurizer heater power supply and associated motive and control power interfaces for PWR plants that is sufficient to establish and maintain natural circulation in hot standby conditions with only onsite power available.

This requirement is applicable to PWRs only, since a BWR design does not include a pressurizer. Therefore, it is not technically relevant to the ABWR design and does not need to be addressed.

20.5.26 10 CFR 50.34(f)(2)(xiv): System Design -- Containment Design - Isolation Dependability (TMI Item II.E.4.2)

Paragraph (2)(xiv) of 10 CFR 50.34(f) requires the provision of containment isolation systems that do the following:

- Ensure all non-essential systems are isolated automatically by the containment isolation signal
- Have two isolation barriers in series for each non-essential penetration, except instrument lines
- Do not result in reopening of the containment isolation valves on resetting of the isolation signal
- Utilize a containment set point pressure for initiating containment isolation as low as is compatible with normal operation

- Include automatic closing on a high radiation signal for all systems that provide a path to the environs.

The staff reviewed GE's containment isolation system for compliance with these requirements. The evaluation results and findings regarding each requirement are as follows:

- All non-essential systems are isolated automatically by the containment isolation system in accordance with the TMI Item II.E.4.2 requirements.

- As discussed in DFSER Sections 6.2.4 and 6.2.4.1, some penetrations do not have two isolation barriers in series that conform to the containment isolation requirements of GDC 56. This was identified in the DFSER as Open Item 20.3-4. (This issue was also identified in the DFSER as Open Item 6.2.4.1-1.) As discussed in Section 6.2.4.1 of this report, GE subsequently provided adequate justification for the alternative containment isolation design that resolved these open items. Therefore, DFSER Open Item 20.3-4 and DFSER Open Item 6.2.4.1-1 are resolved.

- Resetting the containment isolation signal will not result in the automatic reopening of containment isolation valves. The reopening of any containment isolation valve is on a valve-by-valve basis once the isolation signal has cleared and following subsequent logic reset.

- GE has committed to using a high drywell set point pressure of 14 kPag (2 psig) to isolate non-essential penetrations. This is a minimum value and compatible with normal operating conditions. The staff finds a set point value of 14 kPag (2 psig) acceptable.

- The containment purge and vent isolation valves isolate on high radiation levels in the reactor building HVAC air exhaust or in the fuel handling area HVAC air exhaust. As discussed in SSAR Section 6.2.4.1, the containment purge provision has not been found acceptable for ABWR.

Based on the above evaluation, the staff concludes that the ABWR design meets this TMI requirement.

20.5.27 10 CFR 50.34(f)(2)(xv): System Design – Containment Design - Purging (TMI Item II.E.4.4)

Paragraph (2)(xv) of 10 CFR 50.34(f) requires provision of containment venting/purging capability to minimize purging time, consistent with the as-low-as-is-reasonably-achievable (ALARA) principles for occupational exposure. It also requires assurance that the purge system will reliably isolate under accident conditions.

SSAR Section 19A.2.27 stated that during normal power operation, all large valves in the containment ventilation lines are closed and that only the 5-cm (2-in.) nitrogen makeup valves are open. These valves are characterized as air-operated valves with fast closure times which prevent substantial releases from containment should containment isolation be required. GE also stated that the 5-cm (2-in.) nitrogen bleed lines are sufficient to maintain normal containment pressure during normal operation when used in conjunction with containment spray and the drywell cooling system. In the DFSER the staff found that the use of the 5-cm (2-in.) nitrogen bleed lines for normal pressure control is consistent with ALARA considerations and there is high assurance that the purge system will reliably isolate under accident conditions.

SSAR Section 19A.2.27 also stated that the large ventilation valves will be tested not only on a regular basis, but also after any valve maintenance in order to ensure that closing times are within allowable limits. In the DFSER, the staff noted that these tests should include valve T31-F007, that the details of these tests should be submitted by the applicant referencing the ABWR design, and that this should be identified in the SSAR as a COL action item. This was identified in the DFSER as COL Action Item 20.3-2. Subsequently, GE modified SSAR Table 3.9-8, "Inservice Testing of Safety-Related Pumps and Valves", to include both T31-F007 and T31-F010, the isolation valves for the COPS rupture disk. The staff verified that GE established a COL action item to develop the inservice testing program for the components in Table 3.9-8. The test requirements in this table ensure that these valves will be adequately tested. This approach adequately addresses DFSER COL Action Item 20.3-2.

In response to the staff's request for additional information (RAI) dated June 5, 1990, GE stated that the isolation signal to valve T31-F007 would be deleted from SSAR Figure 6.2-39a. As of SSAR Amendment 11, this figure still contained an isolation signal to this valve, the DFSER identified this as Open Item 20.3-5. Subsequently, GE provided SSAR Figure 6.2-39, Sheet 1, which deleted the isolation signal. Therefore, DFSER Open Item 20.3-5 is resolved.

In another response to this RAI, GE agreed to amend the ABWR TS to allow a 24-hour (rather than a 72-hour) window at the beginning and end of a fuel cycle, during which the large diameter 56-cm (22-in.) purge lines can be open in accordance with the STS. This was identified in the DFSER as TS Item 6.2.5-1 and was discussed in Chapter 6 of the DFSER. In addition to proper operation during normal conditions, the staff stated in the DFSER that GE should provide justification that these valves will close during accident conditions. This was identified in the DFSER as Open Item 20.3-6. The staff has confirmed that SSAR Section 6.2.5 has been modified to clarify that these valves receive close signals during accident conditions. Therefore, DFSER Open Item 20.3-6 is resolved.

Based on the additional clarifying information provided in the SSAR, the staff concludes that the ABWR design meets this TMI requirement.

20.5.28 10 CFR 50.34(f)(2)(xvi): System Design -- Design Sensitivity of B&W Reactors - Design Evaluation (TMI Item II.E.5.1)

Paragraph (2)(xiv) of 10 CFR 50.34(f) requires establishment of a design criterion for the allowable number of actuation cycles of the ECCS and RPS of B&W plant designs consistent with the expected occurrence rates of severe overcooling events, considering both anticipated transients and accidents.

This requirement is applicable to B&W-designed plants only, therefore, it is not technically relevant to the ABWR design and does not need to be addressed.

20.5.29 10 CFR 50.34(f)(2)(xvii): Instrumentation and Controls - Additional Accident Monitoring Instrumentation (TMI Item II.F.1)

Paragraph (2)(xvii) of 10 CFR 50.34(f) requires provisions for instrumentation to measure, record, and read out in the control room containment pressure, water level, hydrogen concentration, radiation intensity (high level), and noble gas effluent at all potential accident release points. In addition, it requires continuous sampling of radioactive iodines and particulates in gaseous effluent from all potential accident release points, and an onsite capability to analyze and measure samples. Under TMI Item II.F.1, NUREG-0660 restates these requirements with additional guidance and clarification. NUREG-0660 calls for a human factors analysis, which is to include the use of the indicators listed above by the operator during both normal and abnormal plant conditions, integration of these indicators into plant emergency procedures and operator training, the use of other alarms during an emergency, and the need for prioritization of alarms.

In the DFSER, the staff stated that GE's responses to its RAI dated June 5, 1990, were still under review and that completion of this review was DFSER Open Item 20.3-7. SSAR Section 7.5 compares the ABWR design against the criteria of RG 1.97 (Rev. 3), addressing accident monitoring instrumentation. Section 7.5 lists the variables that are considered essential safety-related information for the operators, and identifies specific exceptions to the guidance of RG 1.97. The capability to monitor the parameters required by the regulation is provided in the control room. Based on its review of SSAR Section 7.5, the staff concludes that the ABWR I&C design meets RG 1.97 as discussed further in Section 7.5.2 of this report and, therefore, also meets this TMI requirement. Therefore, DFSER Open Item 20.3-7 is resolved.

In the DFSER, the staff also reported that GE stated that the human factors aspects of this requirement were beyond the scope of the ABWR design certification review and the COL applicant will be responsible for addressing them in the detailed design implementation. This was identified in the DFSER as COL Action Item 18.7.2.2-3. The staff verified that GE established a COL action item (Item 18.8.1) in SSAR Section 18.8 for the detailed control room development as defined in ABWR CDM Table 3.1 ITAAC and in SSAR Section 18E. Further, GE established a COL action item (Item 18.8.13) in SSAR Section 18.8 to address II.F.1. This approach adequately addresses DFSER COL Action Item 18.7.2.2-3 as discussed in Section 18.7.2.2 of this report.

20.5.30 10 CFR 50.34(f)(2)(xviii): Instrumentation and Controls - Identification of and Recovery from Conditions Leading to Inadequate Core Cooling (TMI Item II.F.2)

Paragraph (2)(xviii) requires provision of instrumentation or controls (primary or backup) proposed for the plant to supplement existing instrumentation (including primary coolant saturation monitors) in order to provide an unambiguous, easy-to-interpret indication of inadequate core cooling (ICC). It also requires a description of the functional design requirements for the system, a description of the procedures to be used with the proposed equipment, the analysis used in developing these procedures, and a schedule for installing the equipment.

Reactor pressure vessel level is the only issue to be considered with respect to this issue since BWRs operate at saturation pressure and saturation monitors are not required for the ABWR. As in operating BWRs, the level instruments for the ABWR RPV are all delta p (dp) instruments. Each instrument uses a reference leg, which is maintained full by a condensing chamber connected directly to the steam space in the RPV, and uses a variable leg which is connected to the RPV water space. All the dp level instruments operate on the same physical principle. Therefore, common-cause failures caused by a design deficiency or maintenance error could result in inaccurate indication of reactor vessel water level.

The staff's concern about common-cause failures is based on experience with potential common-mode failure mechanisms in the reactor water level instruments. For example, during the past two years, anomalies have been observed in reactor vessel water level indication at several BWRs (Millstone-1, Pilgrim, LaSalle, and Washington Nuclear (WNP-2) during controlled depressurization to commence plant outages. These anomalies consisted of "spiking" or "notching" of level indication, and in one instance, a sustained error in level indication. The effect of noncondensable gas in the condensate chamber and reference leg of the cold-leg type of water level instruments has been determined to be the root cause of these level indication anomalies. Testing has shown that under depressurization conditions, noncondensable gases can cause significant errors in the level indication.

The ABWR design for the reactor vessel water level measurement system has the following features:

(1) The ABWR has temperature-compensated RPV level indication for post-accident monitoring of the RPV.

(2) The vertical drop in the drywell for the ABWR RPV water level reference leg instrument lines from the condensing chamber to the drywell wall has been limited to three feet.

(3) The ABWR uses analog level transmitters.

(4) The ABWR safety system utilizes two-out-of-four logic for the automatic safety systems initiated on RPV water level.

The staff reviewed the ABWR design for the reactor vessel water level measurement system and found that it meets the requirements specified in GL 84-23, "Reactor Vessel Water Level Instrumentation in BWRs," dated October 26, 1984. To ensure high functional reliability of the instrumentation, the staff issued GL 92-04, "Resolution of the Issues Related to Reactor Vessel Water Level Instrumentation in BWRs Pursuant to 10 CFR 50.54(f)," dated August 19, 1992, and Bulletin 93-03, "Resolution of Issues Related to Reactor Vessel Water Level Instrumentation in BWRs," dated May 28, 1993, requesting hardware modifications for operating reactors. In response to these generic communications, all BWR licensees have committed to implement hardware modifications to their level instrumentation systems. Similar cold-leg instruments are used in the ABWR design.

Before the noncondensible gas level inaccuracies, there were other problems with dp level instrumentation used in BWRs. The staff issued GL 84-23 to address concerns related to high containment temperature during a depressurization event. High containment temperature combined with reactor depressurization can lead to false water level readings as a result of flashing or boiling in the reference leg within the containment.

In the DFSER, the staff stated that the design of the ABWR reactor vessel water level measurement system met the requirements specified in GL 84-23. However, the staff required GE to discuss GL 92-04 in the SSAR and include any design changes necessary to preclude the potential for false reactor coolant level readings. The staff also required GE to determine if compliance with this TMI requirement was affected by any design changes. This was identified in the DFSER as Open Item 20.3-8. In response to these generic communications, GE changed the design of the ABWR RPV level instrumentation system. GE incorporated a backfill modification system that will constantly purge the reference leg with a very low flow rate of water supplied by the CRD system. The constant flow of water up the reference leg will prevent dissolved gases from migrating down the reference leg.

The known common-mode deficiencies in BWR level instrumentation systems have been addressed at operating BWRs and by GE in the ABWR design. It should also be noted that these particular deficiencies would not have compromised the automatic protective functions of the level instrumentation for accident scenarios initiated while at power, and that no previous incidents at BWRs of inaccurate level indication have been misinterpreted by plant operators so as to lead to unsafe actions. The staff concluded in the advance SER that the ABWR level instrumentation system without the proposed level diversity meets the requirements of 10 CFR Part 50. However, in view of the importance of level instrumentation for safety in BWRs, and the experience discussed above where the potential existed to fail redundant level instruments due to a common cause, the staff believes that the addition of level instrumentation which operates on a diverse physical principle is desirable and prudent for the purpose of guiding operator emergency actions.

GE did not agree with the staff recommendation for diverse water level instrumentation and presented its position in a letter dated October 26, 1993. As part of the letter, GE presented the following summary:

> ABWR water level instrumentation is rugged, simple and highly redundant for failure tolerance. All known operating problems have been addressed in this design and it is incredible to postulate simultaneous common-mode failures which would yield identical errors in all the dp instrumentation.

> Alternate technologies are unqualified for this application; further, there is no need to add this complexity, since the plant operating staff has ample additional indications of an impending problem without relying solely on water level. The EPGs direct the operator to use all information available to him and make conservative (safe) decisions.

In the attachment to the letter, GE also provided a list of indications of inadequate RPV water level which are independent of the dp RPV water level instrumentation. The staff recognizes that other parameters could aid the operator in assessing the adequacy of core cooling under accident conditions. These include instrumentation for indication of reactor power, core neutron flux, the recirculation flow control system response, and feedwater flow and steam flow mismatch. However, the staff believes that these indications could be easily misinterpreted or could be insufficient because they are only indirect methods of inferring reactor water level or core cooling.

Other evolutionary designs, such as the ABB-Combustion Engineering, Inc. (ABB-CE) System 80+, provide diverse methods of RPV level measurement. The inadequate core cooling instrumentation package in the CE System 80+ plant includes reactor vessel level monitoring system probes employing both dp sensors and the heated junction thermocouple concept. The staff is aware of a diverse method of level monitoring that is currently in use in at least one nuclear power plant in Germany employing ultrasonic measurement techniques. In addition, a diverse level measurement system which uses heated junction thermocouples has been in use for the past 5 years at a Swedish BWR, and another Swedish BWR uses float switches for diverse level indication and automatic systems actuation. Other Swedish BWRs have decided in principle to install diverse level measurement systems.

The staff indicated in the advance SER that the diverse method of level measurement is recommended for indication in the control room only (there is diverse instrumentation, namely high drywell pressure, in both the operating BWRs and the ABWR design which provides diverse signals for automatic safety systems actuation for many event scenarios). This would provide a direct and back-up means for the operator to identify inadequate core cooling and to take appropriate manual actions to initiate and control safety systems as identified in the plant EOPs. The staff also recommended that the diverse level measurement device be reliable, redundant, and capable of being powered by onsite power sources. This was Open Item F20.5.30-1 in the advance SER.

The staff issued the draft Commission paper, "Diversity in the Method of Measuring Reactor Pressure Vessel Level in the Advanced Boiling Water Reactor and Simplified Boiling Water Reactor," on November 15, 1993, for public and industry comments. The ACRS discussed the issue in its 430th meeting on December 9 through 11, 1993, and sent its recommendation to the Commission in a letter dated December 16, 1993. The ACRS did not support the staff recommendation on diversity. Based on ACRS deliberations and GE's position, the staff reconsidered the need for the requirement for instrumentation diversity.

All the known common-mode deficiencies in BWR level instrumentation systems have been addressed by GE in the ABWR design. It should also be noted that these deficiencies would not have compromised the automatic functions of the level instrumentation for accident scenarios initiated while at power, and that no previous incidents at BWRs of inaccurate level indication have been misinterpreted by plant operators so as to lead to unsafe actions. In addition, for many events, the ECCS is started in the ABWR on high drywell pressure, as well as low reactor water level, thus providing some diversity. The ABWR EPGs will be used to develop the EOP that will be used with the reactor water level instrumentation.

Even though it may be desirable to provide instrumentation diversity in the ABWR design, there is not sufficient basis to postulate an unidentified potential common-mode failure. Further, diverse level measurement devices have not been demonstrated to be adequate. In light of the enhanced LOCA response in the ABWR and the guidance provided in the ABWR EPGs to address the use of the RPV instrumentation, the staff concludes that reactor vessel level instrumentation diversity is not required for the ABWR. On the basis of the above discussion, Open Item F20.5.30-1 (DFSER Open Item 20.3-8) is resolved. Based on the above discussion, the staff also concludes that the ABWR RPV level instrumentation will provide adequate indication of ICC as required by this TMI item.

In the DFSER the staff reported that GE stated that the human factors aspects of this requirement were beyond the scope of the ABWR design certification review and the COL applicant will be responsible for addressing them in the detailed design implementation. This was identified in the DFSER as COL Action Item 18.7.2.2-3. The staff verified that GE established a COL action item (Item 18.8.1) in SSAR Section 18.8 for the detailed control room development as defined in ABWR CDM Table 3.1 ITAAC and in SSAR Section 18E. Further, GE established a COL action item (Item 18.8.14) in SSAR

Section 18.8 to address II.F.2. This approach adequately addresses DFSER COL Action Item 18.7.2.2-3 as discussed in Section 18.7.2.2 of this report.

20.5.31 10 CFR 50.34(f)(2)(xix): Instrumentation and Controls - Instruments for Monitoring Accident Conditions (TMI Item II.F.3)

Paragraph (2)(xix) of 10 CFR 50.34(f) requires instrumentation adequate for monitoring plant conditions following an accident that includes core damage.

SSAR Section 7.5 compares the ABWR design against the criteria of RG 1.97 (Rev. 3), addressing accident monitoring instrumentation. Section 7.5 lists the variables that are considered essential safety-related information for the operators, and identifies specific exceptions to the guidance of RG 1.97. The list incorporates adequate monitoring capability for post-accident plant conditions that include core damage, including reactor pressure, water level and temperature, containment pressure, temperature and radiation level, and shutdown operation status. Based on its review of SSAR Section 7.5, the staff concludes that the ABWR I&C design meets RG 1.97 as discussed further in Section 7.5.2 of this report and, therefore, also meets this TMI requirement.

20.5.32 10 CFR 50.34(f)(2)(xx): Electrical Power - Power Supplies for Pressurizer Relief Valves, Block Valves, and Level Indicators (TMI Item II.G.1)

Paragraph (2)(xx) of 10 CFR 50.34(f) requires power supplies for pressurizer relief valves, block valves, and level indicators for PWRs such that the level indicators are powered from vital buses, motive and control power connections to the emergency power sources are thorough devices qualified in accordance with requirements applicable to systems important to safety, and electric power is provided from emergency power sources.

This requirement is applicable to PWRs only, therefore, it is not technically relevant to the ABWR design and does not need to be addressed.

20.5.33 10 CFR 50.34(f)(2)(xxi): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- IE Bulletins - Describe Automatic and Manual Actions for Proper Functioning of Auxiliary Heat Removal Systems When Feedwater System Not Operable (TMI Item Issue II.K.1.(22))

Paragraph (2)(xxi) of 10 CFR 50.34(f) requires design of auxiliary heat removal systems such that necessary automatic and manual actions can be taken to ensure proper functioning when the main feedwater system is not operable.

SSAR Section 7.7.1.4(9) states that if the ABWR main feedwater system is not operating, the reactor will be tripped automatically when reactor water level falls to reactor vessel level 3. The operator can manually initiate the RCIC system from the main control room or, if the operator takes no action, reactor water level will continue to decrease as the steam boils off until the low-low level set point (level 2) is reached. At low-low level, the RCIC system is automatically initiated to supply makeup water to the reactor pressure vessel. If level 1.5 is reached, both HPCF pumps will start automatically. These systems will continue automatic injection until the reactor water level reaches level 8, at which time the HPCF and RCIC systems are tripped. The HPCF will restart automatically once the high-level trip signal clears and the level 1.5 signal is received. The RCIC will automatically start after a level 8 trip and a level 2 signal.

If the vessel is isolated, reactor vessel pressure is regulated by automatic or remote manual operation of the main steam relief valves, which exhaust to the suppression pool. In this case, the automatic suppression pool cooling mode of the RHR system will transfer heat to the ultimate heat sink.

The HPCF system will automatically provide the required makeup flow for the accident situations with the reactor vessel at high pressure. No manual operations are required since the HPCF system will cycle on and off automatically as water level reaches vessel level 1.5 and level 8, respectively. If the HPCF system fails under these conditions, the operator can manually depressurize the reactor vessel using the ADS to permit the low-pressure emergency core cooling systems to provide makeup coolant. The RCS will automatically be depressurized if all of the following signals are present: high drywell pressure, level 1 water level, and pressure in at least one low-pressure injection system or one HPCF system. If the low level persists in the RPV, the ADS will also activate without drywell high pressure after a delay of about 8 minutes.

The ABWR design incorporates appropriate automatic and manual action capability to ensure proper heat removal when the main feedwater system is not operable. Therefore, the staff concludes that GE has adequately addressed the requirements of this TMI item for the ABWR design.

20.5.34 10 CFR 50.34(f)(2)(xxii): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Commission Orders on B&W Plants - Analysis and Upgrading of Integrated Control System (TMI Item II.K.2(9))

Paragraph (2)(xxii) of 10 CFR 50.34(f) requires performance of a failure modes and effects analysis of the integrated control system (ICS) of B&W-designed plants to include consideration of failures and effects of input and output signals to the ICS.

This requirement is applicable to B&W-designed plants only, therefore, it is not technically relevant to the ABWR design and does not need to be addressed.

20.5.35 10 CFR 50.34(f)(2)(xxiii): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Commission Orders on B&W Plants - Hard-Wired Safety-Grade Anticipatory Reactor Trips (TMI Item II.K.2(10))

Paragraph (2)(xxiii) of 10 CFR 50.34(f) requires provision, as part of the RPS of B&W-designed plants, an anticipatory reactor trip that would be actuated on loss of main feedwater and on turbine trip.

This requirement is applicable to B&W-designed plants only, therefore, it is not technically relevant to the ABWR design and does not need to be addressed.

20.5.36 10 CFR 50.34(f)(2)(xxiv): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Central Water Level Recording (TMI Item II.K.3(23))

Paragraph (2)(xxiv) of 10 CFR 50.34(f) requires the capability to record reactor vessel water level in one location on recorders that meet normal post-accident recording requirements for BWRs.

SSAR Section 7.5.8 states that the reactor vessel water level wide range instruments and fuel zone instruments are utilized to provide post-accident monitoring indication. The four divisions of wide range level instruments cover the zone from above the core to the main steam lines. The two channels of fuel zone instruments cover the range from below the core to the top of the steam separator shroud.

The SSAR also states that in the event that the vessel water level is below the range of the wide range level sensor and the two channels of fuel zone level instrumentation disagree, the EOPs instruct the operator to use the lower level indicated in the two channels and to return the water level to the range of the wide range level instrumentation. GE adds that by using the four divisions of wide range level instruments, an unambiguous indication of vessel water level can be determined despite a postulated failure of a single instrument channel or division, thereby permitting the operator to implement the EOPs.

The staff concludes that GE's approach acceptably meets the requirements of this TMI item regarding post-accident reactor vessel water level recording for the ABWR design.

20.5.37 10 CFR 50.34(f)(2)(xxv): Emergency Preparedness and Radiation Effects -- Improve Licensee Emergency Preparedness - Short Term; Upgrade Emergency Preparedness (TMI Item III.A.1.2)

Paragraph (2)(xxv) of 10 CFR 50.34(f) requires the provision of an onsite Technical Support Center (TSC), an Operational Support Center (OSC), and a nearsite Emergency Operations Facility (EOF).

GE's proposal is discussed and evaluated in Section 13.3 of this report. The staff verified that GE established a COL action item in SSAR Section 19A.3.4 to detail a TSC and an OSC in the ABWR service building. This SSAR section also states that the COL applicant has an interface requirement to provide a near site emergency operational facility. On the basis of the staff's evaluation in Section 13.3 of this report and GE's establishment of the COL action item, the staff concludes that GE has adequately addressed the requirements of this TMI item for the ABWR design.

20.5.38 10 CFR 50.34(f)(2)(xxvi): Radiation Protection -- Radiation Source Control - Primary Coolant Sources Outside the Containment Structure (TMI Item III.D.1.1)

Paragraph (2)(xxvi) of 10 CFR 50.34(f) requires provision of leakage control and detection in the design of systems outside containment that contain (or might contain) TID-14844 source term radioactive materials following an accident. Applicants are required to submit a leakage control program, including an initial test program, a schedule for retesting these systems, and the actions to be taken for minimizing leakage from such systems. The goal is to minimize potential exposures to workers and the public and to provide reasonable assurance that excessive leakage will not prevent the use of systems needed in an emergency.

The advance SER stated that the staff reviewed SSAR Section 1A.2.34, as well as GE's response (see page 20.3.15-22 of the SSAR) to staff Question 430.227 relating to this TMI item. It also stated that these references identify the applicable systems and the leak reduction measures for the ABWR design. Based on its review, the staff found in the advance SER that the leakage control program for the systems outside the containment for the ABWR design would include periodic leak testing and leak reduction measures for eight listed systems. GE revised and expanded the list of systems in Section 1A.2.34 to include the following: (1) RHR, (2) HPCF, (3) low pressure core flooder, (4) RCIC, (5) suppression pool cleanup, (6) reactor water cleanup, (7) fuel pool cooling and cleanup, (8) post-accident sampling, (9) process sampling, (10) containment atmospheric monitoring, (11) fission product monitor (part of leakage detection system), (12) hydrogen recombiner, and (13) standby gas treatment.

In the advance SER, the staff also found that the COL applicant will be required to develop plant procedures that will prescribe the leak testing methods for the above systems and schedule maintenance programs to monitor leakages and reduce detected leakages to lowest practical levels. GE stated that the leak tests will be conducted periodically at each refueling outage. Furthermore, the staff found that GE identified Item 5.5.2.2 in the administrative controls section of the ABWR TS, which calls for the COL applicant to establish a leakage control program for the subject systems. The TS item further stipulates that the program include preventive maintenance and periodic visual inspection requirements and integrated leak test requirements for each system at refueling cycle intervals or less.

Based on the above, the staff concludes that the ABWR design adequately addresses the requirements of this TMI item for the ABWR design.

The staff will review and evaluate the individual leakage control programs, including the initial test programs, of the COL applicant on a plant-specific basis.

20.5.39 10 CFR 50.34(f)(2)(xxvii): Radiation Protection -- Worker Radiation Protection Improvement - Inplant Radiation Monitoring (TMI Item III.D.3.3)

Paragraph (2)(xxvii) of 10 CFR 50.34(f) requires provisions for monitoring of inplant radiation and airborne radioactivity as appropriate for a broad range of routine and accident conditions.

GE's response adequately addresses the requirements of this TMI item for the ABWR design as discussed further in Section 12.5.1 of this report.

20.5.40 10 CFR 50.34(f)(2)(xxviii): Radiation Protection -- Worker Radiation Protection Improvement - Control Room Habitability (TMI Item III.D.3.4)

Paragraph (2)(xxviii) of 10 CFR 50.34(f) requires an evaluation of the potential pathways for radioactivity and radiation that may lead to control room habitability problems under accident conditions resulting in a TID-14844 source term release, and to make necessary design provisions to preclude such problems.

The staff's review of control room habitability is provided in Section 6.4 of this report. The staff concludes that the ABWR control room design provides an acceptable means of maintaining the control room in a safe and habitable condition by providing adequate protection under accident conditions in accordance with the requirements of this TMI item.

20.5.41 10 CFR 50.34(f)(3)(i): Operating Procedures - Procedures for Feedback of Operating Experience to Plant Staff (TMI Item I.C.5)

Paragraph (3)(i) of 10 CFR 50.34(f) requires the provisions for administrative procedures for evaluating operating, design, and construction experience and for ensuring that applicable important industry experiences will be provided in a timely manner to those designing and constructing the plant.

The development of detailed procedures is beyond the scope of the ABWR design certification review and the COL applicant will be responsible for addressing this TMI item. The staff verified that GE established a COL action item in SSAR Section 13.5.3 for procedure development and a COL action item in SSAR Section 13.2 regarding the incorporation of operating experience into training programs. Additionally, SSAR Section 13.5.3.1 states that the methods and criteria for the development, V&V, implementation, maintenance, and revision of procedures will include considerations of I.C.5. This approach is acceptable to the staff as discussed in Sections 13.5 and 13.2 of this report.

20.5.42 10 CFR 50.34(f)(3)(ii): Quality Assurance (QA) - Expand QA List (TMI Item I.F.1)

NUREG-0660 states that several systems important to safety of the TMI plant were not designed, fabricated, and maintained at a level equivalent to their safety importance. In accordance with the requirements of Criterion 1 of Appendix A to 10 CFR Part 50 and the guidance provided in SRP Section 17.3, "Quality Assurance Program Description," applicants are to include these types of non-safety-related items within the scope of their QA programs. This requirement was incorporated into Paragraph (3)(ii) of 10 CFR 50.34(f).

SSAR Table 3.2-1, "Classification Summary," identifies safety-related and non-safety-related items of the ABWR. Note "e" in the table and SSAR Section 3.1.2.1.1.2 state

(1) the total QA program described in SSAR Chapter 17 is applied to the safety-related items

(2) the non-safety-related items are controlled by the QA program described in SSAR Chapter 17 in accordance with the functional importance of the item

The staff concludes that GE's approach adequately addresses the requirements of this TMI item for the ABWR design.

20.5.43 10 CFR 50.34(f)(3)(iii): Quality Assurance - Develop More Detailed QA Criteria (TMI Item I.F.2)

Paragraph (3)(iii) of 10 CFR 50.34(f) requires the establishment of a quality assurance program. NUREG-0660, Item I.F.2 lists 11 considerations that the NRC should use to develop additional guidance to clarify requirements for the QA function. These considerations resulted in Revision 2 of SRP Section 17.1, "Quality Assurance During the Design and Construction Phases," in July 1981.

SSAR Chapter 17 references NEDO-11209, "Nuclear Energy Business Operation Quality Assurance Program Description," Revision 7 dated May 1987. The staff reviewed this report against the acceptance criteria of Revision 2 of SRP Section 17.1 and found it to be acceptable. The staff concludes that GE's approach adequately addresses the requirements of this TMI item for the ABWR design.

20.5.44 10 CFR 50.34(f)(3)(iv): Consideration of Degraded or Melted Cores in Safety Review - Rulemaking Proceeding on Degraded Core Accidents (TMI Item II.B.8), ".91-Meter (3-Foot) Diameter Equivalent Dedicated Containment Penetration"

Paragraph (3)(iv) of 10 CFR 50.34(f) requires one or more dedicated containment penetrations, equivalent in size to a single .91-m (3-ft) diameter opening, in order not to preclude future installation of systems to prevent containment failure, such as a filtered vented containment system. This requirement is intended to ensure provision of a containment vent design feature with sufficient safety margin well ahead of a need that may be perceived in the future to mitigate the consequences of a severe accident situation. The staff's evaluation of ABWR compliance with the requirement is limited to the effective penetration size for venting provided in the ABWR primary containment design.

In the DFSER, the staff found that the size of the primary containment penetration that could be used during a severe accident situation for venting the containment was smaller than the specific size identified in the TMI requirement. The staff required GE to submit a request for exemption from the requirement and supporting justification. The justification was expected to demonstrate that the penetration size was adequate to permit a vent relief path that is capable of providing the needed overpressure relief for the primary containment to prevent its uncontrolled failure during any credible severe accident situation. This was identified in the DFSER as Open Item 20.3-9.

SSAR Section 19A.2.44 states that the COPS precludes the need for a dedicated penetration equivalent in size to a single .91-m (3-ft) diameter opening. The COPS is part of the atmospheric control system and is discussed in SSAR Section 6.2.5.6. The COPS consists of two 200-mm (8-in.) diameter rupture disks mounted in series in a 250-mm (10-in.) line and is sized to allow 35 kg/sec (15.86 lbm/sec) of steam flow at the opening pressure of 6.3 kg/cm^{2}g (90 psig), which corresponds to an energy flow of about 2.4 percent of rated power. The SSAR states that the COPS is capable of keeping containment pressures below ASME Service Level C limits for an ATWS event with failure of the SLCS and containment heat removal systems.

In Section 19.2.3.3.4 of this report, the staff concludes that the COPS design is acceptable. Although the diameter of the COPS pathway is only 200 mm (8 in.), the staff determined that this exception from the requirement of a .91-m (3-ft) diameter opening is acceptable since: (1) the limiting diameter of the COPS pathway is adequate to permit the needed vent relief path, and (2) a need for venting capability beyond that provided by the COPS has not been identified. This resolved DFSER Open Item 20.3-9. The staff concludes that GE's approach adequately addresses the requirements of this TMI item for the ABWR design. An exemption in accordance with 10 CFR 50.12(a)(2)(ii) is justified since the COPS provides sufficient venting capability to preclude the need for a .91 m (3-ft) diameter equivalent dedicated containment penetration. This exemption is in order based on the staff's conclusion that the underlying purpose of the regulation has been met.

20.5.45 10 CFR 50.34(f)(3)(v): Consideration of Degraded or Melted Cores in Safety Review - Rulemaking Proceeding on Degraded Core Accidents (TMI Item II.B.8), "Containment Integrity During an Accident Involving 100-Percent, Fuel Clad Metal-Water Reaction"

Paragraph (3)(v) of 10 CFR 50.34(f) requires design information to demonstrate that:

(1) Containment integrity will be maintained during an accident that releases hydrogen generated from a 100-percent, fuel clad metal-water reaction accompanied by either hydrogen burning or the added pressure from post-accident inerting, assuming carbon dioxide is the inerting agent. Systems necessary to ensure containment integrity shall also be demonstrated to perform their function under these conditions.

(2) Containment structure loadings produced by an inadvertent full actuation of a post-accident inerting hydrogen control system, but not including seismic or design-basis accident loadings will not produce stresses in steel containments in excess of the limits set forth in the ASME Code.

SSAR Section 19E.2.3.2 provides an evaluation of the capability of the containment to withstand pressurization from a 100-percent, fuel clad metal-water reaction. The staff finds the evaluation acceptable in Section 19.2.3.3.1 of this report. The staff evaluation of GE's assessment of systems necessary to ensure containment integrity under these conditions is provided in Section 19.2.3.3.7 of this report. Criterion (2) is not applicable to the ABWR design, as inerting is accomplished prior to the onset of the accident. On this basis, the staff concludes that GE's approach adequately addresses the requirements of this TMI item for the ABWR design.

20.5.46 10 CFR 50.34(f)(3)(vi): System Design -- Containment Design - Dedicated Penetrations (TMI Item II.E.4.1)

Paragraph (3)(vi) of 10 CFR 50.34(f) requires redundant, dedicated containment penetrations for plant designs with external hydrogen recombiners so that, assuming a single active failure, the recombiner systems can be connected to the containment atmosphere.

SSAR Section 1A.2.13 states that the flammability control system (FCS) uses two permanently installed recombiners located in secondary containment which ensure that the FCS remains operable assuming a single active failure. In the DFSER, the staff noted that the FCS was not described in SSAR Section 6.2.5 and that GE had not demonstrated that redundant dedicated containment penetrations existed for the hydrogen recombiners. The staff required GE to provide

- information to clearly demonstrate that the permanently installed hydrogen recombiners have redundant, dedicated containment penetrations and that the penetrations meet all applicable design requirements. This information was to include
  - how long after a LOCA and at what hydrogen concentration the recombiners are to be utilized
  - line sizes as related to flow requirements
  - duration of recombiner operation
  - interface requirements for referencing applicants with regard to the recombiners
- a clearer copy of SSAR Figure 6.2-40, "Flammability Control System."

This was identified in the DFSER as Open Item 20.3-10. GE provided a new, clearer copy of Figure 6.2-40 and modified SSAR Section 6.2.5 to clarify FCS operation. Based on this additional information, the staff concludes that the FCS can adequately accommodate the effects of hydrogen and mitigate the consequences of hydrogen generated as a result of a LOCA, assuming a single active component failure. Therefore, Open Item 20.3-10 is resolved. The staff concludes that the ABWR design meets the requirements of this TMI item.

20.5.47 10 CFR 50.34(f)(3)(vii): General Implications of TMI for Design and Construction Activities — Management for Design and Construction - Organization and Staffing to Oversee Design and Construction (TMI Item II.J.3.1)

Paragraph (3)(vii) of 10 CFR 50.34(f) requires provision of a description of the management plan for design and construction activities that includes:

- the organizational and management structure singularly responsible for direction of design and construction of the proposed plant
- technical resources directed by the applicant
- details of the interaction of design and construction within the applicant's organization and the manner by which the applicant will ensure close integration of the architect engineer and the nuclear steam supply vendor
- proposed procedures for handling the transition to operation
- the degree of top-level management oversight and technical control to be exercised by the applicant during design and construction, including the preparation and implementation of procedures necessary to guide the effort.

The development of the management plan for organization and staffing to oversee design and construction is beyond the scope of the ABWR design certification review and the COL applicant will be responsible for addressing it. The staff verified that GE established a COL action item in SSAR Section 19A.3.7 to develop the necessary management plan. This approach is acceptable to the staff.

20.6 Generic Communications

As part of its program to disseminate information on operating experience to the industry, the NRC issues generic communications when a significant safety-related event or condition at one facility is believed to potentially apply to other facilities. Using the basic criteria of safety significance and generic implications, many safety issues have been highlighted in generic communications. These generic communications encompass and address both staff positions (in the form of bulletins and most generic letters) and information alerts (in the form of information notices, circulars, and some generic letters). Potential concerns addressed initially by these generic communications may be subsequently revised or amplified. The resolution of these concerns may be incorporated into formal regulatory requirements, such as rules in 10 CFR, or an analysis of such a concern may result in it becoming a USI or GSI.

The staff reviewed the ABWR design for incorporation of important lessons learned from operating experience using NRC bulletins and generic letters. These two classes of documents are used to communicate staff positions on issues potentially affecting operating facilities, thereby ensuring consideration of those issues judged to have significant public health and safety implications. The issues covered by bulletins and generic letters originate in a number of ways, including the staff's systematic review of operating experience. In the context of the NRC program to review and incorporate operating experience, bulletins and generic letters convey the most safety-significant lessons distilled from numerous sources of information. As a contrast, another product of that program, information notices, do not contain any requests for action on the part of licensees. Thus, bulletins and generic letters comprise a sufficient basis for reviewing the ABWR design against operating experience.

In SSAR Section 1.8, GE identifies the experience information that has been or will be included in the design of the ABWR. Experience information is routinely made available and distributed to design personnel. In addition, as a focused effort for the ABWR design activities, GE management surveyed a listing of all regulatory reports in its possession that contain operating experience information. If GE determined a report did not apply to the ABWR, the report was set aside. Each of the remaining reports on the list of potentially applicable experiences was then reviewed individually to determine technical applicability. As a result of this effort, GE prepared a listing of applicable regulatory documents (that is, information notices, generic letters, bulletins, and NUREGs) issued in 1980 or later. Some documents that addressed resolutions of concerns were not included on the resulting summary list to avoid repetition. In addition to these regulatory documents which are publicly available, GE has a collection of in-house proprietary documents that it has prepared over the years as part of its continual and ongoing assistance to various BWR licensees. GE also considered pertinent information from these experience reports.

Although the SRP provides acceptance criteria (some of which are based upon operating reactor experience) for review of a reactor facility design, this document was last revised in 1981. Therefore, potential concerns addressed in generic communications issued subsequent to the issue period of the SRP must be addressed. To ensure the staff's operating experience review is comprehensive, operating experience documents issued in 1980 and later were considered by the staff for applicability to the ABWR design. Because the review of operating experience should include those items that are believed to have the highest safety significance, the staff limited its augmented review to those documents that rose to the status of a bulletin or a generic letter with a staff position or a request for licensee action. The last bulletin and generic letter considered in the review were NRC Bulletin 91-01 (original issuance) and GL 91-17, respectively. This consists of approximately 450 documents. Approximately 220 of these documents were excluded because they were not pertinent to the design review of the ABWR. The remaining list of 230 documents was augmented by any additional bulletins and generic letters GE thought applicable. About 160 on the combined listing were related to GSIs, USIs, TMI item implementation, or an NRC regulation. Operating experience concerns related to these issues are required to be explicitly addressed by GE and acceptability of the ABWR design with respect to these concerns is discussed in appropriate sections of this report. Thus, the staff excluded the disposition of these concerns from the task of ensuring and addressing how operating experience was considered in the ABWR design.

The remaining 71 applicable documents identify undesirable situations that have occurred and should be avoided and that could affect ABWR issues related to either equipment design, analytical methods, construction, operation or maintenance activities, or major programmatic activities. The staff reviewed these documents to determine if additional action is necessary to ensure that this experience has been reflected in the ABWR design and properly resolved. Of the 71 documents considered by the staff, 31 addressed issues that are being resolved during the ongoing preparation of TS or during a future equipment procurement process, and 27 had been superseded by technical developments after the document was issued. The remaining 13 documents raised questions that were considered by the staff in their overall evaluation process of the ABWR design.

Based on the above, the staff concludes that GE has adequately considered operating experience identified by generic letters or bulletins issued since 1980 in the ABWR design.

21 REPORT OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

The advanced boiling water reactor (ABWR) Subcommittee of the Advisory Committee on Reactor Safeguards (ACRS) conducted a review of GE Nuclear Energy's (GE's) application for design certification, which began in February 1988 and was completed in April 1994. The Subcommittee conducted a limited review of GE's certified design material (CDM) and a complete review of the various draft versions of the staff's safety evaluation reports on the ABWR standard design. The ACRS conducted its final meeting on the ABWR during its 408th meeting on April 7 and 8, 1994, and subsequently issued its letter regarding the ABWR on April 14, 1994. This letter, which follows this discussion, reflects approval of the application and includes no recommended actions by either the staff or the applicant. During the full committee meeting held on March 10 through 12, 1994, the Chairman of the ABWR Subcommittee indicated that GE had provided him with an extensive set of draft revisions and markups to the standard safety analysis report (SSAR) and CDM in response to ACRS concerns, which were found to be acceptable but had not yet been incorporated into an SSAR amendment or final CDM submittal. The staff agreed to review GE's SSAR Amendment 34 and the associated CDM revision to ensure that the proposed revisions were included in GE's submittals. Accordingly, a review of Amendments 34 and 35, and the associated CDM was conducted and the staff determined that GE has adequately included the technical and editorial changes in its application for design certification in response to the previous ABWR Subcommittee concerns.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

April 14, 1994

The Honorable Ivan Selin
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Selin:

SUBJECT: REPORT ON SAFETY ASPECTS OF THE GENERAL ELECTRIC NUCLEAR ENERGY APPLICATION FOR CERTIFICATION OF THE ADVANCED BOILING WATER REACTOR DESIGN

During the 408th meeting of the Advisory Committee on Reactor Safeguards, April 7-8, 1994, we completed our review of the General Electric Nuclear Energy (GENE) application for certification of its U.S. version of the Advanced Boiling Water Reactor (ABWR) standard design. This final report is intended to fulfill the requirement of 10 CFR 52.53 that the ACRS "... report on those portions of the application which concern safety." During our review we had the benefit of discussions with representatives of GENE and the NRC staff. We also had the benefit of the documents referenced.

ABWR Application

The U.S. version of the ABWR standard design utilizes a significant portion of the detailed design information developed jointly by GENE, Hitachi, and Toshiba for the international version which is being built in Japan. The application for certification of the U.S. version was filed by GENE in September 1987 under the provisions of Appendix O to 10 CFR Part 50 and the NRC Policy Statement on Nuclear Power Plant Standardization (Ref. 1). The application was docketed in February 1988. In December 1991, GENE requested that the application be considered under 10 CFR 52.45. This request was made effective in March 1992.

The application is based on the ABWR Standard Safety Analysis Report (SSAR), which was submitted in modular form between September 1987 and March 1989. Since then it has been amended frequently, the last submittal for our review was Amendment 34 in March 1994. The application also includes the ABWR Certified Design Material (CDM). The CDM contains the design information from the SSAR that will become a part of the design certification rule. The CDM has been revised, the last submittal that we received was Rev. 2 in December 1993.

ABWR Design Description

The ABWR is a forced circulation boiling water reactor with a rated power of 3926 MWt. The reactor core consists of 872 8x8 fuel assemblies and 205 control rods. The reactor utilizes internal recirculation pumps and fine-motion control rod drives. It is located inside a steel-lined reinforced concrete pressure suppression containment which is enclosed by a reinforced concrete secondary containment, both of which are located in the Reactor Building. The Reactor Building also houses a standby gas treatment system, refueling area, main steam pipe tunnel, and essential systems for emergency core cooling, AC power (including diesel generators), and environmental conditioning.

The Control Building is located between the Reactor Building and the Turbine Building. The Control Building houses a continuation of the main steam pipe tunnel, the main control room, a computer facility, and essential systems for DC power, environmental conditioning, and cooling water. During emergencies, technical support is provided by the Technical Support and Operational Support Centers, which are located in the Service Building, which is immediately adjacent to the Control Building.

The Turbine Building houses equipment for power generation. Steam is supplied to an 1800 rpm turbine-generator which is oriented to minimize damage to safety-related equipment should a turbine failure occur. The Turbine Building also houses systems and equipment that provide various nonessential services for the plant. These include the standby combustion-gas-turbine generator, house boiler, air compressors, and systems for AC and DC power and environmental conditioning.

The Radwaste Building houses equipment for the collection and processing of radioactive waste generated by the plant. An underground pipe tunnel connects the Turbine and Reactor Buildings to the Radwaste Building.

The ABWR design includes a number of features that we believe will enhance safety relative to past BWR designs. Some of these features resulted from the use of PRA methodology by GENE in evaluating the ABWR design as it progressed.

- The use of reactor internal pumps removes the large reactor recirculation piping and connections to the reactor vessel, thereby reducing the size of the largest loss-of-coolant accident (LOCA).

- The use of a fine-motion control rod drive arrangement removes the scram discharge volume and associated piping, provides two reliable means for inserting the rods, and is intended to eliminate the rod drop and rod ejection accidents.

- The Emergency Core Cooling System and supporting auxiliaries are arranged into three physically separated electrical and mechanical divisions, only one of which is needed for handling transients and virtually all accidents.

- A combustion-gas-turbine generator is provided for enhanced on-site AC power capability.

- An AC-independent reactor water addition feature, a depressurization system, lower drywell flooder, cavity floor spreading area, sacrificial layer of basaltic concrete, and containment overpressure protection system are provided to mitigate severe accidents.

- The greatly increased application of digital control systems offers the potential for improved operator interface with the plant and the reliability of control and protection systems. In addition, the use of digital multiplexers and fiber optics reduces the amount of cabling in the plant thereby reducing the fire hazard.

- The reactor vessel is fabricated using ring forgings that eliminate the need for beltline longitudinal welds. This, in combination with improved material specifications, reduces concern for reactor vessel integrity.

Chronology of ACRS Review

Our review of the ABWR application commenced after it was docketed in February 1988. The NRC staff issued a Draft Safety Evaluation Report (DSER) on the first module of the SSAR in August 1989 (Ref. 2). We reviewed this draft and reported our findings in November (Ref. 3). At that time we questioned, in particular, the adequacy of the level of design detail available for review and recommended that the staff revisit the issue of what constitutes an "essentially complete" design.

Subsequent to November 1989, our review activities focused on several ABWR-related design concerns including Control Building flooding, physical separation, environmental protection of sensitive equipment, performance of essential chilled water systems, use of leak-before-break methodology, use of integral low-pressure turbine rotors, and the capability of the floor area beneath the reactor vessel to cope with severe accidents. These preliminary concerns were brought to the attention of the NRC staff in our July 1991 report (Ref. 4).

During 1991 the DSER was completed by the NRC staff in the form of six SECY papers (SECY-91-153, 235, 294, 309, 320, and 355). These papers generally covered most sections of the SSAR through the first eighteen amendments, but contained numerous open items. We reported our findings in April 1992 (Ref. 5). In this report, we reconfirmed the preliminary concerns expressed in our July 1991 report and added several more including adequacy of the PRA, containment hydrodynamic loads, Reactor Water Cleanup System safety implications, plant design life and aging management, station grounding and surge protection, and corrosion control for structures.

In October 1992, the NRC staff issued a Draft Final Safety Evaluation Report (DFSER) (Ref. 6) covering the entire SSAR through Amendment 20. This draft superseded the six SECY papers. The final version of the staff safety evaluation report which we reviewed was the "Advance Copy of Safety Evaluation Report related to the certification of the Advanced Boiling-Water Reactor Design," dated December 1993 (Ref. 7). This copy covered the NRC staff review of SSAR information through about Amendment 32. Additional changes, including those which reflect Amendments 33 and 34, were reviewed by us as page changes to Reference 7.

Between February 1988 when the ABWR application was docketed and April 1992 when we issued our report on the DSER, our ABWR subcommittee held numerous meetings to review the SSAR and the NRC staff safety evaluations. During this same period, our subcommittee on Improved Light Water Reactors held several meetings to review the Electric Power Research Institute (EPRI) Utility Requirements Document (URD) and associated NRC staff safety evaluations for the Advanced Light-Water Reactor (ALWR) evolutionary plant. (The EPRI URD prescribes ALWR design requirements from the utility industry perspective.) Meetings were also held by our subcommittees on Auxiliary and Secondary Systems, Computers in Nuclear Power Plant Operations, Human Factors, and Severe Accidents. These subcommittees reviewed a number of specialized aspects of the proposed ABWR design including those related to fire, digital control and protection systems, human factors, and severe accidents.

Between April 1992 and today, our ABWR subcommittee held additional meetings to review design features proposed beyond Amendment 20 of the SSAR and to review the DFSER and Reference 7. This review covered significant design changes in the SSAR (through Amendment 34) and closure of all open items in the DFSER. It also included a review of written responses by GENE to numerous questions and concerns raised by the subcommittee.

During this time our subcommittee on Improved Light Water Reactors held several meetings to complete its review of the EPRI URD. In addition, ABWR-related meetings were held by our subcommittees on Auxiliary and Secondary Systems, Computers in Nuclear Power Plant Operations, Human Factors, Severe Accidents, Safeguards and Security, and our Ad Hoc Subcommittee on Design Acceptance Criteria (DAC). We did not review most of the CDM portion of the application because we were assured by the NRC staff that it did not contain design features and requirements beyond those found in the SSAR. We did, however, review and comment (Ref. 8) on the viability of the DAC process as a suitable method for establishing future design acceptance requirements in certain areas (i.e., human factors engineering, radiation protection, piping design, and instrumentation and control). We also reviewed the CDM related to these DAC areas.

During our review of the ABWR SSAR, we considered the design-specific requirements which relate to the various evolutionary and advanced light water reactor policy, technical, and licensing issues included in SECY-90-016 (Ref. 9) and its successor, SECY-93-087 (Ref. 10). These issues incorporate staff positions that deviate from or are not embodied in current regulations. Their resolutions will become "applicable regulations" through incorporation into the design certification rule for the ABWR. We have commented previously (Refs. 11 and 12) concerning these issues.

ACRS Conclusion Concerning ABWR Safety

Based on the results of our review of those portions of the GENE ABWR application which concern safety, we believe that acceptable bases and requirements have been established in the application to assure that the U.S. version of the ABWR standard design can be used to engineer and construct plants that with reasonable assurance can be operated without undue risk to the health and safety of the public.

Additional comments by ACRS Members Carlyle Michelson and Charles J. Wylie are presented below.

Sincerely,

J. Ernest Wilkins, Jr.

J. Ernest Wilkins, Jr.
Chairman

Additional Comments by ACRS Members Carlyle Michelson and Charles J. Wylie

Although the Committee has arrived at a favorable conclusion concerning ABWR safety with which we agree, it is our view that this report should discuss the resolution of various issues that were considered by the Committee (Refs. 4 and 5) prior to reaching the favorable conclusion. Some of the resolutions were based on findings that were unanticipated and led to significant design changes. We believe that these findings should be made available to those who must make the final safety and design certification decisions.

As an example, it was found that the rupture of an 8-inch pipe in the non-safety-grade Reactor Water Cleanup (CUW) System which is housed inside of secondary containment creates serious environmental disruption throughout the three separate divisional areas of secondary containment which house redundant portions of the Emergency Core Cooling System (ECCS). Since this 8-inch pipe contains reactor coolant at operating temperature and pressure, the break results in an immediate loss of reactor coolant until isolated and it requires an ECCS response. Steam from the break permeates the entire secondary containment because the divisional barrier doors are forced open by a buildup of steam pressure. This occurs before the primary containment isolation valves for the CUW system have time to close. A similar situation exists for the Reactor Core Isolation Cooling (RCIC) System; however, the resulting environmental conditions for most locations are bounded by those produced by the CUW 8-inch pipe break.

Since these pipe break events cannot be confined, GENE now proposes that safety-related equipment inside of the ABWR secondary containment be environmentally qualified for steam at 15 psig. and about 248°F. It is our view that this is an acceptable, although undesirable, alternative to a design which provides separation barriers and pressure relieving pathways that are capable of isolating a sufficient amount of ECCS equipment from the harsh environment. In addition, GENE has added a third break isolation valve in the 8-inch CUW supply line and located it inside of primary containment. This valve can be closed after the blowdown is over to ensure the interruption of any prolonged loss of ECCS water to secondary containment. It is needed only if both primary containment isolation valves fail to fully close due to the severe blowdown loads or other challenges common to both valves. The added environmental qualification and the third valve are new features.

<u>References</u>:

1. U.S. Nuclear Regulatory Commission, Policy Statement, 10 CFR Part 50, "Nuclear Power Plant Standardization," 52 FR 34884, September 15, 1987
2. Letter dated August 17, 1989, from Charles L. Miller, NRC Office of Nuclear Reactor Regulation, to Patrick W. Marriott, General Electric Company, enclosing Draft Safety Evaluation Report Related to the Final Design Approval and Design Certification of the Advanced Boiling Water Reactor, August 1989
3. ACRS report dated November 24, 1989, from Forrest J. Remick, ACRS Chairman, to James M. Taylor, NRC Executive Director for Operations, Subject: Module I of the Draft Safety Evaluation Report for the Advanced Boiling Water Reactor Design

4. ACRS report dated July 18, 1991, from David A Ward, ACRS Chairman, to James M. Taylor, NRC Executive Director for Operations, Subject: Concerns Related to the General Electric Advanced Boiling Water Reactor Design
5. ACRS report dated April 13, 1992, from David A. Ward, ACRS Chairman, to James M. Taylor, NRC Executive Director for Operations, Subject: Review of the Draft Safety Evaluation Reports on the GE Advanced Boiling Water Reactor Design
6. U. S. Nuclear Regulatory Commission, NUREG-1469, "Draft Final Safety Evaluation Report Related to the Design Certification of the General Electric Nuclear Energy Advanced Boiling Water Reactor," October 1992
7. U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, "Advance Copy of Safety Evaluation Report related to the certification of the Advanced Boiling-Water Reactor Design," December 1993
8. ACRS report dated January 14, 1994, from J. Ernest Wilkins, Jr., ACRS Chairman, to Ivan Selin, NRC Chairman, Subject: Final Report on the Use of the Design Acceptance Criteria Process in the Certification of the General Electric Nuclear Energy Advanced Boiling Water Reactor Design Approval
9. SECY-90-016, dated January 12, 1990, from James M. Taylor, NRC Executive Director for Operations, for the Commissioners, Subject: Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements
10. SECY-93-087, dated April 2, 1993, from James M. Taylor, NRC Executive Director for Operations, for the Commissioners, Subject: Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs
11. ACRS report dated April 26, 1990, from Carlyle Michelson, ACRS Chairman, to Kenneth M. Carr, NRC Chairman, Subject: Evolutionary Light Water Reactor Certification Issues and Their Relationship to Current Regulatory Requirements.
12. ACRS report dated April 26, 1993, from Paul Shewmon, ACRS Chairman, to Ivan Selin, NRC Chairman, Subject: SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs"

22 CONCLUSIONS

The staff performed its review of the U.S. ABWR standard safety analysis report, certified design material, and technical specifications in accordance with the standards for review of design certification applications set forth in 10 CFR §52.48 that are applicable and technically relevant to the U.S. ABWR standard design, including the exemptions and applicable regulations identified in Section 1.6 of this report. On the basis of its evaluation and independent analyses as discussed in this report, the staff concludes that, subject to satisfactory resolution of the confirmatory items identified in Section 1.8 of this report, GE Nuclear Energy's application for design certification meets the requirements of 10 CFR §52.47 that are applicable and technically relevant to the U.S. ABWR standard design. A copy of the report by the Advisory Committee on Reactor Safeguards required by 10 CFR §52.53 is provided in Chapter 21 of this report.

The staff also concludes that issuance of a final design approval, in accordance with Appendix O to 10 CFR Part 52, will not be inimical to the common defense and security or to the health and safety of the public. The financial qualifications of the applicable utility and the indemnity requirements of 10 CFR Part 140 will be addressed during the plant-specific licensing process for an application that references the U.S. ABWR standard design.

A final design approval, issued on the basis of this SER, does not constitute a commitment to issue a permit or license, or in any way affect the authority of the Commission, the Atomic Safety and Licensing Board, and other presiding officers, in any proceeding pursuant to Subpart G of 10 CFR Part 2.

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10. SUPPLEMENTARY NOTES

Docket Nos. 52-001 and STN 50-605; Project No. 671

11. ABSTRACT (200 words or less)

This Safety Evaluation Report (SER) documents the technical review of the U.S. Advanced Boiling Water Reactor (ABWR) standard design by the U.S. Nuclear Regulatory Commission (NRC) staff. The application for the ABWR design was submitted by GE Nuclear Energy. The NRC staff concludes that, subject to satisfactory resolution of the confirmatory items identified in Section 1.8 of this SER, GE's application for design certification meets the requirements of Subpart B of 10 CFR Part 52 that are applicable and technically relevant to the U.S. ABWR standard design.

12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)

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