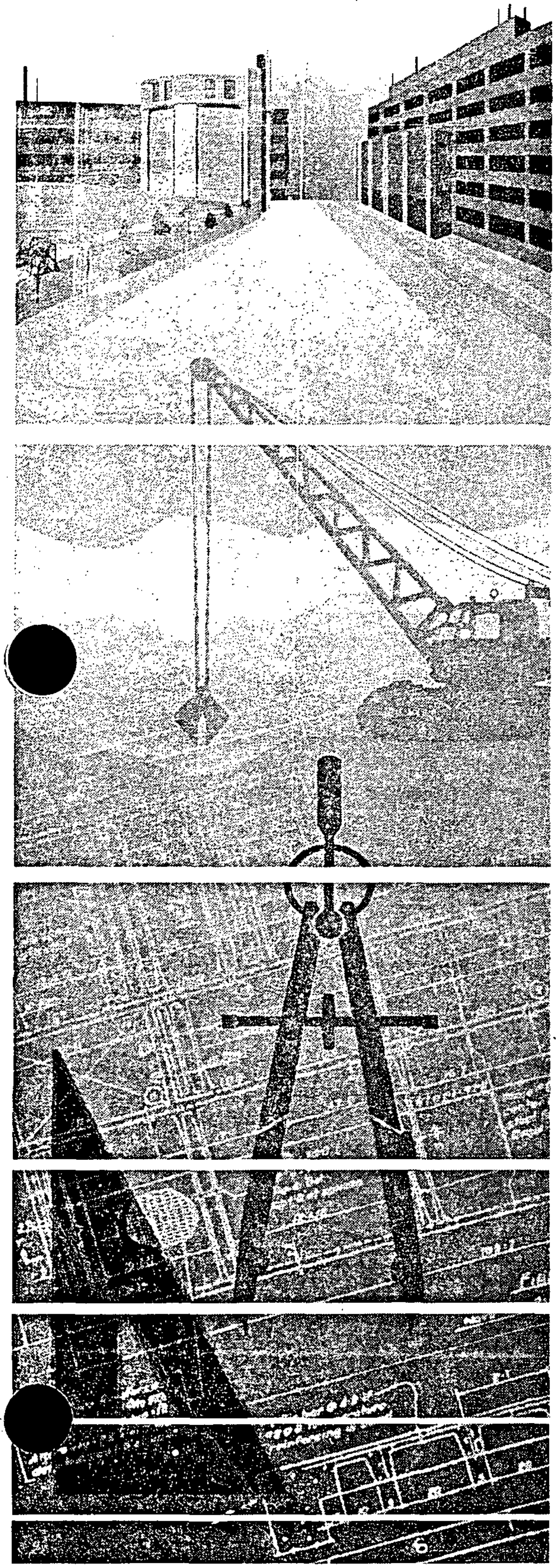

Final Safety Evaluation Report

Related to the Certification of the Advanced Boiling Water Reactor Design

Main Report

U.S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation

July 1994

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Final Safety Evaluation Report Related to the Certification of the Advanced Boiling Water Reactor Design

Main Report

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Associate Directorate for Advanced Reactors and License Renewal
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

ABSTRACT

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 initially submitted by the General Electric Company, now GE Nuclear Energy (GE), in accordance with the procedures of Appendix O of Part 50 of Title 10 of the Code of Federal Regulations (10 CFR Part 50). Later GE requested that its application be considered as an application for design approval and subsequent design certification pursuant to 10 CFR § 52.45.

The U.S. ABWR design is similar to the international ABWR design, which was being built at the Kashiwazaki Kariwa Nuclear Power Generation Station, at the time of the staff's review, by the Tokyo Electric Power Company, Inc. The ABWR is a single-cycle, forced-circulation, boiling water reactor (BWR) with a rated power of 3926 megawatts thermal (MWt) and a design power of 4005 MWt. Many features of the ABWR design are similar to those of BWR designs that the staff had previously approved. To the extent feasible and appropriate, the staff relied on earlier reviews for those ABWR design features that are substantially the same as those previously considered. The SERs for the other BWR designs have been published and are available for public inspection at the NRC Public Document Room, 2120 L Street, N.W., Washington, D.C. 20037. Unique features of the ABWR design include internal recirculation pumps, fine-motion control rod drives, microprocessor-based digital logic and control systems, and digital safety systems.

On the basis of its evaluation and independent analyses, 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. A copy of the report by the Advisory Committee on Reactor Safeguards required by 10 CFR § 52.53 is provided in Chapter 21. 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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1 INTRODUCTION AND GENERAL DISCUSSION

1.1 Introduction

On September 29, 1987, the General Electric Company applied for certification of the U.S. advanced boiling water reactor (ABWR) standard design with the U.S. Nuclear Regulatory Commission (NRC) (hereinafter referred to as the NRC, the Commission, or the staff). The application was made in accordance with the procedures of Appendix O to Part 50 of Title 10 of the Code of Federal Regulations (10 CFR Part 50) and the Policy Statement on Nuclear Power Plant Standardization, dated September 15, 1987. The application was docketed on February 22, 1988 (Docket No. STN 50-605). On December 20, 1991, GE Nuclear Energy (GE) (hereinafter referred to as GE or the applicant) requested that its application be considered as an application for design approval and subsequent design certification pursuant to 10 CFR 52.45. Accordingly, the staff assigned a new docket number (52-001), which became effective on March 13, 1992.

The NRC's licensing project managers that are currently assigned to the U.S. ABWR design review are Thomas H. Boyce, Son Q. Ninh, and David T. Tang. They may be reached by calling (301) 415-7000 or by writing to the Office of Nuclear Reactor Regulation, Mail Stop O-11 H3, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555.

GE's application, the ABWR standard safety analysis report (SSAR), describing the design of the facility, was originally submitted in modular form between September 1987 and March 1989. Subsequently, GE supplemented the information in the SSAR through an amendment process. Amendment 35, which was submitted to the Commission on May 25, 1994, was the last amendment. GE also submitted the ABWR certified design material (CDM) on August 31, 1993. The CDM contains the so-called Tier 1 design information that GE proposes to have certified during the design certification rulemaking. GE submitted Revision 4 to the CDM on May 25, 1994. The application with all SSAR amendments and CDM revisions is available for public inspection at the NRC Public Document Room, 2120 L Street, N.W., Washington, D.C. 20555. A summary of the U.S. ABWR standard design is provided in Sections 1.2 and 1.3 of this report. Section 1.4 identifies agents and contractors.

This safety evaluation report (SER) documents the staff's review of the ABWR SSAR, up to and including Amendment 35, and the CDM, up to and including Revision 4. The staff also reviewed the SSAR, CDM, and final technical specifications to ensure that this information was internally consistent. This resolved Confirmatory Item F1.1-1. This final SER documents the results of the staff's safety review of the U.S. ABWR standard design against the requirements of Subpart B of 10 CFR Part 52 (design certification) and delineates the scope of the technical details considered in evaluating the proposed design. The principal matters of the staff's review are summarized in Section 1.5 of this report.

The staff gave the status of its initial review of the ABWR SSAR in a series of "draft" SERs. These draft SERs and the chapters of the SSAR that were evaluated were submitted to the Commission as follows:

Commission Paper (Date)	SSAR Chapter
SECY-91-153 (May 24, 1991)	2-6, 17
SECY-91-235 (August 2, 1991)	3, 9-11, 13
SECY-91-294 (September 18, 1991)	7
SECY-91-309 (October 1, 1991)	19
SECY-91-320 (October 15, 1991)	18
SECY-91-355 (October 31, 1991)	2, 3, 5, 6, 8-10, 12-15

The staff also issued a "draft final" SER (DFSER) for the ABWR design, SECY-92-349, on October 14, 1992, and an advance copy of the SER related to certification of the ABWR design on December 30, 1993.

In a letter dated November 9, 1993, the NRC directed GE to revise the SSAR and CDM to conform with NRC's metrication policy, which was published in the Federal Register (57 FR 46202) on October 7, 1992. The NRC requested that these revised documents be submitted before March 4, 1994. As stated in the November 9, 1993, letter, the staff used dual units in the final SER. The staff verified that the SSAR and CDM conformed with the NRC's metrication policy.

In its application, GE stated that it was developing the ABWR design to meet the requirements in the Electric Power Research Institute's (EPRI's) Advanced Light Water Reactor Program. The Commission requested, in a staff requirements memorandum (SRM) dated December 15, 1989, that the staff evaluate deviations in the vendor designs from the EPRI "Advanced Light Water Reactor Utility Requirements Document." This was identified as Open Item 1.1-1 in the DFSER. GE stated in a letter dated April 29, 1993, that the SSAR satisfies the objectives of the policy guidance set forth by the Commission in the above SRM. The Commission designated this response to be acceptable in COMSECY-93-040, dated August 10, 1993. This resolved DFSER Open Item 1.1-1.

This SER references information in the SSAR that GE had requested to be withheld as proprietary in accordance with

10 CFR 2.790. The staff's determination regarding GE's request was provided in a letter to GE dated June 7, 1994. This resolved Open Item F1.1-1. Several references to GE topical reports are also made in this SER. Some of these topical reports contain information that the Commission authorized to be exempt from public disclosure, as provided by 10 CFR 2.790. For each such topical report and SSAR amendment containing proprietary information, a nonproprietary version, similar in content except for the omission of the proprietary information, is submitted to the NRC and is also available at the NRC Public Document Room. Reference to topical reports and SSAR information throughout this SER is made to the proprietary version.

Plant-specific applicants who reference the U.S. ABWR standard design in the future will retain architect-engineers, constructors, and consultants, as needed. As part of its review of an application for a combined license (COL), the staff will evaluate, for each plant-specific application that references the certified ABWR design, the technical competence of the applicant and its contractors to manage, design, construct, and operate a nuclear power plant. The plant-specific applicants will also be required to satisfy the requirements of Subpart C of 10 CFR Part 52 and the requirements resulting from the staff's review of this standard design, including the applicable regulations and exemptions identified in Section 1.6 of this report. GE identified additional requirements for the plant-specific applicants in the SSAR and identified them as "COL license information." The staff finds this acceptable. This resolved Open Item F1.9-1 and Confirmatory Item F1.9-1. An applicant for a COL will be required to discuss the COL license information in its application.

A list and definition of the abbreviations used throughout this report are provided in Appendix A. Appendix B gives the references for this report. A chronology of the principal actions related to the processing of this application and the submittal dates of the SSAR amendments is given in Appendix C. Appendix D lists the principal reviewers who evaluated the SSAR and CDM. Appendices E, F, and G give staff positions on shell buckling, steel embedments, and safety-related structures, respectively. Appendices H and I contain the staff's evaluation of structural walls and the ABWR pump and valve inservice testing plan, respectively. The "Human Factors Engineering Program Review Model and Acceptance Criteria for Evolutionary Reactors" is given in Appendix J and a list of PRA-based safety insights is given in Appendix K.

1.2 General Design Description

The U.S. ABWR design is similar to the international ABWR design, which is being built at the Kashiwazaki Kariwa Nuclear Power Generation Station, Units Nos. 6 and 7 (K-6/7), by the Tokyo Electric Power Company, Inc. GE summarized differences between the U.S. ABWR design and the K-6/7 project in a letter to the staff dated February 20, 1992. Since that time, it has made design changes to the U.S. ABWR standard design. GE committed to update the list of differences in its letter of February 20, 1992, and incorporate the final summary of design differences in the SSAR. This was identified as Confirmatory Item 1.2-1 in the DFSER. GE provided the updated list in SSAR Amendment 31. The staff finds this acceptable. This resolved Confirmatory Item 1.2-1.

The design of the K-6/7 project is a cooperative effort between GE, Hitachi, and Toshiba. All three parties review and sign off on the ABWR common engineering design documents for the K-6/7 project, which are generally upper-level design documents. GE has the lead for the reactor vessel and internals, the recirculation system, the control rod system, and the nuclear boiler system. Differences between the K-6/7 design and the U.S. ABWR design are identified and maintained, by an internal GE review process, in a controlled list for future design action and application (DAL). In addition to the design changes, this DAL process identifies other required changes for the ABWR common engineering documents (CEDs), such as U.S. code and regulatory requirements. GE plans to incorporate the DALs into the supporting design documentation for the U.S. ABWR design after design certification, possibly during a first-of-a-kind engineering activity.

A significant portion of the detailed K-6/7 design information used in developing the SSAR is retained at the offices of Hitachi and Toshiba in Japan. GE has stated that the contractual terms between GE and its associates allow GE access to the supporting design record files in Japan through October 29, 2001. In addition to the design information in GE's SSAR submittal, the staff has, in areas such as seismic, tornado, and high wind design, reviewed additional detailed design information that is not typically provided as part of a licensing submittal. Detailed design information determined to be necessary to support the staff's review findings was formally incorporated into the SSAR application.

The staff assessed GE's ABWR design process from March 30 through April 3, 1992, and summarized its assessment in a letter to GE dated May 15, 1992. It performed a subsequent inspection of the ABWR design process in September 1993. This inspection is addressed in Section 17.1.3 of this report. On June 16, 1992, GE responded to the staff's request for information in the May 15, 1992, letter on the following five issues:

1.2.1 GE's Design Process for the U.S. ABWR

GE stated in its letter of June 16, 1992, that "GE and its associates control the review and approval of ABWR Common Engineering design documents with a procedure using the Engineering Review Memorandum . . . . Evidence of design verification is entered into the design records of the responsible design organization." GE also stated that for engineering documents prepared uniquely by GE for the U.S. ABWR, changes to engineering documents are entered into the GE design record files. A COL applicant or holder must establish the design, including the supporting detailed design documentation, consistent with the ABWR design control document (DCD) referenced in the certified design rule. The required design process to establish the detailed design documentation in conformance with the DCD is the responsibility of the COL applicant or licensee. This was identified as COL Action Item 1.2.1-1 in the DFSER. GE provided this requirement in Section 1.1.4 of the SSAR. This is acceptable.

1.2.2 Precertification and Postcertification Design Control Procedures

GE stated in its letter of June 16, 1992, that "the same design control procedures described above apply to both Common Engineering and GE documents issued before as well as after certification of the U.S. ABWR." The staff's position is that GE must certify to the NRC, before final design approval (FDA), that the U.S. ABWR DCD has not been affected by any changes to the ABWR common engineering design documents. This was designated as Open Item 1.2.2-1 in the DFSER. In a letter dated June 11, 1993, GE stated that after the completion of the DCD, GE will certify that the Tier 1 and Tier 2 information has not been affected by any subsequent changes made in the common engineering design documents since the final Tier 1 and Tier 2 submittals. This letter changes DFSER Open Item 1.2.2-1 to Confirmatory Item F1.2.2-1.

Also, GE must provide to the staff a list of the ABWR common engineering design documents and DALs that apply to the U.S. ABWR design and their effective dates. This was identified as Open Item 1.2.2-2 in the DFSER. In a letter dated June 11, 1993, GE stated that after the completion of the DCD, GE will finalize the design action list and provide the corresponding effective dates of CEDs and DALs. This letter changes DFSER Open Item 1.2.2-2 to Confirmatory Item F1.2.2-2.

1.2.3 The role of the COL Applicant or Licensee

In its letter of June 16, 1992, GE discussed the change process for the so-called Tier 1 and Tier 2 information available to a COL applicant or licensee. The staff proposed a change process for Tier 1 and Tier 2 information in an advanced notice of proposed rulemaking (ANPR) that was published in the Federal Register (58 FR 58664) on November 3, 1993. If a final rule certifies the ABWR design, then it will explicitly state the change process to be followed for the Tier 1 and Tier 2 information.

1.2.4 Control of Design Documentation in Support of the Certified Design

GE stated in its letter of June 16, 1992, that "GE will apply its QA [quality assurance] process throughout the development of first-of-a-kind engineering phase (as well as subsequent phases of design definition), and will insure that all commitments made in the Design Certification (i.e., Tier 1 and Tier 2) are satisfactorily being implemented."

1.2.5 GE's Agreement With the Japanese

GE stated in its letter of June 16, 1992, that "in general, GE's agreement with the Japanese for design information exchange and availability falls under the Technical Cooperation Agreement (TCA) which has recently been renewed through October 29, 2001. Under the TCA, GE has been able to obtain detailed design information with a lead time of about two months."

1.2.6 Scope of Certified Design

The regulation in 10 CFR 52.47(b)(1) requires an applicant for certification of an evolutionary nuclear power plant design to provide an essentially complete design scope. Therefore, the scope of the U.S. ABWR design must include all of the plant that can affect safe operation except for its site-specific elements, such as the service water intake structure and the ultimate heat sink. This was identified as Open Item 1.2.6-1 in the DFSER. GE submitted a description of the ABWR standard plant scope, including the site-specific design elements that are either partially or completely outside the scope of the ABWR standard design, in Section 1.1.2 of the SSAR. The staff finds this acceptable. This resolved Open Item 1.2.6-1.

1.2.7 Summary of U.S. ABWR Standard Design

The ABWR is a single-cycle, forced-circulation, boiling water reactor (BWR) with a rated power of 3926 megawatts thermal (MWt) and a design power of 4005 MWt. This power level exceeds the guidance in Regulatory Guide (RG) 1.49, "Power Levels of Nuclear Power Plants," which states that licensed power levels should be limited to a reactor core power level of 3800 MWt or less until January 1, 1979, at the earliest. The intent of this regulatory guidance was to stabilize the maximum size of nuclear plants until sufficient experience was gained with design, construction, and operation of large plants. Since the issuance of RG 1.49, Revision 1, in 1973, the staff has reviewed sufficient operating experience and has determined that licensing the ABWR at a rated power of 3926 MWt is acceptable.

Unique features of the ABWR design include the internal recirculation pumps, fine-motion control rod drives, microprocessor-based digital control and logic systems, and digital safety systems. The reactor building includes the containment, drywell, and major portions of the nuclear steam supply system, steam tunnel, refueling area, diesel generators, essential and nonessential power, emergency core cooling systems (ECCSs), and heating, ventilating, and air conditioning (HVAC) systems. The control building includes the control room, the computer facility, the cable tunnels, some essential switchgear, some essential power, the reactor building cooling water system, and the essential HVAC system. The service building houses the technical support center, the operational support center, and the counting room for analyzing post-accident samples. The turbine building includes all equipment associated with the main turbine generator. The radwaste building includes all equipment associated with the collection and processing of solid and liquid radioactive waste generated by the plant.

<u>Reactor</u>

The reactor design consists of the reactor pressure vessel (RPV), pressure containing appurtenances (control rod drive housings, in-core instrumentation housing, head vent and spray assembly) and internal components. The internal components include the core, the core support structure, the shroud head and steam separator assembly, the steam dryer assembly, the feedwater spargers, the core spray, and the core flooding spargers. Except for zircaloy in the reactor core, the internals will be made of stainless steel or other corrosion-resistant alloys.

The reactor core will consist of 872 fuel bundles in an 8-by-8 array and 205 control rods operating at a power density of 50 kW/liter. The control rods, which will enter from the bottom of the reactor core, will perform dual functions of power distribution shaping and reactivity control. Manipulation of selected patterns of rods will control power distribution, while electro-hydraulic drive mechanisms or hydraulic rapid scram insertion will control reactivity.

<u>Reactor Coolant System</u>

The reactor coolant system (RCS) includes the nuclear boiler system; the main steam, feedwater, recirculation system; the reactor core isolation cooling (RCIC) system; the residual heat removal system; and the reactor water cleanup system. The design is different from current BWR designs in that 10 reactor internal pumps (RIPs) located within the reactor vessel will forcibly circulate reactor coolant. This will eliminate large piping connections to the reactor vessel below the core and also eliminate reactor recirculation system piping. Eighteen safety/relief valves in six groups will provide RCS overpressure protection.

<u>Reactor Protection System</u>

The reactor protection system (RPS) will initiate a rapid, automatic shutdown of the reactor to prevent fuel cladding damage and any nuclear system process barrier damage due to an abnormal transient. The RPS scram logic inputs are from the neutron monitoring system (NMS). The NMS is a system of in-core neutron detectors and out-of-core electronic monitoring equipment.

<u>Containment</u>

The ABWR has a pressure suppression primary containment system. The primary containment includes a drywell and a wetwell. The drywell consists of two volumes, an upper drywell surrounding the RPV and a lower drywell that houses RIPs, control rod drives, and service equipment. The wetwell consists of a suppression pool and an air volume that will serve as a heat sink during normal and accident conditions. A secondary containment surrounds the primary containment and permits monitoring and treating of all potential radioactive leakage from the primary containment.

<u>Electrical Power Distribution Supply</u>

The electrical power distribution system is a complete three load group distribution system with two independent offsite power sources, the main turbine generator, three onsite standby power sources (emergency diesel generators), and a combustion turbine generator located on site. During normal plant operations, the main generator will supply power to the main power transformer (MPT) and three unit auxiliary transformers (UATs) through the main generator output breaker and an isolated phase bus. When the main generator is off line, power will be supplied to the UATs and the MPT by the preferred offsite power source.

Emergency Core Cooling System

In the event of a breach in the reactor coolant pressure boundary that results in a loss of reactor coolant, three independent divisions of the ECCS will maintain fuel cladding below the temperature limit as defined by 10 CFR 50.46. Each division contains one high-pressure and one low-pressure inventory makeup system. The following systems make up the ECCS:

- high-pressure core flooder (HPCF) system
- RCIC system
- low-pressure flooder (LPFL) system
- automatic depressurization system (ADS)

Control Room

The main control room incorporates monitoring and control functions for normal and emergency plant operations. It consists of a single integrated control console staffed by two operators. The console has a low profile so that the operators can see over it from a seated position. A plant process computer system will drive on-screen control video display units (VDUs) for safety system monitoring and non-safety system control and monitoring. Two separate sets of on-screen control VDUs will be used for safety and non-safety system control and monitoring independent of the plant process computer system. Dedicated function switches are also provided on the control console. A large display panel with fixed position display of key plant parameters and major equipment status will be used by the entire control room operating staff. A supervisors console has "monitoring only" capability.

Power Conversion

The power conversion system is designed to convert the heat energy generated in the reactor to electrical energy. This system includes the main steam system, main turbine generator system, main condenser, condenser evacuation system, condensate cleanup system, and condensate feedwater pumping and heating system.

1.3 Comparison With Similar Facility Designs

While many features of the ABWR design are similar to those of BWR designs that the staff had previously approved, some features are unique, as discussed in Section 1.2.7 of this report. To the extent feasible and appropriate, the staff has relied on earlier reviews for those ABWR design features that are substantially the same as those previously considered. Where this has been done, the appropriate sections of this report identify the other designs. The SERs for the other designs have been published and are available for public inspection at the NRC Public Document Room, 2120 L Street, N.W., Washington, D.C. 20037.

1.4 Identification of Agents and Contractors

The ABWR common engineering design documents for the international ABWR design were developed by GE and its associates, Hitachi and Toshiba. The U.S. ABWR design is being developed by GE.

1.5 Summary of Principal Review Matters

The procedure for certifying a design is described in Subpart B of 10 CFR Part 52 and is implemented in two stages. The technical review stage begins with an application filed in accordance with the requirements of 10 CFR 52.45, continues with reviews by the NRC staff and the ACRS, and concludes with the issuance of a final SER related to design certification and notification that the applicable requirements of 10 CFR 52.47, 52.48, and 52.53 have been met. The administrative review stage begins with a Federal Register notice that initiates a rulemaking, in accordance with 10 CFR 52.51, and provides a proposed standard design certification rule. The rulemaking will be conducted by the Commission and also, in the event that there is a request for a hearing and the request is granted, by an Atomic Safety and Licensing Board. The rulemaking will culminate with the denial or issuance of a design certification rule. An advance notice of proposed rulemaking on design certification was published in the Federal Register (58 FR 58664) on November 3, 1993.

The staff's technical review of GE's application for certification of the U.S. ABWR standard design was performed in accordance with Commission guidance and the applicable requirements of Subpart B of 10 CFR Part 52. This final SER documents the results of the staff's technical review. The staff's evaluation of the technical information required by 10 CFR 52.47(a)(1)(i) was performed in accordance with the standard review plan (SRP, NUREG-0800) and is discussed throughout this report. The evaluation of the technically relevant unresolved safety issues, generic safety issues, and Three Mile Island requirements (10 CFR 52.47(a)(1)(ii) and (iv)) is given in Chapter 20 of this report. Site parameters required by 10 CFR 52.47(a)(1)(iii) are evaluated in Chapter 2 and listed in Section 2.6. The design-specific probabilistic risk assessment (10 CFR 52.47(a)(1)(v)) is evaluated in Section 19.1. Inspections, tests, analyses, and acceptance criteria (ITAAC), as required by 10 CFR 52.47(a)(1)(vi) and (viii), are evaluated in Section 14.3. Interface requirements and representative conceptual designs (10 CFR 52.47(a)(1)(vii) and (ix)) are evaluated in Chapters 8 and 9 and are also discussed in Section 14.3. The ABWR technical specifications are evaluated in Chapter 16 of this report. The staff also implemented the Commission's Severe Accident Policy Statement, dated August 8, 1985, and the Commission's SRMs on SECY-90-016 and SECY-93-087 in its resolution of severe accident issues. The staff's evaluation of severe accident issues is given in Section 19.2 of this report.

The regulations in 10 CFR 52.47(a)(2) describe the level of design information needed to certify a standard design. Determining the acceptable level of design detail necessary for the staff to make its safety findings was one of the most challenging aspects of the staff's review. The SRM for SECY-90-377, "Requirements for Design Certification Under 10 CFR Part 52," sets forth the Commission's position on what level of design information is required for a certification application, and the staff has followed that guidance in preparing this report. The staff determined that GE did not provide sufficient detail in the SSAR for the following four areas of the review: pipe stress analysis, radiation shielding and airborne concentrations, instrumentation and controls (I&C), and control room design. The staff based its safety decisions for these areas of the design on the use of design acceptance criteria (DAC). The staff's evaluation of the level of design information to be certified, including DAC (certified design material - CDM) is given in Section 14.3 of this report. As part of its technical review, the staff also made numerous requests for additional information to provide sufficient bases for its safety findings and to meet 10 CFR 52.47(a)(3). Its evaluation of the scope of the design to be certified (10 CFR 52.47(b)(1)) is given in Section 1.2.6. The requirement in 10 CFR 52.47(b)(2) does not apply because the ABWR is an evolutionary reactor design and 10 CFR 52.47(b)(3) does not apply because the ABWR is not a modular design.

The staff used the safety standards in 10 CFR 52.48 as the basis for its review of the U.S. ABWR standard design. It also followed Commission guidance given in the SRMs for various Commission papers referenced throughout this report and identified in Appendix B. As a result of this guidance, the staff proposed design-specific regulations that are applicable to the ABWR design and justified exemptions from existing regulations to complete the regulatory framework of safety standards. An index of these safety standards is contained in Section 1.6 of this report.

In the DFSER and advance copy of the SER, the staff identified many unresolved or open items. These open items were resolved as described throughout this report. After issuance of this report on the Staff's review of the SSAR and CDM, the applicant will submit a design control document (DCD) for the staff's review. This is Confirmatory Item F1.5-1. The DCD will consist of the CDM and Tier 2 information as described in Section 14.3 of this report. Applications that reference the certified ABWR design will be required to conform with the DCD. The DCD will be available for public inspection at the NRC Public Document Room when the proposed rule for design certification is published in the Federal Register.

1.6 Index of Applicable Regulations and Exemptions

In accordance with 10 CFR 52.48, the staff used the applicable regulations in 10 CFR Parts 20, 50, 73, and 100 in performing its review of GE's application for design certification. During this review, the staff identified certain regulations for which application of the regulation to the ABWR design would not serve or was not necessary to achieve the underlying purpose of the rule. These exemptions to the above regulations are discussed in the sections of this report identified below.

In the SRM pertaining to SECY-91-262, "Resolution of Selected Technical and Severe Accident Issues for Evolutionary Light Water Reactor Designs," the Commission approved the staff's recommendation to proceed with design-specific rulemakings as part of design certification rulemakings to establish selected technical and severe accident issues as "applicable regulations" for the ABWR and System 80+ standard designs. These issues included staff positions that deviated from or were not embodied in current regulations applicable to the standard designs. These issues were proposed in various Commission papers, such as SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolution and Advanced LWR Designs." The "applicable regulations" that are specific to the ABWR design are identified and evaluated in this report. The proposed design certification rule will include these "applicable regulations" for the ABWR design for the purposes of 10 CFR 52.48, 52.54, 52.59, and 52.63. These "applicable regulations" are discussed in the sections of this report identified below.

Section	Description of Issue
.1.1	Exemption from operating basis earthquake design requirement
3.9.3.1.1	Applicable regulation for intersystem loss-of-coolant accident
3.9.6	Applicable regulation for inservice testing of pumps and valves
3.11.1	Exemption from 10 CFR 50.49(b)(3)
7.2.6	Applicable regulation for digital instrumentation and control systems
8.2.2.9 and 8.2.3.4	Applicable regulations for electric power system
9.3.2.2	Exemption from postaccident sampling
9.5.1	Applicable regulation for fire protection
17.3	Applicable regulation for design reliability assurance program
8.3.2.2	Exemption from safety parameter display console
19.1.2	Applicable regulation for analysis of external events
19.2.2.1.2	Applicable regulation for station blackout
19.2.3.3.2	Applicable regulation for core debris cooling
19.2.3.3.3	Applicable regulation for high-pressure core melt ejection
19.2.3.3.7	Applicable regulation for equipment survivability
19.2.4	Applicable regulation for containment performance
19.3.2	Applicable regulation for shutdown risk
20.5.44	Exemption from dedicated containment penetration requirement

1.7 Index of Tier 2* Information

The staff determined that certain SSAR commitments discussed in the following sections of this report, if considered for a change by an applicant or licensee that references the certified ABWR design, will require prior NRC review and approval before the change is implemented. These SSAR commitments (so-called Tier 2* information) will be identified in the design control document. This is Confirmatory Item F1.7-1.

Section	Description of Commitment
3.8.1	American Society of Mechanical Engineers Boiler and Pressure Vessel Code 1989 Edition for structural design and construction (referenced twice)
3.8.3, 3.8.5	American Concrete Institute 349 for structural design and construction
3.8.4	American National Standards Institute/American Institute of Steel Construction N690 for structural design and construction
3.9.6.2.2	Design, qualification, and preoperational testing for motor-operated valves
3.10.1	Equipment seismic qualification methods
3.12	Piping DAC
4.2	First cycle fuel, control rod and design, and methods used to analyze these components
7.2.2.1	Essential multiplexing system design criteria
7.2.2.5	Self-test system design testing features
7.2.7	Instrument setpoint methodology
7.2.8	Electromagnetic interference protection criteria and standards; computer development
18.9.3	Human system interface design implementation process

1.8 Index of Confirmatory Items

In the DFSER and advance copy of the SER, the staff identified many confirmatory items. Most of these confirmatory items were resolved as described throughout this report. The following items with an F before the item number are confirmatory at the time of issuance of this final SER. These items will be resolved during the staff's review of the ABWR design control document.

Each confirmatory item was assigned a unique number that identifies the section in this report where the item is discussed. For example, Confirmatory Item F1.5-1 is discussed in Section 1.5 of this report.

Item Number	Description of Item
F1.2.2-1	Certification that DCD not affected by changes to CEDs
F1.2.2-2	Submittal of list of CEDs and DALs
F1.5-1	Submittal of DCD
F1.7-1	Identify Tier 2* information
F14.3.7.5-1	Reliability Assurance Program

2 SITE CHARACTERISTICS

The staff reviewed the site related parameters in Standard Safety Analysis Report (SSAR) Section 2, including the envelope of advanced boiling water reactor (ABWR) bounding site parameters listed in SSAR Table 2.0-1. The staff finds that GE Nuclear Energy's (GE's) list of site characteristics is consistent with that contained in appropriate sections of the Standard Review Plan (SRP) Chapter 2 and 10 CFR Parts 50, 52, and 100. In its review of the ABWR standard design, the staff finds that the acceptance criteria in the SRP for the review of site suitability for the site parameters shown in SSAR Table 2.1-1 are sufficient. The staff has not identified any unique features of the ABWR design that would require additional limitations for the selection of sites compatible with the ABWR design. Therefore, the combined license (COL) applicant may use the applicable SRP guidelines to evaluate the suitability of its site for the construction of the ABWR. To ensure that the ABWR design is enveloped by the site-related parameters, the staff will review site characteristics in detail during the COL application phase. The site-specific information to be provided by COL applicants referencing the ABWR design is discussed below.

It should be noted that the site-specific characteristics, which are discussed here as required at the time of a plant-specific COL application, may also be submitted to and considered by the Nuclear Regulatory Commission (NRC) in connection with an application for an early site permit under 10 CFR Part 52, Subpart A.

2.1 Geography and Demography

The COL applicant should provide site-specific information on site location and description, exclusion area authority and control, and population distribution.

2.1.1 Site Location and Description

The COL applicant should provide site-specific information on site location, including political subdivisions, natural and man-made features, population, highways, railways, waterways, and other significant features of the area. This was draft final safety evaluation report (DFSER) COL Action Item 2.1.1-1. GE has included this action item in Section 2.3.2.1 of the SSAR. This is acceptable.

2.1.2 Exclusion Area Authority and Control

The COL applicant should provide site-specific information on activities that may be permitted within the designated exclusion area. This was DFSER COL Action Item 2.1.2-1. GE has included this action item in Section 2.3.2.2 of the SSAR. This is acceptable.

2.1.3 Population Distribution

The COL applicant should provide population data for the site environs. This was DFSER COL Action Item 2.1.3-1. GE has included this action item in Section 2.3.2.3 of the SSAR. This is acceptable.

2.2 Nearby Industrial, Transportation, and Military Facilities

The COL applicant should provide site-specific information on identifying potential hazards in the site vicinity and evaluating potential accidents. These items are discussed in detail in Sections 2.2.1, 2.2.2, and 2.2.3 below.

2.2.1 and 2.2.2 Identification of Potential Hazards in Site Vicinity

The COL applicant should provide information on industrial, military, and transportation facilities and routes to establish the presence and magnitude of potential external hazards. This was DFSER COL Action Item 2.2.1-1. GE has included this action item in Section 2.3.2.4 of the SSAR. This is acceptable.

2.2.3 Evaluation of Potential Accidents

The COL applicant should identify potential accident situations in the vicinity of the plant and give the reasons why these potential accidents were or were not accommodated in the design. This was DFSER COL Action Item 2.2.3-1. GE has included this action item in Section 2.3.2.5 of the SSAR. This is acceptable.

2.3 Meteorology

2.3.1 Regional Climatology

In SSAR Table 2.0-1, GE originally proposed that the maximum tornado wind speed of 418 km/hr (260 mi/hr) and the tornado recurrence interval of 1 million years (tornado strike probability of 10E-6 per year) be used for the design-basis tornado (DBT). These parameters are based on American National Standards Institute/American Nuclear Society (ANSI/ANS)-2.3 (1983), "Standard for Estimating Tornado and Extreme Wind Characteristics at Nuclear Reactor Sites." The current NRC regulatory position with regard to the DBT is contained in WASH-1300, "Technical Basis for Interim Regional Tornado Criteria (1974)," and Regulatory Guide (RG) 1.76, "Design Basis Tornado for Nuclear Power Plants," Revision 0. WASH-1300 states that the probability of occurrence of a tornado that exceeds the DBT should be on the order of 10E-7 per year per nuclear power plant, and the RG specifies maximum wind speeds of 386 km/hr

(240 mi/hr) to 579 km/hr (360 mi/hr) depending on the regions.

The staff has not endorsed ANSI/ANS-2.3. However, in 1986, the regulatory positions in RG 1.76 was reevaluated, using the considerable quantity of tornado data that was available. The reevaluation is discussed in NUREG/CR-4461. At the heart of this study is the tornado data tape prepared by the National Severe Storm Forecast Center with 30 years of data, 1954 through 1983. This tape contains the data for the approximately 30,000 tornados that occurred during the period.

The reevaluation showed that the tornado strike probabilities range from near 10E-7 per year for much of the Western United States to about 10E-3 per year in the Central States. As a result, wind speed values associated with a tornado having a mean recurrence interval of 10E-7 per year were estimated to be about 322 km/hr (200 mi/hr) for the United States west of the Rocky Mountains and 483 km/hr (300 mi/hr) for the United States east of the Rocky Mountains.

The DBT requirements have been used in establishing structural requirements (e.g., minimum concrete wall thicknesses) for the protection of nuclear plant safety-related structures, systems, and components (SSCs) against the effects not covered explicitly in RGs or the SRP. Specifically, the staff has routinely evaluated some aviation (general aviation light aircraft) crashes, nearby explosions, and explosion debris or missiles by considering the tornado protection requirements. In the DFSER, the staff noted that COL applicants should evaluate the effects on the protection criteria of some external impact hazards, such as general aviation or nearby explosions. This was DFSER COL Action Item 2.3.1-1. Section 2.3.2.6 of the SSAR includes this action item. This is acceptable.

On the basis of updated tornado data and the analysis in NUREG/CR-4461, the staff concluded that it is acceptable to reduce the DBT wind speeds to 322 km/hr (200 mi/hr) for the United States west of the Rocky Mountains and to 483 km/hr (300 mi/hr) for the United States east of the Rocky Mountains. In SECY-93-087, the staff gives its position on the tornado design basis. The Commission in its staff requirements memorandum of July 21, 1993, approved the staff-recommended position that a maximum tornado wind speed of 483 km/hr (300 mi/hr) be used for the DBT for advanced light water reactors. As a result, Table 2-1 shows the DBT parameters that the staff considers acceptable for the ABWR design. This table, which is based on data from NUREG/CR-4461, displays the geographical boundaries for the characteristics of the DBT.

In the DSER (SECY-91-355) and DFSER, the staff requested that GE revise the DBT characteristics for the ABWR to reflect the data in Table 2-1 of this report. This was DSER Outstanding Issue 146 and DFSER Confirmatory Item 2.3.1-1. GE revised Table 2.0-1, "Envelope of ABWR Standard Site Design Parameters," of the SSAR to reflect the DBT characteristics specified in Table 2-1 of this report. The maximum tornado wind speed of 483 km/hr (300 mi/hr) is also specified in Table 5.0, "ABWR Site Parameters," in the ABWR certified design material (CDM). This is acceptable. Therefore, DSER Outstanding Issue 146 and DFSER Confirmatory Item 2.3.1-1 are resolved.

Table 2-1 Design-basis tornado characteristics

	Maximum wind speed		Rotational speed		Translational speed		Radius of maximum rotational speed		Pressure drop		Rate of pressure drop	
Region*	km/hr	mi/hr	km/hr	mi/hr	km/hr	mi/hr	m	ft	kPa	psi	kPa/sec	psi/sec
I	483	300	386	240	97	60	46	150	13.8	2.0	8.3	1.2
II	354	220	274	170	80	50	46	150	6.9	1.0	3.4	0.5
III	322	200	257	160	64	40	46	150	6.2	0.9	2.1	0.3

*See Figure 2-1 (RG 1.76, "Design Basis Tornado for Nuclear Power Plants," Rev. 0) for a map of the tornado intensity regions.

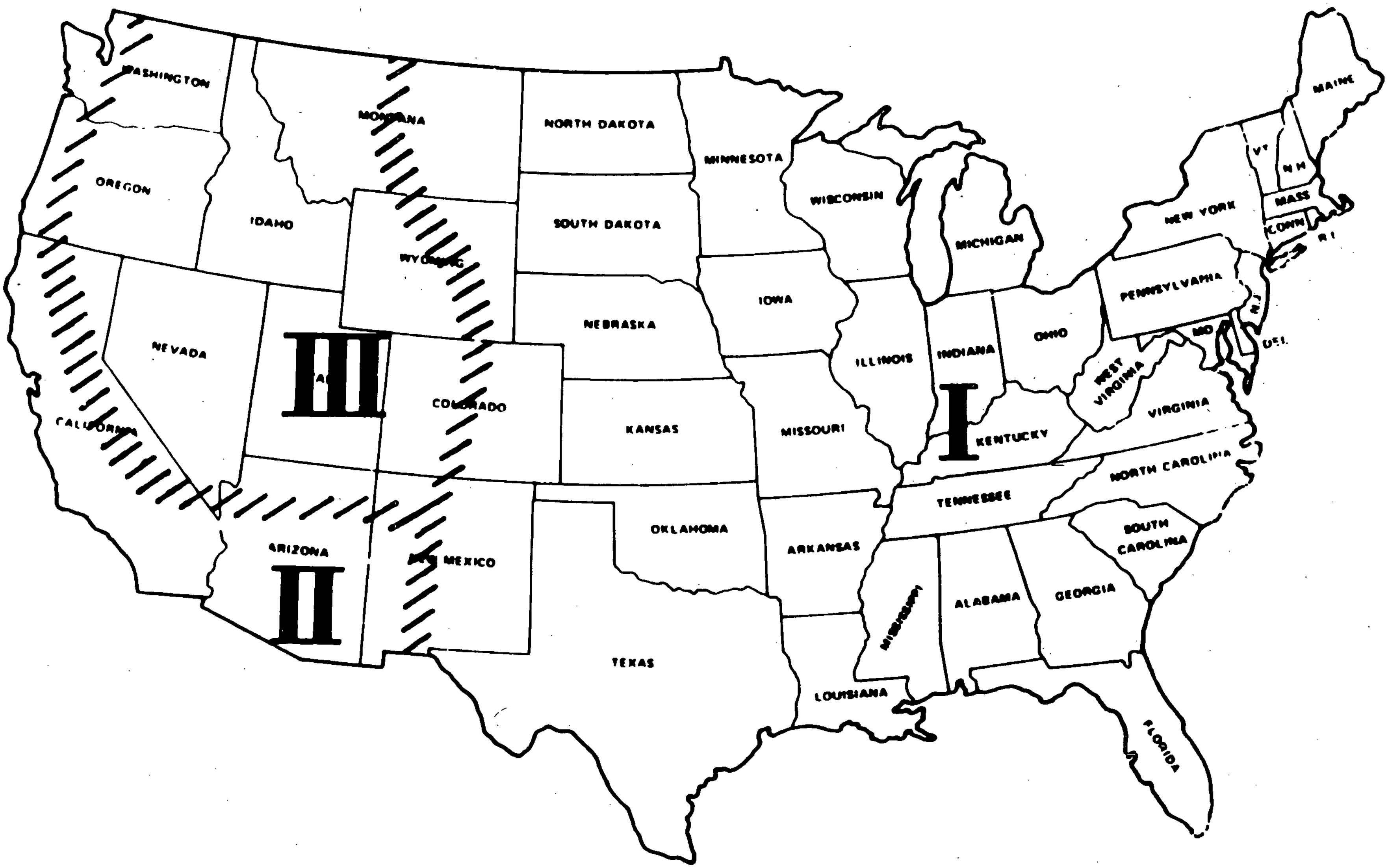


Figure 2-1 Tornado intensity regions

2.3.2 Local Meteorology

Since local meteorology is specifically site-related, the COL applicant will provide local meteorology for review by the staff on a case-by-case basis. This was DFSER COL Action Item 2.3.2-1. Section 2.3.2.7 of the SSAR identifies this action item. This is acceptable.

2.3.3 Onsite Meteorological Measurements Program

Details on the atmospheric diffusion characteristics of a proposed nuclear power plant site are required to determine if postulated accidental, as well as routine operational, releases of radioactive materials are within NRC regulatory guidelines. The meteorological characteristics of a proposed site are determined by staff evaluation of meteorological data collected at the site in accordance with RG 1.23, "Onsite Meteorological Programs," Revision 0.

The COL applicant should provide the onsite meteorological measurements program for review by the staff. This was DFSER COL Action Item 2.3.3-1. Section 2.3.2.8 of the SSAR identifies this action item. This is acceptable.

2.3.4 Short-Term Dispersion Estimates for Accidental Atmospheric Releases

The bounding atmospheric relative concentrations (χ/Q) for design-basis accident evaluations are listed in SSAR Tables 15.6.3, 15.6.7, 15.6.13, 15.6.14, and 15.6.18. The staff concludes, in Section 15.4 of this report, that these values provide a reasonable basis for evaluating the consequences of design-basis accidents. The COL applicant should provide site-specific short-term dispersion estimates in accordance with RG 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," Revision 1, so the staff can ensure that the bounding values of atmospheric relative concentrations are not exceeded. This was DFSER COL Action Item 2.3.4-1. Section 2.3.2.9 of the SSAR identifies this action item. This is acceptable. The bounding χ/Q values for the 2-hour exclusion area boundary (EAB) and 2-hour low population zone (LPZ) are specified in Table 2.0-1 of the SSAR and the CDM.

2.3.5 Long-Term Diffusion Estimates

The staff will evaluate annual average atmospheric dispersion values for routine releases using the guidance in RG 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases From Light-Water-Cooled Reactors," Revision 1. The staff will use these values to perform its 10 CFR Part 50, Appendix I, and 10 CFR Part 20 evaluations for a COL application. The COL applicant should provide to the NRC annual average atmospheric dispersion values for routine releases. This was DFSER COL Action Item 2.3.5-1. GE has included this action item in Section 2.3.2.10 of the SSAR. This is acceptable.

2.4 Hydrologic Engineering

2.4.1 Hydrologic Description

The COL applicant should provide a detailed description of all major hydrologic features on or in the vicinity of the site. It should also provide a specific description of the site and critical elevations of all safety-related structures, exterior accesses, equipment, and systems from the standpoint of hydrology considerations. This was DFSER COL Action Item 2.4.1-1. GE has included this action item in Section 2.3.2.11 of the SSAR. This is acceptable.

2.4.2 Floods

The COL applicant should provide site-specific information on historical flooding and potential flooding at the plant site, including flood history, flood design considerations, and effects of local intense precipitation. This was DFSER COL Action Item 2.4.2-1. GE has included this action item in Section 2.3.2.12 of the SSAR. This is acceptable.

2.4.3 Probable Maximum Flood on Streams and Rivers

The COL applicant should provide site-specific information used for determining design-basis flooding at the power reactor sites and the extent of flood protection required for safety-related SSCs. This was DFSER COL Action Item 2.4.3-1. GE has included this action item in Section 2.3.2.13 of the SSAR. This is acceptable.

2.4.4 Ice Effects

The COL applicant should provide site-specific information on ice effects and demonstrate that safety-related facilities and water supply will not be affected by ice flooding or blockage. This was DFSER COL Action Item 2.4.4-1. GE has included this action item in Section 2.3.2.14 of the SSAR. This is acceptable.

2.4.5 Cooling Water Channels and Reservoirs

The COL applicant should provide the basis for the hydraulic design of canals and reservoirs used to transport and impound plant cooling water and for the protection of safety-related structures. This was DFSER COL Action

Item 2.4.5-1. GE has included this action item in Section 2.3.2.15 of the SSAR. This is acceptable.

2.4.6 Channel Diversion

The COL applicant should provide site-specific information on channel diversion. This was DFSER COL Action Item 2.4.6-1. GE has included this action item in Section 2.3.2.16 of the SSAR. This is acceptable.

2.4.7 Flooding Protection Requirements

The COL applicant should provide site-specific information related to flooding protection requirements. This was DFSER COL Action Item 2.4.7-1. GE has included this action item in Section 2.3.2.17 of the SSAR. This is acceptable.

2.4.8 Cooling Water Supply

The COL applicant should identify natural events that may reduce or limit the available cooling water supply and ensure that an adequate water supply will exist to operate or shut down the plant as required. This was DFSER COL Action Item 2.4.8-1. GE has included this action item in Section 2.3.3.18 of the SSAR. This is acceptable.

2.4.9 Accidental Release of Liquid Effluents in Ground and Surface Waters

The COL applicant should provide information on the capability of the surface water environment to disperse, dilute, or concentrate accidental releases. Effects of these releases on existing and known future uses of surface water resources should also be provided. This was DFSER COL Action Item 2.4.9-1. GE has included this action item in Section 2.3.2.19 of the SSAR. This is acceptable.

2.4.10 Technical Specification and Emergency Operation Requirement

The COL applicant should establish the technical specifications and emergency procedures required to implement flood protection for safety-related facilities and provide assurance of an adequate water supply to shutdown and cool the reactor. This was DFSER COL Action Item 2.4.10-1. GE has included this action item in Section 2.3.2.20 of the SSAR. This is acceptable.

2.5 Geology, Seismology, and Geotechnical Engineering

The COL applicant should provide site-specific, basic geologic and seismic information and site-specific information on vibratory ground motion, surface faulting, stability of subsurface materials and foundations, slopes, and embankments and dams as described in the following sections.

2.5.1 Basic Geologic and Seismic Information

The COL applicant should provide site-specific information on regional and site physiography, geomorphology, stratigraphy, lithology, and tectonics. This was DFSER COL Action Item 2.5.1-1. GE has included this action item in Section 2.3.2.21 of the SSAR. This is acceptable.

2.5.2 Vibratory Ground Motion

In the DFSER, the staff noted that the COL applicant should provide site-specific information on the seismicity, geologic, and tectonic characteristics of the site and region; correlation of earthquake activity with geologic structure or tectonic provinces; maximum earthquake potential; seismic wave transmission characteristics of the site; safe shutdown earthquake (SSE); and operating basis earthquake (OBE). This was DFSER COL Action Item 2.5.2-1. The Commission has approved the staff recommendation that the OBE be eliminated as discussed in Section 3.1.1 of this report. GE has modified SSAR Section 2.3.2.22 by stating that the COL applicant will develop site-specific geological, seismological, and geotechnical data and will compare the site-specific SSE ground response spectra with the design ground response spectra of SSAR Section 2.3.1.2. This is acceptable.

2.5.3 Surface Faulting

The COL applicant should provide detailed geological and geophysical information related to the potential for surface faulting affecting the site.

Originally, GE imposed no limit for surface faulting (SSAR Table 2.1-1). However, it is the staff position, as stated in RG 4.7, "General Site Suitability Criteria for Nuclear Power Stations," Revision 1, that "sites that include capable faults, as defined in Appendix A to 10 CFR Part 100, are not suitable for nuclear power stations." Therefore, in the DFSER, the staff noted that the COL applicant should develop site-specific information to ensure that no potential exists for surface faulting affecting the site. This was DFSER COL Action Item 2.5.3-1. GE has included this action item in Section 2.3.2.23 of the SSAR. This is acceptable.

2.5.4 Stability of Subsurface Materials and Foundations

In response to NRC staff Question 241.1a, GE stated that the COL applicant will provide site-specific geotechnical data to demonstrate that they are comparable to the site design parameters given in SSAR Table 2.0-1. The staff requires that the COL applicant's submittals meet the guidance in Section 2.5.4 of RG 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," Revision 3. A summary of appropriate guidance is given below.

The staff will review the geotechnical engineering aspects of a COL applicant's design, design criteria, and design bases related to the stability of subsurface materials and foundations of safety-related facilities for an ABWR standard plant in accordance with SRP Section 2.5.4.

In SSAR Appendix 3A, which deals with seismic soil-structure interaction (SSI) analyses, GE characterized the site conditions in terms of (1) soil deposit depth above bedrock, (2) ground water level, and (3) soil profile and properties, and also gave parameter variations in each of these three areas for establishing the site envelope.

(1) GE accounted for the variations in soil deposit thickness by considering three representative soil deposit depths: (1) shallow soil deposits (46 m (150 ft)), (2) intermediate soil deposits (61 m (200 ft)), and (3) deep soil deposits (91 m (300 ft)). It assumed a minimum depth of embedment of 26 m (85 ft) for the case where the building will be supported directly by rock.

    Appendix 3A of the SSAR states that the SSI analyses were performed using the same minimum embedment depths for the different site categories. Whether the reactor building will be supported onrock or soil, the minimum embedment depth will be 26 m (85 ft). GE stated that the ABWR design "does not allow for depths of embedment less than 26 m (85 ft) even if competent rock will be available at a site at depths much less than 26 m (85 ft). SSAR Tables 3A.3-2 and 3A.3-6 show that a 26-m (85-ft) embedment depth will be used in SSI analyses even when hard rock (HR) and extra hard rock (EHR) are at ground surface.

(2) GE evaluated the effects of variations in water table locations on structural response by considering three water table locations: low, intermediate, and high. It assumed the high water table will be located at 0.6 m (2 ft) below grade, while the low water table will be located at 26 m (85 ft) below grade at the base of the reactor building foundation basemat. The intermediate water table was assumed to be located at 12 m (40 ft) below grade, approximately at the midheight of the reactor building embedment.

(3) GE considered a range of soil profiles based on the shear wave velocity profiles used in GESSAR II (NUREG-0979, April 1983) and selected six velocity profiles for the SSI analyses.

- The first soil profile was assumed to consist of seven horizontal layers. The shear wave velocity, V_s, of the soil at a depth, y, below the ground surface was calculated as a function of the effective mean pressure at that depth and a modulus parameter.

- The second through the sixth soil profiles were selected on the basis of three generalized soil zones: a soil zone for the second profile (sands, silts, clays, and gravelly soils), a transition zone for the third and fourth profiles, and a soft rock and well-cemented soil zone for the fifth and sixth profiles. Their velocity profiles are smooth curves representative of the average variation of shear modulus with depth.

- The seventh profile represents an HR site with a uniform V_s of 1,525 m/sec (5,000 ft/sec).

- The eighth profile represents an EHR site with a uniform V_s of 3,050 m/sec (10,000 ft/sec).

GE considered the variation in shear modulus and material damping of soil with shear strain for various soil profiles. It assumed the best-estimate values of soil-shear modulus to be not less than 40 percent of their low-strain values. GE limited the values of hysteretic soil damping to a maximum of 15 percent of critical as recommended by SRP Section 3.7.2. On the basis of the above constraints, GE developed the shear-modulus reduction factors and damping ratios at various strain levels. The above assumptions are acceptable.

In the DFSER, the staff noted that the COL applicant should develop and submit to the NRC site-specific geotechnical data to demonstrate that they are comparable to the design assumptions. This was DFSER COL Action Item 2.5.4-1. GE has modified Section 2.3.2.24 in the SSAR by stating that the COL applicant will provide information concerning the properties and stability of site-specific soil and rocks under both static and dynamic conditions including the vibratory ground motions associated with the site-specific SSE. This is acceptable.

2.5.4.1 Site and Facilities

The COL applicant should provide a detailed description of the site conditions and geologic features. The description should include site topographical features and the location of various seismic Category I structures and appurtenances (pipelines, channels, and so forth) with regard to the source of normal and emergency cooling water. This was DFSER COL Action Item 2.5.4.1-1. GE has included this action item in Section 2.3.2.25 of the SSAR. This is acceptable.

2.5.4.2 Field Investigations

The COL applicant should submit to the NRC a discussion of the type, quantity, extent, and purpose of all field exploration. Logs of all borings and test pits should be provided. Results of geophysical surveys should be presented in tables and profiles. Records of field plate load tests, field permeability tests, and other special field tests (e.g., bore-hole extensometer or pressuremeter tests) should also be given. This was DFSER COL Action Item 2.5.4.2-1. GE has included this action item in Section 2.3.2.26 of the SSAR. This is acceptable.

2.5.4.3 Laboratory Investigations

The COL applicant should provide tables of the number and type of laboratory tests and the location of samples and discuss the results of laboratory tests on disturbed and undisturbed soil and rock samples obtained from field investigations. This was DFSER COL Action Item 2.5.4.3-1. GE has included this action item in Section 2.3.2.27 of the SSAR. This is acceptable.

2.5.4.4 Subsurface Conditions

The COL applicant should investigate and define the subsurface conditions and provide the engineering classifications and descriptions of soil and rock supporting the foundations. The information should include the history of soil deposition and erosion, past and present ground water levels, glacial or other preloading influences, rock weathering, and any rock or soil characteristics that may present a hazard to plant safety. Profiles through the seismic Category I structures should be provided that show the generalized subsurface features beneath these structures. This was DFSER COL Action Item 2.5.4.4-1. GE has included this action item in Section 2.3.2.28 of the SSAR. This is acceptable.

2.5.4.5 Excavation and Backfilling for Foundation Construction

The COL applicant should provide site-specific information on the thickness and properties of the soil between the base of the foundation and the underlying rock. The configuration, along with detailed longitudinal sections and cross-sections of other safety-related structures of the plant, including the ultimate heat sink and seismic Category I buried pipes and electrical ducts, should be provided. The COL applicant should provide data on the extent (horizontally and vertically) of all seismic Category I excavations, fills, and slopes. The locations, elevations, and grades for excavated slopes should be described and shown on plot plans and typical cross-sections. The COL applicant should discuss, as appropriate, excavating and dewatering methods, excavation depths below grade, field inspection and testing of excavations, protection of foundation excavations against deterioration during construction, and the foundation dental fill work. The sources, quantities, and static and dynamic engineering properties of borrow materials should be described. The compaction requirements; results of test fills; and fill properties, such as moisture content, density, permeability, compressibility, and gradation also should be provided. This was DFSER COL Action Item 2.5.4.5-1. GE has included this action item in Section 2.3.2.29 of the SSAR. This is acceptable.

2.5.4.6 Effect of Ground Water

The COL applicant should analyze the ground water condition for the specific site and evaluate the effect of ground water level on such site geotechnical properties as total and effective unit weights, cohesion and angle of internal friction, and dynamic soil properties used in dynamic response analysis. This was DFSER COL Action Item 2.5.4.6-1. GE has included this action item in Section 2.3.2.30 of the SSAR. This is acceptable.

2.5.4.7 Liquefaction Potential

GE stated in response to NRC staff Question 241.1c that one of the eight conditions in Section 3A.1 of Appendix 3A of the SSAR required that no potential for liquefaction of soils shall exist at the plant as a consequence of the OBE and the SSE as reviewed and concurred in by the NRC staff. That condition further required that the liquefaction potential of the foundation and site soils be investigated and reported for a long-duration, New Madrid-type earthquake. GE clarified the statement regarding the New Madrid-type earthquake by stating that the maximum ground motion will be the same as the SSE and the actual duration of the earthquake chosen for a specific site will be reviewed by the NRC staff and approved at the time of an individual application. GE further stated that, without knowing the exact site location and its seismic hazard, it cannot specify the earthquake magnitude and the number of strong motion cycles for the liquefaction evaluation of a site.

The COL applicant should justify the selection of the soil properties used in the liquefaction potential evaluation (e.g., laboratory tests, field tests, and published data), the magnitude and duration of the site-specific earthquake, and the number of cycles of earthquakes. This was DFSER COL Action Item 2.5.4.7-1. GE has included this action item in Section 2.3.2.31 of the SSAR. This is acceptable.

2.5.4.8 Response of Soil and Rock to Dynamic Loading

In Appendix 3A of the SSAR, GE provides standard curves showing the variation in shear modulus and material damping with shear strain for the various soil and rock profiles (except for the EHR profile) described in Section 2.5.4 of this report. Further, GE limits the reduced values of the soil shear modulus to not less than 40 percent of their low strain values and the values of internal (hysteretic) soil damping to a maximum of 15 percent of critical, as shown in SSAR Tables 3A.3.3 and 3A.3.4. For the HR and EHR profiles, initial shear modulus and a nominal material damping of 0.1 percent were used in the SSI analyses.

In the DFSER, the staff noted that the COL applicant should establish and document site-specific geotechnical properties to demonstrate that they are comparable to the conditions used for the seismic design envelope discussed in Section 3.7.2 of this report. This was DFSER COL Action Item 2.5.4.8-1. GE has revised Section 2.3.2.32 determine dynamic soil properties of the site in terms of shear modulus and material damping as a function of shear strain. These strain-dependent properties will be used in determining the site-specific SSE ground motion. This is acceptable.

2.5.4.9 Maximum Soil-Bearing Pressures

In Appendix 3H of the SSAR, GE gives a method for calculating maximum bearing pressure under the reactor building foundation mat for three load cases. Load Case 1 considers dead load, live load, and a combination of horizontal and vertical SSE components with the vertical component acting downward. Load Case 2 is the same as Load Case 1 except that the vertical seismic component acts upward and the live load is omitted. Load Case 3 is the same as Load Case 2, except for the addition of the buoyancy effect. Because the soil bearing pressure based on static equilibrium in the case of uplift would be very conservative, GE used the energy balance method. GE specified a minimum static soil bearing capacity of 0.72 megapascal (MPa) (15 kips/ft^2). The maximum bearing pressure for the ABWR standard plant, which is due to dead load alone, as calculated by GE, is 0.63 MPa (13.1 kips/ft^2). In the DFSER, the staff noted that the COL applicant should provide the site-specific maximum soil pressures along with supporting calculations and compare them with the allowable values. This was DFSER COL Action Item 2.5.4.9-1. SSAR Table 2.0-1 lists the minimum static bearing capacity of 0.72 MPa (15 kips/ft^2) as the site design parameter. GE has revised Section 2.3.2.33 of the SSAR by stating that the COL applicant will demonstrate that the site has a minimum static bearing capacity at the foundation level of the reactor and control buildings. For other safety-related plant facilities, the COL applicant will demonstrate that the foundation material has adequate bearing capacity to withstand the site-specific loads. This is acceptable.

2.5.4.10 Earth Pressures

The COL applicant should provide a site-specific discussion and evaluation of static and dynamic lateral earth pressures and hydrostatic ground water pressures acting on plant safety-related facilities. This was DFSER COL Action Item 2.5.4.10-1. GE has included this action item in Section 2.3.2.34 of the SSAR. This is acceptable.

2.5.4.11 Soil Properties for Seismic Analysis of Buried Pipes

The COL applicant should provide and justify the soil properties used for the seismic analysis of seismic Category I buried pipes and electrical conduits. This was DFSER COL Action Item 2.5.4.11-1. GE has included this action item in Section 2.3.2.35 of the SSAR. This is acceptable.

2.5.4.12 Static and Dynamic Stability of Facilities

The COL applicant should perform stability evaluation or analysis of all safety-related facilities. These analyses should include foundation rebound, settlement, differential settlement, and bearing capacity. Assumptions made in stability analyses should be confirmed by as-built data. This was DFSER COL Action Item 2.5.4.12-1. GE has included this action item in Section 2.3.2.36 of the SSAR. This is acceptable.

2.5.4.13 Subsurface Instrumentation

The COL applicant should describe instrumentation, if any, proposed for monitoring of the performance of the foundations for safety-related structures. The type, location, and purpose of each instrument and significant details of installation methods should be provided. For example, the location and the installation procedures for permanent benchmarks and markers required for monitoring the settlement of Category I structures should be described. In the case of safety-related, water-control structures (such as dams, slopes, canals), the installation of instrument such as piezometers, slope indicators, and settlement plates should be described in detail. A schedule for installing and reading all instruments and for interpreting the data should be given. Limiting values for continued safety should be identified. This was DFSER COL Action Item 2.5.4.13-1. GE has included this action item in Section 2.3.2.37 of the SSAR. This is acceptable.

2.5.4.14 Stability of Slopes

The COL applicant should provide information about the static and dynamic stability of all soil and rock slopes, the failure of which could adversely affect the safety of the plant. The staff will evaluate the stability of all slopes at the site, using the state-of-the-art procedures available at the time of the application. This was DFSER COL Action Item 2.5.4.14-1. GE has included this action item in Section 2.3.2.38 of the SSAR. This is acceptable.

2.5.4.15 Embankments and Dams

The COL applicant should provide information about the static and dynamic stability of all embankments and dams (if used) that will impound water required for safe operation and shutdown of the ABWR plant. This was DFSER COL Action Item 2.5.4.15-1. GE has included this action item in Section 2.3.2.39 of the SSAR. This is acceptable.

2.6 Site Parameter Envelope

The staff reviewed GE's analysis and evaluation of the ABWR design in terms of the bounding site parameters in SSAR Table 2.0-1. In the DFSER, the staff noted that the list of bounding site parameters in Table 2.0-1 should be comprehensive and include any additional items from the ABWR certified design material (CDM). Since this document was still being developed, the staff also noted in the DFSER that GE should ensure that the final list of site parameters in Table 5.0 of the CDM agrees with SSAR Table 2.0-1. Since GE had adopted the bounding site parameters identified in Table 1.2-6 of Electric Power Research Institute's (EPRI's) "Advanced Light Water Reactor (ALWR) Utility Requirements Document - Evolutionary Plant Designs," the staff also asked GE to adequately address the issues identified in Section 1.4 (Open Issue 1) of the SER on EPRI's ALWR requirements document for evolutionary plants when developing the final list. This was DFSER Open Item 2.6-1.

As part of this open item, the staff also requested that GE, in its revision to SSAR Table 2.0-1 and Table 5.0 of the CDM, include the following changes:

- All units and dimensions in these tables should be in the metric system with English units or dimensions provided in brackets.
- The following information should be added to the bounding site parameter for tornado:
  - rate of pressure drop: 8.3 kPa (1.2 lb/in^2/sec)
- The following additions or changes should be made for the seismology bounding site parameter:
  - Note (10) should be added to the fourth bullet and modified to read: "SSE Time History: Envelope SSE Response Spectra$^{(10)}$"
  - Note (9) should be changed to read: "The minimum bearing capacity should be referred to as the static bearing capacity."
  - New Note (10) should read: "The response spectra of the SSE time history to be used in the free field must envelop the free field design response spectra for all damping values to be used in the response analysis. In addition, the time history should also be justified to show its adequacy by demonstrating sufficient energy at the frequencies of interest through the generation of the power spectrum density (PSD) function, which is greater than the target PSD function throughout the frequency range of significance."

GE responded to this open item by letters to the staff dated April 16 and June 16, 1993. The staff and GE also discussed it in several conference calls and in the meeting on ITAAC (inspections, tests, analyses, and acceptance criteria) on July 27 through 29, 1993. To provide a comprehensive list of site parameters, the following site parameters were added to SSAR Table 2.0-1, "hazards in site vicinity," and "tornado, rate of pressure drop." This is acceptable. To address the issues identified in Sections 1.4 and 4.5.2 of the SER on EPRI's ALWR requirements document for evolutionary plant, SSAR Table 2.0-1 defines minimum "static" bearing pressure and establishes a requirement, "SSE time history: envelope SSE response spectra." This is acceptable. The staff also reviewed the enveloping meteorological dispersion values in Table 2.0-1 of the SSAR. The staff finds that GE determined these enveloping values using acceptable methodology given in RGs 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents From Light-Water-Cooled Power Reactors," Revision 1, and 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," Revision 1. The methods used by GE for determining these enveloping values are described in Sections 2.6.7 and 15.4 of this report. The staff finds the enveloping meteorological dispersion values acceptable. SSAR Section 3.7 addresses the issue noted in suggested Note (10) above. The staff's evaluation of this subject is discussed in detail in Section 3.7.1 of this report. The staff reviewed Table 2.0-1 of the SSAR and determined that it provides an adequate amount of information in support of Table 5.0 of the ABWR CDM. This is acceptable. In its letter of April 15, 1993, GE explained the use of only the Japanese metric units for the SSAR. Subsequently, GE agreed to use the International System of Units for the SSAR in accordance with the Commission-approved staff recommendations on implementing the metrication policy for evolutionary and revolutionary reactors. This is acceptable. On the basis of the above, DFSER Open Item 2.6-1 is resolved. The site parameter evaluation is summarized in the following sections.

2.6.1 Wind and Tornado Design Site Parameters

The staff's review of the wind and tornado loadings for the ABWR design is contained in Section 3.3 of this report. The bounding site parameters that were considered in the staff's evaluation are as follows:

- Basic Wind Speed: For the design of ABWR non-safety-related structures, the basic wind speed for a 50-year recurrence interval is 177 km/hr (110 mph). An "importance factor" of 1.0 should be used in accordance with the velocity pressure formula of SSAR Section 3.3.1.1. For the design of safety-related structures, the basic wind speed for a 100-year recurrence interval is 197 km/hr (122.1 mph). This value is obtained by multiplying the 50-year speed of 177/km/hr (110 mph) by an importance factor of 1.11, as noted in Appendix 3H of the SSAR.

- Maximum Tornado Wind Speed: 483 km/hr (300 mi/hr)

- Translational Velocity: 97 km/hr (60 mi/hr)

- Radius of Maximum Rotational Speed: 45.7 m (150 ft)

- Maximum Atmospheric Pressure Drop: 13.8 kPa ($2.0\ lb/in^2$)

- Missile Spectra: SRP Section 3.5.1.4, Spectrum I

The staff concludes in Sections 3.3.1, 3.3.2, and 3.5.1.4 of this report that the ABWR standard plant has been adequately designed for the above bounding site parameters.

2.6.2 Water Level (Flood) Design Site Parameters

The staff's review of the ABWR water level (flood) design is contained in Section 3.4 of this report. The bounding site parameters that were considered in the staff's evaluation are as follows:

- Floods: The ABWR should be located on the site so that the level of the design-basis flood is no higher than 30.5 cm (1 ft) below the plant grade.

- Potential Dam Failures (Seismically Induced): Failure of existing and potential upstream or downstream water control structures should not contribute to the water level exceeding 30.5 cm (1 ft) below grade nor compromise the ultimate heat sink.

- Probable Maximum Surge and Seiche Flooding: The probable maximum surge and seiche flooding should be no higher than 30.5 cm (1 ft) below grade (see SSAR Table 2.1-1).

- Probable Maximum Tsunami: The probable maximum tsunami flooding should be no higher than 30.5 cm (1 ft) below grade (see SSAR Table 2.1-1).

- Ground Water: The ABWR is intended to be compatible with ground water levels up to 61 cm (2 ft) below plant grade.

The staff concludes in Sections 3.4.1 and 3.4.2 that the ABWR standard plant has been adequately designed for the above bounding parameters.

2.6.3 Seismology Site Parameters

The staff's review of the seismic design of ABWR structures, systems and components (SSCs) is contained in Section 3.7 of this report. The bounding site parameter that was considered in the staff's evaluation is as follows:

- Vibratory Ground Motion: In SSAR Table 2.0-1, GE specifies the 0.3*g* peak ground acceleration (PGA) value for the high frequency anchor for the SSE response spectra. The SSE response spectra are constructed in accordance with RG 1.60, "Design Response Spectra for Seismic Design of Nuclear Power Plants," Revision 2.

Although the SSE PGA of 0.3*g* anchoring RG 1.60 design response spectra could generally be considered an adequate envelope for many sites in the Central and Eastern United States, the NRC staff knows that localized seismic activity exceeding this envelope cannot be ruled out categorically. Therefore, the staff will require that site-specific geological, geotechnical, and seismological factors be reviewed for each application to ensure that no site-specific seismic hazard will cause the RG 1.60 spectrum anchored at 0.30g to be exceeded.

2.6.4 Soil Properties Site Parameters

The staff's review of the soil properties considered in the seismic design of ABWR SSCs is contained in Sections 2.5.4 and 3.7 of this report. The bounding site parameters that were considered in the staff's evaluation are as follows:

- Minimum Shear Wave Velocity: The minimum embedment depth for the reactor building should be 26 m (85 ft). The minimum shear wave velocity of soil should be 305 m/sec (1,000 ft/sec).
- Liquefaction Potential: No liquefaction potential should exist for soils under and around all seismic Category I structures, including seismic Category I buried pipelines and electrical ducts.
- Minimum Bearing Capacity (demand): The minimum (demand) bearing capacity of soil should be 0.72 MPa (15 kips/ft^2).

As discussed in Section 3.7.1 of this report, the staff concludes that the ABWR standard plant design has adequately defined the above bounding parameters.

2.6.5 Precipitation (for Roof Design) Site Parameters

The bounding site parameters that were considered in the staff's evaluation are as follows:

- Maximum Rainfall Rate: 49.3 cm/hr (19.4 in./hr)
- Maximum Snow Load: 2.35 kPa (50 lb/ft^2)

The staff concludes in Section 3.8.4 of this report that the design of the ABWR SSCs to accommodate the precipitation site parameters is acceptable.

2.6.6 Design Temperature Site Parameters

The bounding site parameters that were considered in the staff's evaluation are as follows:

- Ambient 1% Exceedance Values:
  - Maximum: 37.8 °C (100 °F) dry bulb/25 °C (77 °F) coincident wet bulb and 26.7 °C (80 °F) noncoincident
  - Minimum: -23.3 °C (-10 °F)
- Ambient 0% Exceedance Values (Historical Limit):
  - Maximum: 46.1 °C (115 °F) dry bulb/26.7 °C (80 °F) coincident wet bulb and 27.2 °C (81 °F) noncoincident
  - Minimum: -40 °C (-40 °F)
- Emergency Cooling Water Inlet: 35 °C (95 °F)
- Condenser Cooling Water Inlet: $\leq$37.8 °C ($\leq$100 °F)

The staff concludes in Sections 9.4 and 9.5 of this report that the ABWR SSCs to accommodate the design temperature bounding site parameters are acceptable.

2.6.7 Atmospheric Dispersion (χ/Q)

The staff's review of the use of the χ/Q dispersion factors for the exclusion area boundary (EAB) and the low population zone (LPZ) is given in Chapter 15 of this report. The bounding site parameters considered in the staff's evaluation are listed in Table 2-2.

Table 2-2 EAB and LPZ Atmospheric dispersion characteristics (Chi/Q)

0-2 Hours	0-8 Hours	8-24 Hours	1-4 Days	4-30 Days
EAB	LPZ	LPZ	LPZ	LPZ
1.37E-3	1.95E-4	1.22E-4	4.69E-5	1.12E-5

The Table 2-2 bounding atmospheric relative concentration values for the LPZ (the 8-hour 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 design basis accident loss-of-coolant accident (LOCA) not to exceed the dose reference values given in 10 CFR Part 100 3E+3 MSv (300 rem) for the thyroid and 2.5E+2 MSv (25 rem) for the whole body. As shown in Table 2.0-1 of the SSAR, the 2-hour LPZ atmospheric relative concentration value and an annual average (8760 hours) concentration value were obtained by logarithmic extrapolation of these calculated LPZ bounding atmospheric relative concentrations.

The staff concludes in Section 15.4 of this report that GE's proposed bounding atmospheric relative concentrations (χ/Q) for the EAB and for the LPZ of the ABWR plant, in conjunction with the engineered safety features 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.

3 DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

3.1 General

The staff reviewed the information in SSAR Section 3.1 to verify that the ABWR standard plant meets the GDC of Appendix A to 10 CFR Part 50.

To review the design of structures, components, equipment, and systems, the staff relied heavily on industry codes and standards that represent accepted industry practice. The staff found those codes and standards cited in this report acceptable unless otherwise noted.

The staff evaluated the use of a single-earthquake design (i.e., elimination of operating basis earthquake (OBE)) for structures, systems, and components (SSCs). Originally, implementation of the OBE was included in applicable sections of the ABWR SSAR. After many discussions with the staff, GE stated that it might opt to use the single-earthquake design approach. In a letter to GE dated September 11, 1992, the staff gave preliminary guidance concerning what types of analyses and information would be required in the SSAR for the staff to approve design of SSCs for the ABWR without the OBE. This was Open Item 3.1-1 in the DFSER. The staff's evaluation of this issue is discussed in the following sections.

3.1.1 Elimination of Operating Basis Earthquake from Design Consideration

Appendix A to 10 CFR Part 100 requires, in part, that all SSCs of the nuclear power plant necessary for continued operation without undue risk to the health and safety of the public shall be designed to remain functional and within applicable stress and deformation limits when subject to an OBE. The NRC is proposing changes to Appendix A to Part 100 to redefine the OBE to such a level such that the function of the OBE can be satisfied without performing explicit response analyses. In addition, Appendix A to Part 100 requires that the maximum vibratory ground acceleration of the OBE be at least one-half the maximum vibratory ground acceleration of the SSE. When the OBE is redefined to an inspection level earthquake for the ABWR, the maximum vibratory ground acceleration of the OBE will be established at one-third of the maximum vibratory ground acceleration of the SSE.

There exist special circumstances for granting these exemptions from the requirements of Appendix A to Part 100 pursuant to 10 CFR 50.12. The purpose of designing SSCs necessary for continued operation without undue risk to the health and safety of the public to withstand an OBE is to ensure that these SSCs remain functional and within applicable stress and deformation limits when subjected to the effects of the OBE vibratory ground motion. However, Appendix A to Part 100 also requires that these SSCs are designed to withstand the SSE and remain functional. Thus, when these SSCs are designed to remain functional for the SSE, they will also remain functional at a lesser earthquake level (one-third the SSE) provided all design functions at the OBE are accounted for. The basis for selecting one-third the SSE as the earthquake level at which the plant will be required to shutdown and be inspected for damage was that at this level the likelihood of damage and the frequency of earthquakes occurring was judged to be low based on actual earthquake experience. It should be noted that certain design functions had been only verified for the OBE loads in the past. These design functions were the evaluations of (1) fatigue damage caused by earthquake cycles and (2) relative seismic anchor motions in piping systems. With the elimination of the OBE from design, these design functions would not have been explicitly verified. Consequently, for the ABWR these design functions will be verified in conjunction with the SSE using applicable stress and deformation limits as described in Section 3.1.1.2 of this report.

Accordingly, the special circumstances described by 10 CFR 50.12(a)(2)(ii) exist in that the regulation need not be applied in this particular circumstance to achieve the underlying purpose of the rule because GE has proposed acceptable alternative analysis methods that accomplish the intent of the regulation. On this basis, the staff concludes that the exemption is justified because the alternative analyses performed for the SSE and the need to perform an inspection of the plant following an earthquake at or above one-third the SSE accomplish the design objectives of the OBE design analyses.

3.1.1.1 Background

In SECY-90-016, "Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements," the staff requested the Commission's approval to decouple the level of the OBE ground motion from that of the safe-shutdown earthquake (SSE). The Commission approved the staff's position in its staff requirements memorandum (SRM) of June 26, 1990.

In SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs," the staff further requested that the Commission approve eliminating the OBE from the design of SSCs in both evolutionary and passive advanced reactors designs. The proposed amendment to 10 CFR Part 100, Appendix A, would allow, as an option, that the OBE be eliminated from design certification when the OBE is established at less than or equal to one-third the SSE. In this manner, the OBE serves the function as an inspection-level earthquake below which the effect on the health and safety of the public would be insignificant and above which the licensee would be required to shut down the plant and inspect for damage. The Electric Power Research Institute (EPRI) requested the elimination of the OBE from design and the Advisory Committee on Reactor Safeguards (ACRS) also recommended OBE elimination in its letter of April 26, 1990.

In SECY-93-087, the staff discussed the safety impact of eliminating the OBE as it pertains to civil structures, piping systems, and equipment seismic qualification. The staff made several recommendations to ensure that eliminating the OBE would not result in a significant decrease in the overall plant safety margin. In its July 21, 1993, SRM, the Commission approved and agreed with the following staff positions:

- Use two SSE events with 10 maximum stress cycles per event to account for earthquake cycles in the fatigue analyses of piping systems performed until the new guidance is issued; alternatively, the number of fractional vibratory cycles equivalent to that of 20 full SSE vibratory cycles may be used (but with an amplitude not less than one-third of the maximum SSE amplitude) when derived in accordance with Appendix D to IEEE 344-1987.

- The effects of anchor displacements in the piping caused by an SSE be considered with the Service Level D limit.

- Eliminate the OBE from the design of SSCs. When the OBE is eliminated from the design, no replacement earthquake loading should be used to establish the postulated pipe rupture and leakage crack locations.

- The mechanistic pipe break and high-energy leakage crack locations determined by the piping high-stress and fatigue locations may be used for equipment environmental qualification and compartment pressurization purposes.

- With the elimination of the OBE, two alternatives exist that will essentially maintain the requirements provided in IEEE 344-1987 to qualify equipment with the equivalent of five OBE events followed by one SSE event. Of these alternatives, the equipment should be qualified with five one-half SSE events followed by one full SSE event. Alternatively, a number of fractional peak cycles equivalent to the maximum peak cycles for five one-half SSE events may be used in accordance with Appendix D to IEEE 344-1987 when followed by one full SSE.

- The OBE will continue to be used as a threshold criterion for conducting inspections following an earthquake event.

The following sections contain the staff's evaluation of the commitments specified in the SSAR to ensure that appropriate measures and adequate safety margins are maintained when the OBE is eliminated from the design. The sections evaluate (1) American Society of Mechanical Engineers (ASME) Code Class 1, 2, and 3 components and core support structures, (2) concrete and steel structures, (3) equipment seismic qualification, and (4) preearthquake planning and postearthquake operator actions.

3.1.1.2 ASME Code Class 1, 2, and 3 Components and Core Support Structures

The dynamic analysis methods for seismic analyses of ASME Code Class 1, 2, and 3 components and core support structures in the ABWR use those methods described in the ABWR SSAR as approved by the NRC staff in Sections 3.9.2 and 3.12 of this report. The loads and load combinations used for evaluating ASME Code Class 1, 2, and 3 components and core support structures are also stated in the ABWR SSAR and discussed in Section 3.9.3.1 of this report. Conformance to existing staff guidelines that ensure the operability of safety-related equipment under SSE loading conditions are discussed in Section 3.10 of this report. Similarly, the function of the supported system has also been taken into account. As specified in Regulatory Guide (RG) 1.124, "Service Limits and Loading Combinations for Class 1 Linear-Type Components Supports," Revision 1, to ensure that systems — whose normal function is to prevent or mitigate consequences of events associated with the SSE — will operate adequately regardless of plant condition, the Code Level B service limits of Subsection NF or other justifiable limits approved by the staff have been used.

The elimination of the OBE from ASME Code Class 1, 2, and 3 components and core support structure design requires all current OBE design-related checks to be performed for the SSE. With regards to primary stress effects (seismic inertial stresses), the elimination of the OBE from Service Level B could have a potential impact on design in those cases where the load combination includes other dynamic loadings (i.e., operational transients) in Service Level B but not in Service Level D. The staff explored this possibility specifically for piping systems and found that when the OBE is established at one-third of the SSE, the load combinations with the SSE generally control the design. Therefore, for primary stresses in piping systems, the staff finds that eliminating the OBE from piping stress load combinations will not cause a reduction in existing safety margins because the load combination with the SSE loading is generally controlling.

For cyclic and secondary stress effects (e.g., fatigue and seismic anchor motion), the elimination of the OBE would have a direct impact on the current methods used to evaluate their adequacy in piping design. Because the cyclic (fatigue) effects of earthquake-induced motions in piping systems and the relative motion effects of piping anchored to equipment and structures at various elevations are currently evaluated only for OBE loadings, the elimination of the OBE from the load combination could lead to uncertainty concerning how these effects should be evaluated. The staff's evaluation of the ABWR guidelines discussed in the SSAR for treating these effects is discussed next.

Fatigue

In order to ensure adequate design considerations for the fatigue effects of earthquake cycles, GE needs to establish a bounding load definition and the number of earthquake cycles to account for the more frequent occurrences of lesser earthquakes and their aftershocks. In Section 3.7.3.2 of the SSAR, GE used a cyclic load basis for fatigue analysis of earthquake loading for ASME Code Class 1, 2, and 3 components and core support structures equal to two SSE events with 10 maximum stress cycles per event (20 full cycles of the maximum SSE stress range). This basis for analysis is acceptable because it is equivalent to the cyclic load basis of one SSE and five OBE events as currently recommended in SRP Section 3.9.2. Alternatively, an equivalent number of fractional vibratory cycles to that of 20 full SSE vibratory cycles may be used (but with an amplitude not less than one-third of the maximum SSE amplitude) when derived in accordance with Appendix D to IEEE 344-1987.

Seismic Anchor Motion

For the ABWR, the effects of displacement-limited, seismic anchor motions (SAM) that are due to an SSE are evaluated for safety-related ASME Code Class 1, 2, and 3 components and component supports to ensure their functionality during and following an SSE. The SAM effects include (but are not limited to) relative displacements of piping between building floors and slabs, at equipment nozzles, at piping penetrations, and at connections of small-diameter piping to large-diameter piping.

For piping systems, the effects of SAMs caused by an SSE are combined with the effects of other normal operational loadings that might occur concurrently as specified in Table 3.9.2 of the SSAR.

Piping Stress Limits

For ASME Code Class 1, 2, and 3 piping, GE needs to meet the design requirements in the 1989 Edition of the ASME Boiler and Pressure Vessel Code, Section III, Subsections NB, NC, and ND. In addition, the following changes and additions to paragraphs NB-3650, NC-3650, and ND-3650 will be used for piping systems when the OBE is eliminated from the design.

ASME Code Class 1 Piping Stress Limits

(a) For primary stress evaluation (NB-3654.2), earthquake loads are not required to be evaluated for consideration of Level B Service Limits for Eq. (9).

(b) For satisfaction of primary plus secondary stress intensity range (NB-3653.1), in Eq. (10), M_i should be either (1) the resultant range of all loads considering one-half the range of the SSE or (2) the resultant range of moment owing to the full range of the SSE alone, whichever is greater. The use of the SSE is intended to provide a bounding design for the cumulative effects of earthquakes of a lesser magnitude and is therefore to be included in consideration of Level B Service Limits for Eq. (10). A reduced range (with an equivalent number of fractional vibratory peak cycles) of the SSE moment may be used for consideration of Level B Service Limits (but with a range not less than one-third of the maximum SSE moment range).

(c) For satisfaction of peak stress intensity (NB-3653.2), the load sets developed in NB-3653.1 that are based on the preceding Position b should be used in calculating the peak stress intensity, S_p, and the alternating stress intensity, S_{alt}, for evaluating the fatigue effects and cumulative damage.

(d) For simplified elastic-plastic discontinuity analysis (NB-3653.6), if Eq. (10) cannot be satisfied for all pairs of load sets, then the alternative analysis as described in NB-3653.6 will be followed. For treatment of seismic anchor motion moments, the following condition should be satisfied in consideration of Level D Service Limits:

$$S_{SAM} = C_2 \frac{D_o}{2I} (M_i^* + M_i^{**}) \leq 6.0 S_m$$

where: S_{SAM} is the nominal value of seismic anchor motion stress

M_i^* is the same as M_i^* in Eq. (12)

M_i^{**} is the same as M_i in Eq. (10), except that it includes only moments caused by seismic anchor motion displacements that are caused by an SSE

The combined moment range ($M_i^* + M_i^{**}$) should be either (1) the resultant range of thermal expansion and thermal anchor movements plus one-half the range of the SSE anchor motion or (2) the resultant range of moment owing to the full range of the SSE anchor motion alone, whichever is greater.

ASME Code Class 2 and 3 Piping Stress Limits

(a) For consideration of occasional loads (NC/ND-3653.1), earthquake loads (i.e., inertia and seismic anchor motion) are not required for satisfying Level B Service Limits for Eq. (9).

(b) For consideration of thermal expansion or secondary stresses (NC/ND-3653.2), M_c in Eq. (10) is not required to include the moment effects of SAMs caused by an earthquake.

(c) For consideration of secondary stresses in Level D Service Limit (NC/ND-3655), the following condition will be satisfied:

$$S_s = i \frac{M_c^* + M_c}{z} \leq 3.0 S_h \qquad \text{Eq. (10b)}$$

where: M_c^* is the range of moments owing to SAMs due to an SSE

M_c is the range of moments owing to thermal expansion

The combined moment range ($M_c + M_c^*$) should be either (1) the resultant range of moments owing to thermal expansion plus one-half of the range of moments owing to the SSE anchor motions, or (2) the resultant range of moment owing to the full range of the SSE anchor motion alone, whichever is greater.

Upon reviewing these supplemental criteria to be used when the OBE is eliminated from the design of piping systems, the staff finds that the criteria are intended to maintain the existing design margins of the ASME Boiler and Pressure Vessel Code, Section III, although some criteria appear to be more stringent and others more relaxed. The net effect results in safety margins equivalent to that of the ASME Code, Section III, rules and provides a more controlled check of piping system stresses in those areas where actual failures of piping systems owing to seismic loadings have occurred. The staff concludes that the piping criteria for the ABWR meet the staff recommendations in SECY-93-087 for considering earthquake cycles in fatigue analyses and for evaluating the effects of anchor displacements in the piping caused by an SSE and are acceptable.

3.1.1.3 Pipe Break Postulation Without OBE

The staff recognizes that pipe rupture is a rare event that might only occur under unanticipated conditions, such as those that might be caused by possible design, construction, or operational errors; unanticipated loads or unanticipated corrosive environments. From observation of actual piping failures, the staff found that they generally occur at high stress and fatigue locations, such as at the terminal ends of a piping system at its connection to component nozzles. Currently, in accordance with SRP (NUREG-0800) Section 3.6.2, Revision 2, dated June 1987, pipe breaks are postulated in high-energy piping at locations of high stress and high fatigue usage factor. The load combination used in calculating the high stress and high usage factor includes normal and upset load conditions (i.e., pressure, weight, thermal, OBE, and other operational transient loadings).

From a historical viewpoint, the criteria for postulating high-energy breaks at specified locations were first introduced in the early 1970s. The basis for the mechanistic approach for selecting pipe break locations was derived from the premise that although pipe breaks could result from random events induced by unanticipated conditions, the failure mechanism and the expected location of failure would likely be caused by local conditions of high stress or high fatigue in the piping. To ensure that a sufficient number of pipe breaks would be postulated, the staff recommended that breaks be postulated for a wide spectrum of events to envelope the uncertainties of unanticipated failure mechanisms. Breaks were postulated at terminal ends of the piping, at high-stress and high-fatigue locations, and, as a minimum, at two additional intermediate locations when the stresses were below the high-stress threshold limit. The resulting criteria, which were incorporated in SRP Section 3.6.2, resulted in many postulated pipe break locations and caused the installation of numerous pipe rupture mitigation devices in nuclear plants.

In the mid-1980s, the NRC's Executive Director for Operations (EDO) initiated a comprehensive review of nuclear power plant piping to identify areas where changes to the piping requirements could improve the licensing process as well as the safety and reliability of nuclear power plants. The NRC's Piping Review Committee (PRC) in an integrated effort with the nuclear industry under the Pressure Vessel Research Council conducted a comprehensive study of piping criteria, including the mechanistic pipe break postulation guidelines. The PRC found that when an excessive number of pipe rupture mitigation devices (i.e., pipe whip restraints and jet impingement shields) are installed on high-energy piping systems, the potential exists for piping systems to be overly constrained. This condition was found in several nuclear plants in which massive pipe restraints adversely affected the ability of the high temperature piping to freely expand during normal plant operation. The PRC also found through numerous dynamic tests and field observations of non-seismically designed piping systems that had undergone high seismic loadings that butt-welded piping possesses an inherent ability to withstand large seismic inertial loadings without failure.

As a result of the PRC's effort, the NRC staff recognized that the mechanistic pipe rupture criteria for selecting locations of pipe breaks resulted in an excessive number of pipe rupture mitigation devices that could hinder the normal operation of the plant and that may not contribute significantly to the overall safety of the plant. Accordingly, the SRP was revised to reduce the number of postulated pipe breaks by (1) eliminating the need to postulate pipe breaks at the two arbitrary intermediate locations and (2) providing a leak-before-break (LBB) approach in lieu of postulating pipe breaks when the system and material specific information is adequate to justify its application.

Recent dynamic pipe tests, conducted by the EPRI and NRC, have demonstrated that the piping can withstand seismic inertial loadings higher than an SSE without rupturing. Thus, the staff believes the likelihood of a pipe break in a seismically designed piping system owing to an earthquake magnitude of one-third SSE is remote. Operating experience has shown that pipe breaks are more likely to occur under conditions caused by normal operation (e.g., erosion-corrosion, thermal constraint, fatigue, and operational transients).

On the basis of the previous discussion, the staff concludes that no replacement earthquake loading should be used to establish postulated pipe break and leakage crack locations. Instead, the criteria for postulating pipe breaks and leakage cracks in seismically designed, high- and moderate-energy piping systems should be based on factors attributed to normal and operational transients only. SSAR Sections 3.6.1 and 3.6.2 conform to this staff position of pipe break postulation. However, for establishing pipe breaks and leakage cracks caused by fatigue effects, the calculation of the cumulative usage factor will continue to include seismic cyclic effects. The revised criteria are intended to ensure that breaks and leakage cracks are postulated to occur at the most likely locations and to reduce the number of pipe rupture mitigation devices (e.g., pipe whip restraints and jet impingement shields) that might hinder plant operation without providing a compensatory level of safety.

The elimination of earthquake loads in the following revised pipe break criteria is justified, in part, on the fact that the equipment environmental qualification and compartment pressurization analyses for the ABWR are based on a worst-case break assumption in each compartment and are not postulated at mechanistic break locations. In addition, GE has committed in SSAR Section 6.6.7.2 to a monitoring program for erosion-corrosion that provides assurances that procedures or administrative controls are in place to ensure that the NUMARC program (or another equally effective program) is implemented and the structural integrity of all high-energy (two-phase as well as single-phase) carbon-steel systems is maintained as discussed in Generic Letter (GL) 89-08 and NUREG-1344, "Erosion/Corrosion-Induced Pipe Wall Thinning in U.S. Nuclear Power Plants," April 1989.

Consistent with this staff finding, the guidelines provided in SRP Section 3.6.2, Branch Technical Position MEB 3-1, "Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment," have been revised for the ABWR as follows:

B.1.b.(1).(a): Footnote 2 should read, "For those loads and conditions in which Level A and Level B stress limits have been specified in the Design Specification (excluding earthquake loads)."

B.1.b.(1).(d): "The maximum stress as calculated by the sum of Eqs. (9) and (10) in Paragraph NC-3652, ASME Code, Section III, considering those loads and conditions thereof for which level A and level B stress limits have been specified in the system's Design Specification (i.e., sustained loads, occasional loads, and thermal expansion) excluding earthquake loads should not exceed $0.8(1.8\ S_h + S_A)$."

The ABWR criteria are consistent with this staff position for postulating pipe breaks and cracks and are acceptable.

3.1.1.4 Concrete and Steel Structures

The current design practice for considering OBE and SSE ground motion effects in the seismic design of nuclear plant structures was established in the 1960s with conceptual goals of (a) maintaining continued plant operation without damage to the structures for OBE level earthquakes and (b) ensuring safe shutdown of plant and maintaining the plant in a safe-shutdown condition during and after the occurrence of an SSE. To achieve these goals, the structural responses are designed at or below the material yield stresses to preclude the onset of plastic deformation for load combinations owing to accident conditions plus the SSE. For load combinations owing to operating conditions plus the OBE, stresses are limited at 1/2 to 5/8 yield stress. The current load combinations provided in SRP Section 3.8 were developed from this design philosophy.

For seismic Category I steel structures, the staff's guidance on load combinations is provided in SRP Section 3.8.4. The staff's review of the controlling load combinations finds that, in general, the load combinations with the SSE control the design of steel structures although there may be specific cases where the load combinations with the OBE control the design. Similarly, an examination of the pertinent load combinations for concrete structures, including the containment structure, should lead to the same conclusion that the OBE loads, in most cases, do not control the outcome of the structural design.

In the design of the containment, the staff reviewed the extent to which the elimination of the OBE from the load combinations would lead to a reduction of the safety margin. An examination of the nuclear structural design practice and the SRP load combination equations, however, show that the major dynamic load for the overall design of structures is either the OBE or the SSE. All other potential dynamic loads are conservatively accounted for in the definition of equivalent dead and live loads or only produce local effects that are handled by local reinforcement details. Therefore, the staff concludes that no reduction in safety margins of concrete and steel structures results from the elimination of the OBE as a design requirement.

For the ABWR, the following criteria for structures are used to ensure that when the OBE is eliminated from design, the structures will continue to be designed appropriately for earthquake effects.

SSE Relative Displacements Between Structures

In Appendix 3A to the SSAR, the seismic response (building displacements, structural member forces, floor response spectra (FRS), etc.) of the reactor building (RB) is discussed. GE has considered the through-soil, structure-to-structure interaction effect under SSE loading in the analyses of ABWR structures, including the control building, ultimate heat sink pump house, radwaste building, and turbine building. Therefore, the staff concludes that the effects of through-soil, structure-to-structure interaction under SSE loadings for all structures housing seismically designed piping have been adequately considered under SSE loadings to establish the relative displacements between buildings (seismic anchor movement for piping systems).

Seismic Instrumentation

GE committed in SSAR Section 3.7.4.4 to placing seismic instrumentation in the free field so that the control room operator can be immediately informed through the event indicators when the response spectral level and the cumulative absolute velocity (CAV) experienced at this location exceed the shutdown level and can take the necessary actions. The staff concludes that the ABWR meets the staff's recommendations for pre-earthquake planning with respect to the location of seismic instrumentation.

Use of Regulatory Guides 1.143 and 1.27

The staff guidelines in RG 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants," Revision 1, and in RG 1.27, "Ultimate Heat Sink for Nuclear Power Plants," Revision 2, recommend a seismic design of radwaste buildings and ultimate heat sink features based on the OBE. With the elimination of the OBE, GE committed to designing these structures and features to withstand the SSE. The structural design criteria, using the SSE loading, use the corresponding loads and load combinations provided in SRP Section 3.8.4. The staff finds that designing these structures and features to the SSE provides a bounding design comparable to that recommended in the regulatory guides and is acceptable. The staff will review alternative methods to ensure the seismic adequacy of these structures and features on a case-by-case basis.

3.1.1.5 Equipment Seismic Qualification

The proposed elimination of OBE from explicit design consideration affects different aspects of equipment qualification in different manners. In the area of equipment qualification, the requirements in the regulations

(10 CFR Parts 50 and 100) are interpreted by the staff through Section 3.10 of the SRP, which deals with seismic and dynamic qualification of mechanical and electrical equipment.

When the equipment qualification is performed by analysis, the acceptance criteria are derived from the ASME Code. The effect of eliminating the OBE on equipment qualification by analysis should be negligible. It is well known that mechanical equipment such as pumps and valves are, in general, seismically rugged when adequately anchored and that their operability limits are generally established through maximum permissible moments and forces or tolerance limits based on available clearances that are controlled by the SSE rather than the OBE. Therefore, for mechanical equipment, elimination of OBE from qualification analysis should not reduce any safety margin. Also, some electrical equipment are allowed to be qualified by analysis that requires demonstration that five OBE events followed by one SSE event do not cause its failure to perform safety functions. With the elimination of OBE, analysis checks for fatigue effects may be performed at a fraction of the SSE (e.g., 50 cycles at one-half of the SSE peak amplitude or 150 cycles at one-third of the SSE peak amplitude).

When equipment qualification for seismic loadings is performed by analysis, testing, or a combination of both, the staff recommends the use of the IEEE 344-1987 as endorsed in RG 1.100, "Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants," Revision 2. For analysis, the selection of the level of service limit for different loading combinations should ensure the functionality of the equipment during and following an SSE. For testing, IEEE 344-1987 has detailed requirements for performing seismic qualification using five OBE events followed by an SSE event. Where complex mathematical models are based solely on calculated structural parameters, verification testing should be performed.

With the elimination of the OBE, and in order to maintain the equivalent qualification requirements provided in IEEE 344-1987 to qualify equipment with the equivalent of five OBE events followed by one SSE event, the staff recommended in SECY-93-087 that equipment be qualified with five half SSE events followed by one full SSE event. Alternatively, the staff recommended that a number of fractional peak cycles equivalent to the maximum peak cycles for five half SSE events may be used in accordance with Appendix D to IEEE 344-1987 when followed by one full SSE. In Section 3.7.3.2 of the SSAR, GE committed to these staff recommendations as stated in SECY-93-087 and, thus, the criteria proposed for equipment seismic qualification when the OBE is eliminated from design are acceptable.

3.1.1.6 Pre-Earthquake Planning and Post-Earthquake Operator Actions

The design certification of the ABWR, using a single-earthquake SSE design, is predicated on the adequacy of pre-earthquake planning and post-earthquake damage inspections that are to be implemented by the COL applicant.

The COL applicant will be required to demonstrate to the NRC staff as a part of its application the procedures it plans to use for pre-earthquake planning and post-earthquake actions. For design certification, the NRC staff reviewed the criteria developed by the EPRI in EPRI Reports EPRI NP-5930, EPRI NP-6695, and EPRI TR-100082 for evaluating the need to shut down the plant following an earthquake and the commitments for the ABWR for ensuring that these actions can be taken as provided in SSAR Section 3.7.4.

3.1.1.7 EPRI NP-5930

The staff finds from its review of EPRI NP-5930 that this report is adequate for and may be used by the COL applicant with the following exceptions:

1. A free field instrument must be used for determining the CAV and the spectral acceleration level.

2. The response spectrum check is as follows:

   The 5-percent damped ground response spectrum for the earthquake motion at the site exceeds (1) one-third the corresponding SSE response spectral acceleration between 2 and 10 Hz or it exceeds a spectral acceleration of 0.20g between 2 and 10 Hz, whichever is greater, or (2) one-third the corresponding SSE response spectral velocity between 1 and 2 Hz or a velocity of 15.24 cm/sec (6 in/sec) between 1 and 2 Hz, whichever is greater.

3. The licensee shall consider as sufficient evidence to shut down the plant the simultaneous exceedance of the 5-percent damped ground response spectrum enumerated in Item 2 and the CAV exceedance of 0.16 g-sec for any one frequency on any one component of the free field ground motion. The CAV shall be determined in accordance with EPRI Report TR-100082. Also, any evidence of significant damage observed during the plant walkdown in accordance with EPRI Report NP-6695 recommendations shall be sufficient cause for plant shutdown.

4. The instrumentation installed at the nuclear power plant shall be capable of on-line digital recording of all three components of the ground motion and of converting the recorded (digital) signal into the standardized CAV and the 5-percent damped response spectrum. The digitizing rate of the time history of the ground motions shall be at least 200 samples per second and the band-width shall be at least from 0.20 Hz to 50 Hz. The pre-event memory of the instrument shall be sufficient to record the onset of the earthquake.

5. The system must be capable of routinely calibrating the response spectrum check of 0.20g. Also, the CAV of 0.16 g-sec should be calibrated with a copy of the October 1987, Whittier, California earthquake or an equivalent calibration record provided for this purpose by the manufacturer of the instrumentation. In the event that an actual earthquake has been recorded at the plant site, the above calibration shall be performed to demonstrate that the system was functioning properly at the time of the earthquake.

In SSAR Section 3.7.4, GE committed to these guidelines. This is acceptable.

3.1.1.8 EPRI NP-6695

The staff finds from its review of EPRI NP-6695 that this report may be used by the COL applicant with the following exceptions:

Section 3.1, Short-Term Actions

Item 3, "Evaluation of Ground Motion Records"

Within four hours, the licensee must determine if the shutdown criterion has been exceeded. After an earthquake has been recorded at the site, the licensee must provide a response spectrum calibration record and CAV calibration record to demonstrate that the system was functioning properly.

Item 4, "Decision on Shutdown"

Exceedance of the EPRI criterion as amended by the NRC or observed evidence of significant damage as defined by EPRI NP-6695 shall constitute a condition for mandatory shutdown unless conditions prevent the licensee from accomplishing an orderly shutdown without jeopardizing the health and safety of the public.

Add Item 7, "Documentation"

The licensee must record the chronology of events and control room problems while the earthquake evaluation is in progress.

Section 4.3, "Guidelines" (p. 4-3)

Because earthquake-induced vibration of the reactor vessel could lead to changes in neutron fluxes, the licensee need to promptly check the neutron flux monitoring instruments, which indicate whether the reactor is stable. Therefore, this check should be added to the checks listed in this section.

Section 4.3.4, "Pre-Shutdown Inspection"

Exceeding the EPRI criterion or evidence of significant damage should constitute a condition for mandatory plant shutdown, as the staff stated in its comment on Section 3.1, Item 4, "Decision on Shutdown."

Section 4.3.4.1, "Safe-shutdown equipment" (p. 4-7)

In addition to the safe shutdown systems on this list, containment integrity must be maintained following an earthquake. Since the containment isolation valves may have malfunctioned during the earthquake, inspection of the containment isolation system is necessary to ensure continued containment integrity.

3.1.1.9 Conclusions

On the basis of the evaluation of the changes to the existing seismic design criteria previously discussed, the staff concludes that eliminating the OBE from the design of SSCs in the ABWR standard plant will not reduce the level of safety provided in current regulatory guidelines for seismic design. On the contrary, the changes enhance safety by refocusing current design requirements to emphasize those areas where failure modes are more likely to occur and by precluding the need for seismic design requirements that do not significantly contribute to the overall safety of the plant. The SSAR includes this information in Sections 3.6.1 and 3.6.2 and Tables 3.9-1 and 3.9-2. The staff further concludes that the elimination of the OBE from the design of SSCs in the ABWR standard plant meets the Commission-approved staff recommendations in SECY-93-087 and is acceptable. On the basis of these conclusions, DFSER Open Item 3.1-1 is resolved.

3.1.2.5.2 Fracture Prevention of Containment Boundary

The ABWR primary containment vessel is a reinforced concrete structure with ferritic parts (the removable head, personnel locks, equipment hatches, and penetrations) made of materials that will have a nil-ductility transition temperature (RT_{NDT}) of at least 17 °C (30 °F) below the minimum service temperature. GDC 51 is only applicable to parts of containment that are made of ferritic materials.

The staff requested that GE clarify the applicability of GDC 51 because in the original SSAR it seemed that GE intended to apply GDC 51 to the concrete parts of the containment (Question (Q) 251.12). Subsequently, GE responded that GDC 51 is applicable to the removable drywell head, personnel locks, equipment hatches, and penetrations that are made of ferritic materials and revised SSAR Section 3.1.2.5.2 to reflect this clarification. The staff reviewed the revision and concluded in the DFSER that the applicable revision has satisfactorily complied with the requirements of GDC 51 because ferritic parts in the concrete structure will be made of materials that will have RT_{NDT} of at least 17 °C (30 °F) below the minimum service temperature. This ensures that the ferritic materials in the containment structure will not undergo brittle fracture and the probability of a rapidly propagating fracture will be minimized. This is acceptable.

3.2 Classification of Structures, Systems, and Components

3.2.1 Seismic Classification

GDC 2 requires, in part, that nuclear power plant SSCs important to safety be designed to withstand the effects of earthquakes without loss of capability to perform their safety functions. Certain of these features that are safety-related are necessary to ensure (1) the integrity of the reactor coolant pressure boundary (RCPB), (2) the capability to shut down the reactor and maintain it in a safe-shutdown condition, and (3) the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures that are comparable to the guidelines in 10 CFR Part 100. The earthquake for which these safety-related plant features are designed is defined as the SSE in Appendix A to 10 CFR Part 100. The SSE is based on an evaluation of the maximum earthquake potential and is that earthquake that produces the maximum vibratory ground motion for which SSCs are designed to remain functional. Those plant features that are designed to remain functional, if an SSE occurs, are designated seismic Category I in RG 1.29, "Seismic Design Classification," Revision 3. In addition, regulatory Position C.1 in RG 1.29 states that the pertinent quality assurance (QA) requirements of Appendix B to 10 CFR Part 50 should be applied to all activities affecting the safety-related functions of seismic Category I SSCs. The staff reviewed the ABWR SSAR in accordance with SRP Section 3.2.1, which references RG 1.29.

The SSC and equipment of the ABWR standard plant that are required to be designed to withstand the effects of an SSE and remain functional are identified as seismic Category I in SSAR Table 3.2-1. In addition, piping and instrumentation diagrams (P&IDs) in the SSAR identify the interconnecting piping and valves and the boundary limits of each system identified as seismic Category I. In Section 3.2.1 of the DSER (SECY-91-153), the staff reported that the seismic classifications of the main steamline (MSL) between the seismic interface restraint and the turbine stop valve and MSL branch lines up to and including the first valve in the branch line were still under review. This was DSER Outstanding Issue 3. The staff's review also included quality group (QG) and safety class (SC) designations and QA requirements for these same components. The relationship between QG and SC is discussed in Section 3.2.2 of this report. The resolution of DSER (SECY-91-153) Outstanding Issue 3 is discussed below.

The ABWR design eliminates the main steam isolation valve leakage control system for the ABWR plant design. Instead, the design relies on the use of an alternative leakage path that takes advantage of the large volume and surface area in the main steam piping, main steam drain lines, turbine bypass line, and condenser to hold up and plate out the release of fission products following core damage. In this manner, the alternative leakage path and condenser are used to mitigate the consequences of an accident and are required to remain functional during and after an SSE.

For this reason, the staff position, which was discussed in Section II.E of SECY-93-087 and was approved by the Commission in its SRM dated July 21, 1993, is that the main steam piping beyond the outermost isolation valve up to the seismic interface restraint and connecting branch lines up to the first normally closed valve be classified as QG B (SC 2) and seismic Category I. The MSL from the seismic interface restraint up to but not including the turbine stop valve (including branch lines to the first normally closed valve) will be classified as QG B and inspected in accordance with the applicable portions of ASME Section XI. This portion of the steamline may be classified as non-seismic Category I if it has been analyzed using a dynamic seismic analysis method to demonstrate its structural integrity under SSE loading conditions. However, all pertinent QA requirements of Appendix B to 10 CFR Part 50 are applicable to ensure that the quality of the piping material is commensurate with its importance to safety during normal operational, transient, and accident conditions. To ensure the integrity of the remainder of GE's proposed alternative leakage path, the staff position is that (1) the main steam piping between the turbine stop valve and the turbine inlet, the turbine bypass line from the bypass valve to the condenser, and the main steam drain line from the first valve to the condenser are not required to be classified as safety-related or as seismic Category I, but should be analyzed using a dynamic seismic analysis to demonstrate their structural integrity under SSE loading conditions, and (2) the condenser anchorage shall be seismically analyzed to demonstrate that it is capable of sustaining the SSE loading conditions without failure.

The seismic interface restraint shall provide a structural barrier between the seismic Category I portion of the MSL in the RB and the non-seismic Category I portions of the MSL in the turbine building. The seismic interface restraint shall be located inside the seismic Category I building. The classification of the MSL in the turbine building as non-seismic Category I is needed for consistency with the classification of the turbine building. Therefore, the staff positions related to the quality and safety guidelines imposed on the MSL from the outermost isolation valve up to the turbine stop valve are equivalent to the staff guidelines in SRP Section 3.2.2, Appendix A to 10 CFR Part 50, and RG 1.29.

In response to staff requests for additional information related to this issue, Amendment 23 to the ABWR SSAR, GE revised SSAR Table 3.2-1 to respond to the staff's concerns. Table 3.2-1 of the SSAR contains the following classifications:

- The MSL, including supports, from the outermost isolation valve to the seismic restraint, including branch lines up to the first valve is SC 2, QG B; QA will be in accordance with Appendix B to 10 CFR Part 50 and seismic Category I.

- The MSL from the seismic restraint to the turbine stop valve, including branch lines up to the first valve is QG B; QA will be in accordance with Appendix B to 10 CFR Part 50 and non-seismic Category I, but dynamically analyzed for the SSE. Although not explicitly stated in SSAR Table 3.2-1, the commitment to QG B requires inservice inspections in accordance with applicable portions of ASME Section XI.

- The turbine stop valve, the MSL from the turbine stop valve to the turbine, the turbine bypass valve, the turbine bypass line from the bypass valve to the condenser and the main steam drain line from the first valve to the condenser, are all non-safety class, non-seismic Category I, but dynamically analyzed for the SSE.

- The condenser is non-safety class, non-seismic Category I, but the condenser anchorage is seismically analyzed for the SSE. GE considers piping inlets to the condenser as anchor points.

The commitment to dynamically analyze the above components for the SSE is in Section 3.2.5.3, "Main Steamline Leakage Path" of the SSAR. The components are designed by using appropriate dynamic seismic analyses to withstand the SSE design loads in combination with other appropriate loads, within the limits specified. The mathematical model for the dynamic seismic analyses of the main steamlines and the branch line piping includes the turbine stop valves and piping to the turbine casing. The dynamic input for the analysis and design of the main steamlines are derived from a time history analysis (or an equivalent method) of the turbine building as described in SSAR Section 3.7.

In the DFSER, the staff noted that, to demonstrate the structural integrity of the main steam piping under SSE loading conditions, a dynamic analysis method should be used for this portion of the main steam piping. In addition, GE should discuss how the turbine building response spectra input to the main steam piping analysis will be generated when, as discussed in the following paragraph, no dynamic analysis method of the turbine building is proposed. The staff position as delineated in SRP Section 3.7.2.5 for the generation of floor response spectra (FRS) is that the development of the FRS is acceptable if a time history approach is used. Alternative methods, other than the time history approach, used for generating FRS shall be submitted to the staff for review and approval on a case-by-case basis. This was DFSER Open Items 3.2.1-1 and 3.7.2-6. The resolution of this issue is discussed in the "Turbine Building" portion of Section 3.7.2 in this report.

Note f in SSAR Table 3.2-1 and SSAR Sections 3.7.2.8 and 3.7.3.13 state that equipment, structures, and piping in the ABWR that are non-seismic Category I but that could damage seismic Category I items if their structural integrity failed, are analyzed and designed to ensure their integrity is maintained under seismic loading from the SSE. At the interface between seismic and non-seismic Category I piping systems, the seismic analysis for the seismic Category I system will be extended to either the first anchor point in the non-seismic system or to sufficient distance in the non-seismic system so as not to degrade the validity of the seismic Category I analysis. In the DFSER, the staff noted that this commitment is in conformance with RG 1.29. However, as a part of the resolution of this issue, the staff noted that the turbine building and any other applicable structure or equipment should be seismically analyzed for the SSE (RG 1.60 ground response spectrum anchored to 0.3g peak acceleration) to ensure that an earthquake will not adversely affect the structural integrity of the main steam piping, bypass line, valves, and instruments mounted on these pipes, and the main condenser. In Amendment 20 to the SSAR, GE added Section 3.7.3.16, which described a static, in lieu of dynamic, analysis for the turbine building. This proposed analysis was in accordance with the Uniform Building Code (UBC) Zone 2A, but was substantially less conservative than the specified SSE for which the main steam and bypass lines should be analyzed. The staff asked GE to provide a clear set of criteria for the turbine building, which is designed to UBC criteria, so that it will neither suffer no loss of function at the specified SSE nor adversely affect safety-significant piping, pipe-mounted equipment, or the condenser itself. This issue was DFSER Open Items 3.2.1-2 and 3.7.2-7. The resolution of this issue is discussed in the "Turbine Building" portion of Section 3.7.2 in this report.

In addition, the staff's position is that plant-specific walkdowns of non-seismically designed SSCs overhead, adjacent to, and attached to the alternative leakage path (i.e., the main steam piping, the main steam drain lines to the condenser, the bypass line to the condenser, and the main condenser) be conducted by the COL applicant before commercial operation to assess potential failures. The walkdowns, which should identify potential failure modes, will provide high confidence that the alternative leakage paths in ABWRs can retain their structural integrity during and following an SSE. In the DFSER, the staff noted that these walkdowns should be performed as a part of an inspection, test, analysis, and acceptance criteria (ITAAC) for verification of non-seismic/seismic interaction. This was DFSER Open Item 3.2.1-3 and DFSER Confirmatory Item 14.1.3.3.3.8-2. Upon further consideration, the staff found that the design detail and as-built and as-procured information for non-seismically designed SSCs in the turbine building as they affect the alternate leakage function of the main steam, bypass, and drain lines, and main condenser are not required for design certification, and the spacial relationship between these SSCs and the main steam piping, bypass, and drain lines, and the main condenser should be assessed to ensure compliance with GDC 2. Subsequently, GE revised the SSAR and added SSAR Section 3.2.5.3, which contains a commitment to perform plant-specific walkdowns consistent with the previously stated staff position. The commitment in SSAR Section 3.2.5.3 to perform walkdowns is acceptable. Therefore, DFSER Open Item 3.2.1-3 and DFSER Confirmatory Item 14.1.3.3.3.8-2 are resolved. The resolution of this issue is also discussed in Section 3.12.3.8 of this report.

The staff concludes that these ABWR commitments in SSAR Section 3.2.5.3, including the plant-specific walkdowns, provide reasonable assurance that the main steam piping from the outermost isolation valve up to the turbine, including the turbine stop valve and branch lines to the first normally closed valve, the turbine bypass line up to the condenser, the main steam drain line from the first valve to the condenser, and the condenser, will retain their structural integrity during and following an SSE. Therefore, from the structural integrity standpoint, the elimination of the MSIV leakage control system in the ABWR is acceptable, and the ABWR design meets the Commission-approved staff recommendations related to the classification of main steamlines in BWRs. The staff's evaluation of the radiological analysis for this issue is discussed in Section 15.4.4.2 of this report. On the basis of this evaluation, Outstanding Issue 3 in the DSER (SECY-91-153) is resolved.

On the basis of its review of SSAR Table 3.2.1, the applicable P&IDs, and other relevant information in the SSAR, including the previous discussion related to the elimination of the MSIV leakage control system, the staff concludes that the safety-related SSCs of the ABWR are properly classified as seismic Category I in accordance with RG 1.29 or an acceptable equivalent to ensure the structural integrity of the alternative leakage path. This constitutes an acceptable basis for satisfying, in part, GDC 2.

3.2.2 System Quality Group Classification

GDC 1 requires that nuclear power plant SSCs important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed. This requirement is applicable to both pressure-retaining and non-pressure-retaining SSCs that are part of the RCPB and other systems important to safety, when reliance is placed on these systems to (1) prevent or mitigate the consequences of accidents and malfunctions originating within the RCPB, (2) permit shutdown of the reactor and maintain it in a safe-shutdown condition, and (3) retain radioactive material.

In addition to the seismic classifications discussed in Section 3.2.1 of this report, SSAR Table 3.2-1 identifies the SC, QG, and QA requirements necessary to satisfy the requirements of GDC 1 for all safety-related SSCs and equipment. Applicable P&IDs identify the classification boundaries of interconnecting piping and valves. The staff reviewed Table 3.2-1 and the P&IDs in accordance with

SRP Section 3.2.2, which references RG 1.26, "Quality Group Classification and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," Revision 3, as the principal document used in the staff review for identifying on a functional basis the pressure-retaining components of those systems important to safety as NRC QG A, B, C, or D. Conformance of ASME Code, Section III, Class 1 components that are part of the RCPB to 10 CFR 50.55a is discussed in Section 5.2.1.1 of this report. These RCPB components are designated in RG 1.26 as QG A; certain other RCPB components that meet the exclusion requirements of 10 CFR 50.55a(c)(2) are classified QG B.

GE uses American Nuclear Society (ANS) SC 1, 2, 3, and non-nuclear safety (NNS) as defined in American National Standards Institute (ANSI)/ANS 52.1-1983 for the classification of system components as an alternative method of meeting RG 1.26. SSAR Tables 3.2-2 and 3.2-3 provide a correlation between (1) ABWR SC 1, 2, 3 and NNS, (2) NRC QG A, B, C, and D in RG 1.26, and (3) ASME Code Section III classes. The relationship between the three methods of classification for pressure-retaining components in the SSAR is shown below.

NRC QG	ABWR SC	Section III
A	1	1
B	2	2
C	3	3
D	NNS	-

All pressure-retaining components classified as QG A, B, or C are constructed in accordance with ASME Code, Section III, Class 1, 2, or 3 rules, respectively. Construction as defined in ASME Code, Section III, Subsections NB/NC/ND-1110(a) and used herein is an all-inclusive term comprising materials, design, fabrication, examination, testing, inspection, and certification required in the manufacture and installation of components. Components classified as QG D are designed to the applicable standards identified in SSAR Table 3.2-3.

Because the staff has not endorsed ANSI/ANS 52.1-1983, it cannot rely on those safety classifications for determining the acceptability of non-pressure-retaining SSCs. Therefore, in performing the review of SSAR Table 3.2-1 for non-pressure-retaining components, the staff concentrated on an evaluation of QA in accordance with Appendix B to 10 CFR Part 50 and seismic classifications.

In Section 3.2.2 of the DSER (SECY-91-153), the staff stated its concern about the QA requirements of the new and spent fuel storage racks and the storage container for defective fuel. This was Outstanding Issue 2 in SECY-91-153. The staff's position is that the new and spent fuel storage racks are important to safety and, as a minimum, should meet the applicable QA requirements of Appendix B to 10 CFR Part 50, in addition to being classified as seismic Category I. This position is consistent with RG 1.29. The storage containers for defective fuels are designed to the same QA requirements as the fuel storage racks. However, because these storage containers are stored in the seismic Category I fuel storage racks, they do not have to be seismic Category I. Note e in SSAR Table 3.2-1 defines QA Category E, which is applicable to the new and spent fuel storage racks and the storage container for defective fuel. Category E states that elements of Appendix B to 10 CFR Part 50 are generally applied to these components, commensurate with the importance of the component's function. This commitment satisfies the applicable guidelines in RG 1.29 and is acceptable. Therefore, DSER Outstanding Issue 3 is resolved.

In Section 3.2.2 of the DSER (SECY-91-153), the staff reported a concern about the QG classification of the containment spray piping, including the spargers, within the outermost containment isolation valve. This was identified as Outstanding Issue 1 in SECY-91-153. SSAR Table 3.2-1, Item E1.4, classifies this piping as SC 2 and QG B. The staff agrees with these classifications. However, SSAR Figure 5.4-10 originally classified this portion of piping as SC 3 and QG C. In response to Outstanding Issue 1, the SSAR, Figure 5.4-10, Sheets 2 and 7, contains the classifications which agree with those in Table 3.2-1. This is acceptable and resolved DSER Outstanding Issue 1.

The reactor internal pump (RIP) recirculation motor cooling subsystem is classified by GE as SC 2 and seismic Category I. This subsystem is connected to the RIP motor casing, which is classified as SC 1 and is part of the RCPB. In response to a staff request for GE's basis for the SC 2 classification, GE stated that in the event of a postulated failure of the RMC subsystem piping, the small annulus between the outside diameter of the RIP shaft and the inside diameter of the stretch tube in the RIP assembly acts as a flow restrictor to limit the flow of water from the reactor pressure vessel to a low level. The reactor can be shut down and cooled in an orderly manner, and reactor coolant makeup can be provided by a normal make up system (e.g., control rod drive (CRD) return or reactor core isolation cooling (RCIC) system). Therefore, per 10 CFR 50.55a(c)(2), the RMC subsystem can be classified as SC 2. This basis is acceptable.

The subject of safety and QG classification of the reactor water cleanup (CUW) system has been included as a part of several meetings between the Advisory Committee on

Reactor Safeguards (ACRS), GE, and the staff. In the SSAR Table 3.2-1 and Figure 5.4-12, "CUW System P&ID," the piping up to and including the outermost containment isolation valve is classified as Safety Class 1 and QG A. Beyond the outside isolation valve, the piping is non-nuclear safety, non-seismic, but QG C. In accordance with the SSAR Table 3.2-3, QG C components are designed to ASME Section III, Subsection ND (ASME Class 3) rules. Therefore, from the structural integrity standpoint, this portion of the CUW piping is equivalent to Safety Class 3. These criteria are consistent with the acceptance criteria in SRP Section 5.4.8.II.3, and are acceptable. The transition from QG A to C at the outside isolation valve meets Position 2.c in RG 1.26, and is acceptable because the two containment isolation valves in this system are motor operated and are designed, qualified, and tested in accordance with the criteria discussed in Sections 3.9.2.2, 3.9.3, 3.9.6.2.2, and 3.10 of this report.

The staff concludes that the QG classifications of all pressure-retaining and non-pressure-retaining SSCs important to safety that are identified in SSAR Table 3.2-1 are in conformance with RG 1.26, and with staff positions on previously licensed boiling-water reactor (BWR) plants and are acceptable. Table 3.2-1, in part, identifies major components in fluid systems (such as pressure vessels, heat exchangers, storage tanks, pumps, piping, and valves) and in mechanical systems (such as cranes, refueling platforms, and other miscellaneous handling equipment). In addition, SSAR P&IDs identify the classification boundaries or interconnecting piping and valves. All of the above SSCs are constructed in conformance with applicable ASME Code and industry standards. Conformance to RG 1.26, previous staff positions, and applicable ASME Codes and industry standards provides assurance that component quality will be commensurate with the importance of the safety function of these systems. This constitutes the basis for satisfying GDC 1 and is acceptable.

3.3 Wind and Tornado Loadings

3.3.1 Wind Design Criteria

As described in SSAR Section 3.3.1 and Table 2.0-1, Amendment 32, the basic wind speed is 177 km/hr (110 mi/hr) at an elevation of 10 m (33 ft) above grade with a recurrence interval of 50 years. This basic wind speed is scaled by an importance factor (as defined in ANSI/American Society of Civil Engineers (ASCE) 7-88, "Minimum Design Loadings for Buildings and Other Structures") of 1.0 and 1.11, shown on SSAR Table 3.3-1, for non-safety-related and safety-related structures, respectively.

The importance factor provides basic wind speeds associated with mean recurrence intervals of 100 and 25 years (annual probability of being exceeded equal to 0.01 and 0.04, respectively). The basic wind speed values of the map in the ANSI/ASCE 7-88 Standard are for a 50-year mean recurrence interval (annual probability of 0.02). The standard recommends that the basic wind speeds associated with a 100-year mean recurrence interval be used for the design of buildings and other structures where a high degree of hazard to life and property exists and when these buildings or other structures are considered to be essential facilities. The guideline in SRP Section 3.3.1, "Wind Loadings," refers to the design wind speed as the 100-year return period fastest mile of wind. Therefore, the use of importance factor of 1.11 is acceptable for the design of safety-related structures to result in the basic wind speed of 197 km/hr (122.1 mi/hr) because it accounts for the recurrence interval from 50 to 100 years.

An exposure category that adequately reflects the characteristics of ground surface irregularities should be determined for the site at which the building or structure is to be constructed. Account shall be taken of large variations in ground surface roughness that arise from natural topography and vegetation as well as constructed features. In SSAR Appendix 3H, all seismic Category I structures are specified to be assessed by exposure D which represents flat, unobstructed areas exposed to wind flowing over large bodies of water. This is acceptable.

Therefore, all seismic Category I structures within the ABWR standard plant that will be exposed to wind forces are designed to withstand the effects of the design wind that has a velocity of 197 km/hr (122.1 mi/hr) and exposure D.

The staff concludes that the ABWR plant design is acceptable and meets the requirements of General Design Criterion 2. This conclusion is based on the following:

> GE meets the requirements of GDC 2 with respect to the capability of the structures to withstand design wind loading so that their design reflects the following:
>
> (1) appropriate consideration for the SSAR Table 2.0-1 basic design wind with an appropriate margin;
>
> (2) appropriate combinations of the effects of normal accident conditions with the effects of the natural phenomena; and

(3) the importance of the safety function to be performed.

GE has met the requirements of GDC 2 by using ANSI/ASCE 7-88 and ASCE Paper 3269 to transform the wind velocity into an effective pressure on structures and to select pressure coefficients corresponding to the geometry and physical configuration of the structures.

The ABWR plant structures are designed with sufficient margin to prevent structural damage during the design basis wind loadings as described above so that the requirements of Item 1 listed are met. In addition, the design of seismic Category I structures, as required by Item 2 previously listed, includes, in an acceptable manner, load combinations that occur as a result of the design basis wind load and the loads resulting from normal and accident conditions.

The procedures used to determine the loadings on structures induced by the design basis wind are acceptable because they have been used in the design of conventional structures and have been proven to provide a conservative basis that ensures that the structures will withstand such environmental forces. The use of these procedures provides reasonable assurance that, in the event of design-basis winds, the structural integrity of the plant structures that have to be designed to be protected from wind will not be impaired and, consequently, safety-related systems and components located within these structures are adequately protected and will perform their intended safety functions, if needed, thus satisfying the requirements of Item 3.

In SSAR Amendment 32, as a COL applicant action item, GE added Section 3.3.3.3 and stated that all non-safety-related plant SSCs not designed for wind loads shall be analyzed using the 1.11 importance factor or shall be checked so that their mode of failure will not affect the ability of safety-related SSCs performing their intended safety functions. This was in response to the staff's concern in the discussion of these related subjects after issuing the DFSER. As a result of its review, the staff concludes that the action meets the guidelines of SRP Section 3.3.1 and is acceptable.

3.3.2 Tornado Design Criteria

In the DSER (SECY-91-153), the staff identified an open item related to the maximum wind speed for the design basis tornado (Outstanding Issue 4).

SSAR Section 3.3.2 specifies that all seismic Category I structures exposed to tornado forces are designed to resist a maximum tornado wind speed of 483 km/hr (300 mi/hr) and translational wind velocity of 97 km/hr (60 mi/hr). This also implies a maximum tangential velocity of 386 km/hr (240 mi/hr). Also specified are a simultaneous atmospheric pressure drop to 13.8 kPa (2.00 lbf/in^2) at the rate of 8.3 kPa/sec (1.20 lbf/in^2/sec) and the radius of maximum tornado is 45.7 m (150 ft). In SECY-93-087, the staff recommended that the Commission approve its position to employ a maximum tornado wind speed of 483 km/hr (300 mph) in the design of evolutionary and passive ALWRs. In its SRM dated July 21, 1993, the Commission approved the staff's position. On the basis of this evaluation, the staff concludes that the ABWR design meets the Commission-approved staff recommendation for design basis tornado and is acceptable. This resolved Outstanding Issue 4 of the DSER (SECY-91-153).

The consideration of tornado loadings in the design of seismic Category I structures is in accordance with Bechtel Topical Report BC-TOP-3, "Tornado and Extreme Wind Design Criteria for Nuclear Power Plants," which NRC approved as a reference in plant applications. The procedures used to transform the tornado wind velocity into pressure loadings in BC-TOP-3 are similar to those used for the design wind loadings discussed in Section 3.3.1 of this report. The effects of tornado missiles are determined using the procedures discussed in Section 3.5 of this report.

The total effect of the design tornado on seismic Category I structures is determined by appropriate combinations of individual effects of the tornado wind pressure, tornado wind drop, and tornado-associated missiles. By using the maximum wind speed of 483 km/hr (300 mph) and other associated parameters, the ABWR design meets the guidelines of SRP Section 3.3.2. This is acceptable.

By using BC-TOP-3, GE designed the ABWR plant structures with sufficient margin to prevent structural damage during the most severe tornado loadings determined to be appropriate for most sites. In addition, the design of seismic Category I structures includes load combinations involving tornado load and the loads resulting from normal plant operation and accident conditions.

The procedures of ANSI/ASCE 7-88 and ASCE Paper 3269 to determine the tornado-induced loadings on structures have been used in the design of conventional structures for the most severe winds. They are based on the performance of such structures, and are conservative. Conservative load combinations, as discussed in SRP Section 3.8.4, together with the conservative wind loading assessment ensures that the structures designed in this manner will withstand such severe environmental forces.

The use of these procedures provides reasonable assurance that, in the event of a design-basis tornado, the structural integrity of the plant structures that have to be designed for tornados will not be impaired and, consequently, safety-related systems and components located within these structures will be adequately protected to enable the performance of their intended safety functions.

In the DFSER, the staff discussed the COL applicant's action to ensure that the collapse of non-seismic Category I SSCs that are not designed for tornado loads, such as cooling towers or stacks outside the nuclear island, will not endanger seismic Category I SSCs, and that site-dependent effects of blast loads will be less than those of design tornado pressures or provide justification for any deviations. This was DFSER COL Action Item 3.3.2-1. In response, in SSAR Amendment 33, GE revised Section 3.3.3.4 (formerly Section 3.3.3.3) and stated, as a COL applicant action item, that all remaining plant SSCs not designed for tornado loads shall be analyzed for the site-specific loadings to ensure that their mode of failure will not affect the ability of the seismic Category I ABWR standard plant SSCs to perform their intended safety functions. This is acceptable. On the site-dependent blast loads, as a COL applicant action item, GE stated in SSAR Section 2.3.2.5 that if the site-dependent blast loads are larger than those of design tornado pressures, all load combinations should be changed accordingly. This addresses the concern of site-dependent blast loads by meeting RG 1.91 and is also acceptable.

3.4 Water Level (Flood) Design

3.4.1 Flood Protection

The staff reviewed the ABWR flood design in accordance with SRP Sections 3.4.1 and 3.4.2 to ensure conformance with the requirements of GDC 2 and 10 CFR Part 100, Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants," Section IV(c), as related to protecting structures, systems, and components (SSCs) important to safety from the effects of floods and investigating seismically-induced floods and water waves. The review addressed the overall plant flood protection design, including safety-related SSCs whose failure as a result of flooding could prevent safe shutdown of the plant or result in uncontrolled release of radioactivity.

SSAR Section 3.4.1 discusses the flood protection measures that are applicable to the standard ABWR plant seismic Category I SSC for both external flooding and postulated flooding from plant component failures. The reactor and control buildings are designed to seismic Category I standards. In addition, portion of the radwaste building that is below plant grade is also designed to seismic Category I standards. The flood levels and conditions are described in SSAR Table 2.0-1. GE assumed maximum ground water and flood levels to be 61.0 cm (2 ft) and 30.5 cm (1 foot) below grade, respectively. As discussed in SSAR Section 3.4.2 and Table 2.0-1, GE did not consider dynamic force resulting from flooding because the design flood elevation is assumed to be below the plant finish grade. GE considered only the hydrostatic pressure caused by the design flood water level, ground water, and soil pressures in its analysis of the seismic Category I structures. However, GE indicated that the seismic Category I SSCs in the ABWR standard nuclear island were analyzed and designed for the maximum hydrostatic and hydrodynamic forces in accordance with the loads and load combinations documented in SSAR Subsections 3.8.4.3 and 3.8.5.3. In addition, GE also showed that the seismic Category I SSCs are in a stable condition with respect to either moments or uplift forces that result from the proper load combinations, including the design basis flood. GE's approach for the structural design against flooding complies with the guidelines of SRP Sections 3.7.2 and 3.8.4 and is acceptable. Furthermore, GE identified exterior or access openings and penetrations that will be below the design flood level in SSAR Table 6.2-9.

Safety-related systems and components that may be affected by external floods are protected either because of their location above the design flood level or because they are enclosed in reinforced concrete seismic Category I structures that have a wall thickness of not less than 0.6 m (2 ft) for portions of the structures below the flood level. The structures are provided with waterproof coating up to 8 cm (3 in.) above the plant ground grade level, water stops in all construction joints below flood level, watertight doors and penetrations installed below design flood level, and roofs designed to prevent pooling of large amounts of water in accordance with RG 1.102, "Flood Protection for Nuclear Power Plants." In addition, below-grade tunnels are designed with interconnecting seals such that the tunnel structural walls do not penetrate exterior building walls. In the DSER (SECY-91-235), the staff identified an open item regarding the lack of information on how safety-related buildings will withstand the effects of standing water on the roofs (Outstanding Issue 5). In Appendix 3H to the SSAR, GE stated that the design rainfall is 493 mm/sq. km/hr (19.4 in./sq mile/hr), and the roof of the seismic Category I structures is to be designed to have such parapet heights that would prevent excessive ponding of water. This design rainfall together with the drainage system and suitable parapet design form a reasonable basis to conclude that there will be no roof overloading. This is acceptable, and DSER Outstanding Issue 5 is resolved. Furthermore, on the basis of the evaluation of the plant layout drawings in SSAR Section 1.2, the staff concludes that all the safety-related systems and components located inside seismic Category I structures are protected from ground water seepage and external floods and meet the guidance of SRP Section 3.4.1.

GE analyzed compartment flooding from postulated component or system failures separately for the reactor, control, radwaste, service, and turbine buildings. GE considered single failure of an active component for compartment flooding. GE analyzed the failure of moderate-energy piping larger than 2.54 cm (1-in.) diameter in accordance with ANSI/ANS 56.11 "Standard Design Criteria for Protection Against the Effects of Compartment Flooding in Light Water Reactor Plants," and Crane Co. Technical Paper 410, 1973, "Flow of Fluids Through Valves, Fittings, and Pipe." GE did not consider the effect of drain sump pump operation. In the DSER (SECY-91-235), the staff noted GE's position that high-energy line breaks (HELBs) inside the main steam tunnel (MST) were excluded from evaluation because this area will be instrumented for detection of leaks before a line break occurred. The staff stated that an LBB analysis should use plant-specific data such as piping geometry, materials, fabrication procedures, and pipe support locations (LBB is discussed in Section 3.6.3 of this report). In a meeting with the staff on May 5, 1992, GE committed to removing references to LBB from the SSAR. This was DFSER Confirmatory Item 3.4.1-1. In SSAR Amendment 34, GE removed all references to LBB from SSAR Section 3.4.1. The staff reviewed the SSAR and concluded that Appendix 3F of the SSAR that discussed LBB had been deleted as agreed. DFSER Confirmatory Item 3.4.1-1 is, therefore, resolved.

GE provided the results of a flooding analysis for the reactor building (RB). The analysis identified flooding hazards on a floor-by-floor basis and identified features used to ensure that multiple divisions of safety-related equipment will not be adversely affected as a result of an internal flood. The primary means of protection include divisional separation, elevation differences, and placing safety-related equipment on raised pads. Flooding in multiple divisions located below plant grade is prevented by the use of watertight doors and penetrations between divisions. Flooding above plant grade is directed to the building basement via stairways, elevator shafts, and drains. Additional features used to limit flood effects to one division include:

(1) divisional flood walls at the -8200 mm (-26 ft-10 7/8 in.) elevation of the control and reactor buildings are 0.6 m (24 in.) or thicker,

(2) doors and penetrations rated as 3-hour fire barriers are assumed to prevent water spray from crossing divisional boundaries,

(3) floors are assumed to prevent water seepage to lower levels,

(4) penetrations between floors for pipe, cable, HVAC duct, and other equipment will be designed to prevent water seepage to lower elevations from 200 mm of standing water through the use of seals or curbs,

(5) equipment access hatches shall prevent water seepage to lower elevations from 200 mm (8 in.) of standing water. Hatches to filter/demineralizer compartments may not be required to prevent water seepage,

(6) water from a pipe break is assumed to flow under non-watertight doors and spread evenly over the available areas. Water sensitive safety-related equipment is raised 200 mm (8 in.) above the floor. The depth of water is limited to less than 200 mm by using available floor space and limiting the volume of water sources. As mentioned earlier, floor drains, stairways, and elevator shafts provide drain paths to the building basements.

A compliance review will be conducted of the as-built design against the assumptions and requirements that are the basis of the flood analysis in SSAR Section 3.4.1 and SSAR Appendix 19R. The results of this review will be documented in a Flood Analysis Report and will include an assessment and disposition of any non-compliances found between the as-built facility and the design information. The criterion for determining the appropriate disposition of any non-compliances will be that the as-built facility conforms to the design criteria and flood protection characteristics in the original analysis.

GE also provided the results of a flood analysis for the control building. Both the staff and the Advisory Committee on Reactor Safeguards (ACRS) raised concerns regarding flood protection for the main control room as a result of a pipe break in the MST and as a result of a pipe break in the portion of the reactor service water (RSW) system in the control building.

GE stated that the greatest flood hazard in the MST occurs as a result of a feedwater line break. The amount of water associated with the break will be limited by manually closing the feedwater isolation valves. The water in the MST will collect in the large cavity at the RB end of the tunnel; any overflow from the cavity will flow to the turbine building. There are no openings or penetrations between the MST and the control building to provide a path for water to enter the control building from the MST. This ensures that the control building is protected from flooding as a result of pipe failures in the MST. In the DFSER, the staff noted this to be acceptable subject to inclusion of this information in the SSAR. This was DFSER Confirmatory Item 3.4.1-2. GE has included this information in the SSAR. On the basis of this evaluation, DFSER Confirmatory Item 3.4.1-2 is resolved.

The control room area sits on a raised floor. The outside wall of the control room complex is sealed to prevent water in the corridor from entering the control room. Drinking water and bathroom facilities are the only sources of water inside the control room. Fire hoses and standpipes are located in the corridors.

The evaluation of control building flooding included events that may result from the failure of fire protection systems. Flow of the water supplied from the failure of these systems to the control areas will be prevented by diverting the water to the building basement and by locating safety-related equipment in the building basement at least 400 mm above the basement floor. Safety-related equipment on higher levels are raised at least 200 mm (8 in.) above the floor.

Fire-fighting activities inside the control room will introduce water into this area. Because the control room is continuously manned, it is expected that fires inside the control room will be identified and quickly extinguished, thus limiting the amount of water which will accumulate. Water will either collect in the control room subfloor and drain to the building basement or flow out into the hallways, down stairways and elevator shafts, and to the building basement. The staff expects that minimal damage to safety-related operations will occur due to the limited volume of water generated and the collection and diversion of the water.

GE also addressed control building flooding as a result of an RSW failure. A break in the RSW line inside the control building will allow water from the open cycle water source to flood the control building. The three redundant divisions of RSW supply cooling water to the reactor building cooling water (RCW) system heat exchangers located in the control building basement. Any flooding from sources above the basement that could adversely affect safe-shutdown equipment will be directed through floor drains, stairways, and elevator shafts to the basement. Each division is physically separated by watertight doors, is equipped with a sump pump, and contains two sets of safety-grade level sensors in a two-out-of-four logic. The first (lower) set will alarm to alert the control room operator of the presence of excessive water in the room. The second set of sensors will inform the control room operator that a serious flood situation exists in the RCW/RSW room. In addition, these high-level sensors will trip the RSW pump and close the isolation valves for the affected division. As a result, a maximum of ~5 m (~16.4 ft) of water will collect in a divisional room. This amount of water will be contained in the affected divisional room and will not adversely affect the safety function of the redundant divisions of the RCW and RSW. This analysis demonstrates that the safety-related equipment is adequately protected from the effects of a pipe break in the RSW system. In the DFSER, the staff noted that this evaluation was acceptable subject to its inclusion in the SSAR. This was DFSER Confirmatory Item 3.4.1-3. The SSAR now includes this information, and Confirmatory Item 3.4.1-3 is resolved.

The radwaste building contains no safety-related equipment and is isolated from the other plant structures with the exception of a tunnel which connects the reactor, control, turbine, and radwaste buildings. Liquid radwaste from these buildings is transferred to the waste processing system via lines running through this tunnel. The tunnel is sloped and connects to the turbine building at elevation 8,800 mm (28 ft-7/8 in.), the radwaste building at -1,500 mm (4 ft-11 in.), and the reactor and control buildings at -8,200 mm (-26 ft-10 7/8 in.). All ends of the tunnel are sealed to protect safety-related equipment from flooding which may originate in another building. All penetrations are designed to withstand the maximum expected hydrostatic and hydrodynamic loads. SSAR Subsection 3.12.3 provides design requirements for this tunnel and any other non-safety-grade tunnels that may be used in the ABWR design. Tunnel design includes provisions for water tightness, accessibility, leak detection, and water removal. Failure of the tunnel will not adversely effect the ability to bring the plant to a safe-shutdown condition and will not damage the seals at the interface with buildings housing safety-related equipment.

SSAR Subsection 3.12.2 provides design requirements for safety-related tunnels. These tunnels will:

(1) meet the applicable requirements regarding seismic, flood, fire, and environmental conditions,

(2) be routed independently or through separate compartments to assure divisional separation,

(3) withstand both hydrostatic and hydrodynamic effects from pipe breaks. Provisions for pressure relief shall be included as necessary,

(4) ensure the integrity of the piping penetrations at interfacing buildings under design conditions,

(5) allow periodic inspection of piping, cables, and penetrations,

(6) include provisions for leak detection and water removal,

(7) prevent unauthorized access to the tunnel,

(8) include provisions for preventing fuel oil from accumulating near structures housing safety-related equipment.

The service building is a non-seismic concrete structure, does not house any safety-related equipment, and provides access tunnels to the reactor, control, and turbine buildings. The access corridors below plant grade which lead into buildings that house safety-related equipment have watertight doors to prevent seepage into the corridors. The service building has floor drains and two sumps to collect and transfer flood water.

In the DFSER, the staff stated that GE had not included an analysis of flooding in the turbine building and its potential effects on safety-related equipment. The staff requested that GE provide a flood analysis to characterize the nature of the hazards and the design features to protect safety-related equipment from flooding in the turbine building. This was DFSER Open Item 3.4.1-1. Subsequently, GE provided a flood analysis for the turbine building. The major flood hazards in the turbine building occur as a result of a failure in the circulating water system (CWS) or the turbine service water system (TSW). Both of these systems are open-cycle systems. The CWS, because of its larger capacity, is the bounding hazard. Leak detectors in the condenser pit will alert the control room and automatically isolate the CWS on indication of building flooding. TSW has no automatic isolation function because any flooding resulting from a break in this line will be slow enough for operators to manually shut down the system on a flood alarm. A failure in either of these lines can result in flooding of the building up to grade. A non-watertight truck door at grade level will allow release of the flood water onto the ground. As was stated earlier, the tunnel connecting the radwaste, reactor, and turbine buildings is sealed to prevent water from entering the RB. The staff finds the design to be an acceptable means to protect safety-related SSCs from the effects of flooding as a result of pipe failures in the turbine building. GE has also included this information in the SSAR. Therefore, Open Item 3.4.1-1 is resolved.

As discussed in SSAR Section 10.1, safety-related instrumentation is provided in the turbine building to detect the fast closure of the turbine main steam stop and control valve oil pressure, stop valve position, turbine first-stage pressure, and main condenser pressure. This instrumentation is designed to fail safe should it be damaged due to fire, flood, missiles, or pipe failures. The staff finds this acceptable.

SSAR Table 3.4-1 provides details about access openings between buildings. All penetrations and access ways below flood level are watertight and designed to withstand the maximum hydrostatic loads.

In the DSER (SECY-91-235), the staff identified a need for an interface requirement to provide a flood analysis for structures outside the ABWR design scope that house safety-related equipment (Outstanding Issue 6). Upon further evaluation, the staff determined that it is sufficient to identify a COL action in the SSAR for the COL applicant to provide this information because the information is plant-specific and is not needed for design certification. This issue was reclassified as COL Action Item 3.4.1-1 in the DFSER. SSAR Section 3.4.3.3 provides this information. This is acceptable and resolved DSER Outstanding Issue 6.

Based on the above review of the information provided in the SSAR, the staff concludes that safety related equipment and instrumentation housed in the reactor, control, and turbine buildings is adequately protected from the effects of both internal and external flooding and meets the requirements of GDC 2.

The COL applicant is responsible for identifying external flood and precipitation hazards beyond those assumed in the ABWR flood analysis. With this, the applicant referencing the ABWR design will be able to identify site-specific flood hazards and provide measures to protect safety-related equipment from these hazards. This will meet the requirements of 10 CFR Part 100, Appendix A, Section IV(c), regarding the required investigation for seismically-induced floods and water waves.

GE originally submitted the design descriptions and the inspections, tests, analyses, and acceptance criteria (ITAAC) for the in-scope structures. These include verifications related to flood protection. During the preparation of the DFSER, the staff review was in progress. This was DFSER Open Item 3.4.1-2. Subsequently, GE provided revised design descriptions and ITAAC. The adequacy and acceptability of the design description and the ITAAC are evaluated in Section 14.3 of this report. Therefore, DFSER Open Item 3.4.1-2 is resolved.

The staff's flood protection review included all systems and components whose failure could prevent the safe shutdown of the plant and maintenance thereof or result in significant uncontrolled release of radioactivity. From this review of the applicant's design criteria, design bases, and safety classification for safety-related SSCs necessary for a safe plant shutdown during and following the flood condition from either external or internal causes, the staff concludes that the design of the facility for flood protection conforms to the Commission's regulations as set forth in GDC 2 and 10 CFR Part 100 Appendix A with respect to protection of SSCs important to safety from the effects of floods, tsunamis, and seiches.

The staff concludes that the design provides adequate protection against floods and the effects of floods, meets the guidelines of SRP Section 3.4.1 and the requirements of GDC 2 as it relates to flood protection, and is, therefore, acceptable.

3.4.2 Water Level (Flood) Design Procedure

GE identified the design-basis flood elevation to be up to 30.5 cm (1 ft) below grade and the design-basis ground water level to be up to 61.0 cm (2 ft) below grade for the ABWR standard plant structures. GE performed calculations to determine the hydrostatic pressure and the hydrodynamic forces (as a result of earthquakes) on the structures that would result from these levels in combination with other loads as specified in the SSAR for structures. The staff finds that GE's calculations were performed using load combinations that are consistent with those specified in SRP Section 3.8.4 and meet the requirements of GDC 2. The staff evaluation for the design of the seismic Category I structures against the hydrostatic and hydrodynamic forces resulting from design-basis ground water during an earthquake is also provided in Section 3.4.1 of this report.

The staff concludes that the plant design on the basis of the specified water level is acceptable.

By limiting the design-basis flood elevation to 30.5 cm (1 ft) below grade and the design-basis ground water level to 61.0 cm (2 ft) below grade in the design of plant structures, and by employing the load combinations specified in SSAR Sections 3.8.4.3 and 3.8.5.3, the ABWR design provides sufficient margin to prevent structural damage that is due to the most severe flood or ground water and the associated dynamic effects that were determined appropriate for the flood levels. In addition, the design of seismic Category I structures includes, in an acceptable manner, load combinations that occur as a result of the most severe flood or ground-water-related loads and the loads resulting from normal and accident conditions.

Because the bounding flood elevation for the ABWR is below plant grade, GE considered only the hydrostatic force in determining the loadings on seismic Category I structures. This method of calculating loads is consistent with that of SRP Section 3.4.2 and provides a conservative basis that, together with other engineering design considerations, such as combination of flood loads with other loads, and consideration of floatation as indicated in Section 3.8.5 of this report, ensures that the structures will withstand such environmental forces.

To meet the requirements of GDC 2 and Appendix A to 10 CFR Part 100, and the guidelines of SRP Section 3.4.1, the COL applicant should provide a specific description of its site and the elevations for all safety-related structures, exterior accesses, equipment, and systems, from the standpoint of hydrological considerations and flood history (including date, level, peak discharge, and related information for major historical floods in the region of the site). To determine water level for the site, the COL applicant should consider the following factors, if applicable:

- probable maximum precipitation
- runoff and stream-course models
- maximum flood flow
- coincident wind-wave activity

In the DFSER, the staff noted that the COL applicant should also demonstrate in its plant-site-unique application that all the seismic Category I structures either will be protected against flood damage or will not be subject to damaging flooding. Hydrostatic and hydrodynamic effects of the flood should be considered and described for all postulated design flood levels for the conditions set for the future site as previously outlined. This was DFSER COL Action Item 3.4.2-1. SSAR Section 2.3.2 states that the COL applicant shall provide identification and description of all differences from SRP Section II, "Acceptance Criteria," for site characteristics (as augmented by SSAR Table 2.1-1). Also, SSAR Sections 2.3.2.11 through 2.3.2.13 identify the COL actions related to plant-specific hydrologic description, floods, and probable maximum flood on streams and rivers. These address the staff's concern and are acceptable.

The staff finds that the ABWR flood design provides reasonable assurance that, in the event of floods or high ground water level, the structural integrity of the plant seismic Category I structures will not be impaired and, consequently, seismic Category I systems and components located within these structures will be adequately protected and will be expected to perform necessary safety functions, as required.

3.5 Missiles

3.5.1 Missile Selection and Description

Seismic Category I structures have been analyzed and designed to be protected from a wide spectrum of missiles (e.g., missiles from rotating and pressurized equipment, gravitational missiles, and missiles generated from tornado winds).

Once a potential missile is identified, its statistical significance is determined (a significant missile is one which could cause unacceptable consequences or violate the guidelines of 10 CFR Part 100).

If the probability of occurrence of a missile (P_1) is less than 10^{-7} per year, the missile is not considered significant. If P_1 is greater than 10^{-7} per year, the probability that it will impact a significant target (P_2) is determined. If the product of these probabilities ($P_1 \times P_2$) is less than 10^{-7} per year, the missile is not considered significant. If the above product is greater than 10^{-7} per year, the probability of significant damage (P_3) is determined. If the combined probability ($P_1 \times P_2 \times P_3$) is less than 10^{-7} per year, the missile is not considered significant. Finally, if the above combined probability is greater than 10^{-7} per year, missile protection of safety-related SSCs is provided by one or more of the following:

(1) locating the system or component in a missile-proof structure,

(2) separating redundant systems or components for the missile path or range,

(3) providing local shields and barriers for systems and components,

(4) designing the equipment to withstand the impact of the most damaging missile,

(5) providing design features to prevent the generation of missiles,

(6) orienting missile sources to prevent missiles from striking equipment important to safety.

The SSAR identified the design and operational criteria for missile protection as well as the systems requiring missile protection.

3.5.1.1 Internally-generated missiles (Outside Containment)

The staff reviewed the ABWR design for protecting SSCs important to safety against internally-generated missiles outside the containment in accordance with SRP Section 3.5.1.1. Specifically, the review included the missile-protection design features for the SSCs whose failure could prevent safe shutdown of the facility or result in significant uncontrolled release of radioactivity. The SRP acceptance criteria specify that the design meet GDC 4, "Environmental and Dynamic Effects Design Bases" as it relates to protecting the SSCs outside the containment against the effects of missiles that can be internally generated during facility operation, and RG 1.115, "Protection Against Low-Trajectory Turbine Missiles," Revision 1. The staff's review of turbine-generator analyses is provided in Section 3.5.1.3 of this report. The review included all areas outside the containment that are within the scope of the ABWR design.

GE evaluated potential internally-generated missiles resulting from failure of the plant equipment within the nuclear island and located outside the containment. Potential missiles are categorized into two groups:

(1) potential missiles that could result from the failure of rotating equipment such as the main turbine, reactor core isolation cooling (RCIC) turbine, pumps, fans, blowers, diesel generators, and compressors

(2) pressurized high-energy fluid system components considered potential missile sources, including valve bonnets, stems, pressure vessels, thermowells, retaining bolts, and blowout panels

GE also performed probability calculations for certain rotating equipment and pressurized components to identify qualifying missiles. Piping failures were not included as sources of potential internally-generated missiles because the whipping section remains attached to the remainder of the pipe. The dynamic effects associated with this type of break are addressed in Section 3.6 of this report.

SSCs important to safety are protected from internally-generated missiles by

- placing them in individual missile-proof enclosures
- providing localized protective shields and barriers
- physically separating redundant components to prevent damage from a missile

- orienting missile sources to prevent unacceptable consequences of missile generation

The adequacy of structures, shields, and barriers provided for missile protection is evaluated in Section 3.5.3 of this report.

All electrically-powered rotating equipment such as pumps and fans are ac powered and their speeds are governed by an ac power supply. Since the ac power supply frequency variation is limited to a narrow range, this rotating equipment is unlikely to attain an overspeed condition. Fan blade casings are designed with sufficient thickness so that a fan blade breaking off at rated speed will not penetrate the fan casing. GE submitted PED-18-0389, a proprietary missile generation study, which provides the details of the methodology used to ensure that safety-related SSCs will be protected from missiles that may be generated from rotating equipment. On the basis of the information provided in the study, the staff finds this methodology to be an acceptable means of protecting safety-related SSCs from missile damage.

Valve bonnets have sufficient design safety factors (based on the ultimate strength of the materials) to prevent them from becoming credible missiles. Valve stems have design features such as stem threads and backseats to prevent their ejection. Nuts, bolts, nut and bolt combinations, and nut and stud combinations have insufficient stored energy to require missile-protection analysis. GE analyzed the thermowells and concluded that their maximum ejection velocity was insufficient to damage safety-related systems. Blowout panels are restrained by hinges to prevent the panels from becoming credible missiles. Air bottles are located, oriented, and restrained so as to prevent them from becoming missiles.

GE states that adequate physical separation is provided between redundant trains for all safety-related systems. Further, each safety-related system is contained in its own room of a seismic Category I building. The walls, floor, and ceiling of this room act as the missile barrier or shield for the safety-related components in the room, protecting these components from missiles generated outside the room. Also, stored spent fuel is protected from internally-generated missiles because no high-energy piping systems or rotating machinery are in its vicinity.

SSAR Section 3.5.1.1.1.3 and SSAR Figure 3.5-2 provide information that shows the orientation of the turbine-generator in relation to the in-scope buildings which house safety-related equipment and identifies the low-trajectory missile strike zone. On the basis of this information, the staff concludes that no safety-related equipment is located within the turbine-generator missile strike zone. As a result, this equipment is protected from a missile ejected from the turbine-generator. Therefore, the staff concludes that the ABWR design meets the guidelines of RG 1.115 and the requirements of GDC 4.

In the DSER (SECY-91-153), the staff stated that GE did not describe the means by which safety-related systems will be protected from missiles generated by non-safety-related components (Outstanding Issue 6). To address the staff's concern, GE stated (in response to request for additional information (RAI) Q410.9) that no local barriers or shields outside the containment will be used for mitigating missile effects. For non-safety-related components, no local shields and barriers will be required. Non-safety-related components are arranged in such a way that any missile-generating component will not cause the failure of more than one division of safety-related equipment. On the basis of GE's responses, which described the physical arrangement of equipment, the staff considered, in the DFSER, the chance of more than one division of a safety-related system being struck by missiles generated by non-safety-related components to be extremely low, and considered GE's resolution to this concern acceptable. This resolved DSER Outstanding Issue 6.

As discussed in SSAR Section 10.1, safety-related instrumentation is provided in the turbine building to detect the fast closure of the turbine main steam stop and control valve oil pressure, stop valve position, turbine first-stage pressure, and main condenser pressure. This instrumentation is designed to fail safe should it be damaged due to fire, flood, missiles, or pipe failures. The staff finds this acceptable.

In the DFSER, the staff noted that the COL applicant should provide design details of all pressurized gas bottles as well as details of missile protection features for SSCs that are outside of the ABWR scope. This was DFSER COL Action Item 3.5.1.1-1. SSAR Section 3.5.1.1.2.2, subsection (3), provides design guidance to ensure that the pressurized gas bottles will not become the source of missiles. This is acceptable.

In the DSER (SECY-91-153), the staff incorrectly identified Outstanding Issue 5 related to the use of induction motors. This was deleted in the DFSER.

Verification of adequate separation of redundant divisions of safety-related systems ensures that these systems will not be rendered inoperable as a result of an internally-generated missile outside of containment. In the DFSER, this verification was identified as Open Item 3.5.1.1-1. Subsequently, GE submitted design descriptions and ITAAC for the buildings within the ABWR design scope. These include verifications related to missile protection for safety-related SSCs. The adequacy and acceptability of the design description and ITAAC are evaluated in Section 14.3 of this report. On the basis of the above, DFSER Open Item 3.5.1.1-1 is resolved.

The review of possible effects of internally generated missiles (outside containment) included SSCs whose failure could prevent safe shutdown of the plant or result in significant uncontrolled release of radioactivity. From its review of the SSAR design bases and criteria for safety-related SSCs necessary to maintain a safe plant shutdown, the staff concludes that, on the basis of the foregoing evaluation of separation of redundant divisions, the SSCs will be protected from internally generated missiles (outside containment). Therefore, the ABWR design for protecting SSCs from internally generated missiles outside the containment meets the guidelines of SRP Section 3.5.1.1 and the dynamic effect design bases requirements of GDC 4 and is acceptable.

3.5.1.2 Internally-generated missiles (Inside Containment)

The staff reviewed the design of the facility for protecting SSCs important to safety against internally-generated missiles inside the containment in accordance with SRP Section 3.5.1.2. Specifically, the review included missile-protection design features for the SSCs whose failure could prevent safe shutdown of the facility or result in significant uncontrolled release of radioactivity. The SRP acceptance criteria specify that the design must meet GDC 4, "Environmental and Dynamic Effects Design Bases," as it relates to protecting the SSCs against the effects of missiles that can be generated inside the containment during facility operation.

GE evaluated potential internally-generated missiles resulting from plant equipment and component failures within the containment structure. The potential missiles identified by this analysis were categorized into three groups: missiles generated by rotating equipment (e.g., pump impellers, compressors, and fan blades), missiles generated by pressurized components (e.g., valve bonnets, thermowells, nuts, bolts, studs, valve stems, and accumulators), and gravitational missiles.

GE's analysis of failures of rotating equipment indicates that the equipment design prevents such equipment from becoming a source of potential missiles. Pumps are unlikely to achieve an overspeed condition. All electrically-powered rotating equipment such as pumps and fans are ac powered and their speeds are governed by an ac power supply. Since the ac power supply frequency variation is limited to a narrow range, this rotating equipment is unlikely to attain an overspeed condition. Fan blade casings are designed with sufficient thickness so that a fan blade breaking off at rated speed will not penetrate the fan casing. GE submitted PDE-18-0389, a proprietary missile generation study, which provides the details of the methodology used to ensure that safety-related SSCs will be protected from missiles that may be generated from rotating equipment. On the basis of the information provided in the study, the staff finds this methodology to be an acceptable means of protecting safety-related SSCs from missile damage.

GE stated in SSAR Section 3.5.1 that missiles generated by rotating equipment will be contained by the equipment housing. In the DFSER, the staff noted that GE should provide design information supporting this assertion or clearly state that this information will be provided by the COL applicant. This was DFSER Open Item 3.5.1.2-1. Subsequently, GE submitted the proprietary missile generation study previously discussed, which clarifies GE's approach to containment of missiles generated by rotating equipment. In addition, GE included this study as a reference in the SSAR Section 3.5.5. On the basis of the information provided in the study, the staff found this methodology an acceptable means of protecting safety-related SSCs from missile damage; therefore, DFSER Open Item 3.5.1.2-1 is resolved.

Pressurized components and equipment such as valve bonnets, valve stems, nuts, bolts, nut and bolt combinations, nut and stud combinations, thermowells, and blowout panels inside the containment are not considered credible missiles for the same reasons as stated in Section 3.5.1.1 of this report (e.g., design features or insufficient stored energy). Automatic depressurization system (ADS) accumulators are moderate energy vessels and are not considered as credible missile sources.

Fine motion control rod drive mechanisms under the reactor vessel are not credible missiles because the housings are designed to prevent any significant nuclear transient in the event of a drive housing break. Specifically, the pressure boundary containing the fine motion control rod drive mechanisms, including the bolted flange connections, will be stressed below the ASME Code limits and will meet all code requirements. To prevent control rod drop accidents, internal restraints will be provided to support the fine motion control rod drive housing in the event the housing-to-nozzle weld or the housing fails.

GE evaluated the potential for gravitational missiles inside the containment. Non-safety-related components are seismically supported to prevent their collapse during an SSE. These components include all cable trays for both Class 1E and non-Class 1E circuits as well as non-

Class 1E conduits and non-safety-related piping that could be potential hazards to safety-related equipment. Also, equipment undergoing maintenance either will be removed during operation or will be seismically restrained to prevent it from becoming a missile. This was DFSER COL Action Item 3.5.1.2-1. SSAR Section 3.5.4.6 states that the COL applicant will provide these procedures. This is acceptable.

In the DFSER, the staff noted that the COL applicant should provide procedures ensuring that equipment undergoing maintenance will be removed from containment during operation or will be seismically restrained to protect it from becoming a missile. This was DFSER COL Action Item 3.5.1.2-1. In responding to this comment, SSAR Section 3.5.4.6, states that the COL applicant will provide procedures to ensure that all equipment inside containment that is required during maintenance will either be removed prior to operation, moved to a location where it is not a potential hazard to safety-related equipment, or seismically restrained to prevent it from becoming a missile. This is acceptable.

As discussed in SSAR Section 3.6.1, redundant divisions of safety-related equipment inside containment are physically separated to the extent practicable to maintain independence in order to prevent loss of safety function due to a single pipe break event. Specifically, redundant divisions of safety-related equipment are widely spaced around containment. A high-energy line separation analysis (HELSA) is used to determine which high-energy lines meet the spatial separation requirements. Although the HELSA evaluation is performed to determine if safety-related systems and components are adequately protected from the effects of pipe breaks, it is also helpful in determining if adequate protection is provided for these systems and components from missiles which may be generated inside containment.

As part of this evaluation, safety-related systems and components which are more than 9.14 m (30 ft) from any high-energy lines are considered as meeting the spatial separation requirements. Safety systems and components which are less than 9.14 m (30 ft) from high-energy piping are evaluated to see if damage could occur to more than one division. If damage can occur to only one division of safety-related systems, the requirement for separation of redundant equipment is met. If more than one division can be damaged by high-energy piping, barriers, shields, and enclosures will be utilized to protect safety systems and equipment. Based on this information regarding protection of safety systems and components from pipe failures, the staff concludes that this approach is equally effective in protecting these systems and components from missiles generated inside containment.

In the DSER (SECY-91-153), the staff stated that credible secondary missiles (concrete fragments) resulting from the impact of primary missiles on containment structures should be addressed in an interface requirement (Outstanding Issue 7). In the DFSER, the staff noted that this issue should be addressed as part of the ITAAC program. However, as a result of further evaluation, the staff determined that the protection of safety-related SSCs from secondary missiles is addressed as part of the evaluation in Section 3.5.3 of this report. Thus, DSER Outstanding Issue 7 is withdrawn and resolved.

As mentioned in SSAR Section 10.1, safety-related instrumentation is provided in the turbine building to detect the fast closure of the turbine main steam stop and control valve oil pressure, stop valve position, turbine first-stage pressure, and main condenser pressure. This instrumentation is designed to fail safe should it be damaged due to fire, flood, missiles, or pipe failures. The staff finds this acceptable.

As a result of review of this information, the staff concludes that safety-related equipment is adequately protected from missiles that may be generated inside containment since each redundant division of safety-related systems is housed in its own separate missile-proof enclosure. Therefore, the requirements of GDC 4 are met.

Verification of adequate separation of redundant divisions of safety-related systems ensures that these systems will not be rendered inoperable as a result of an internally-generated missile inside of containment. In the DFSER, this verification was identified as Open Item 3.5.1.2-2. Subsequently, GE submitted design descriptions and ITAAC for the buildings within the ABWR design scope. These include verifications related to missile protection for safety-related SSCs. The adequacy and acceptability of the design description and ITAAC are evaluated in Section 14.3 of this report, therefore, DFSER Open Item 3.5.1.2-2 is resolved.

The review of possible effects of internally generated missiles (inside containment) included SSCs whose failure could prevent safe shutdown of the plant or result in significant uncontrolled release of radioactivity. Based on the review of the SSAR design bases and criteria for safety-related SSCs necessary to maintain a safe plant shutdown, the staff concludes that the SSCs to be protected from internally-generated missiles (inside containment) meet the requirements of GDC 4.

The staff concludes that the ABWR design for protecting SSCs from internally generated missiles inside the containment meets the guidelines of SRP Section 3.5.1.2 and the dynamic effect design bases requirements of GDC 4 and is acceptable.

3.5.1.3 Turbine Missiles

The staff requested that GE provide additional information (Q251.13 and Q251.14) on turbine missile generation. GE responded by revising SSAR Section 3.5.1.1.1.3 to include a layout of the turbine-generator building indicating the ± 25 degree low-trajectory turbine missile ejection zone. The turbine-generator building is favorably oriented with respect to the primary containment building so that any postulated turbine missile will not strike the primary containment building. GE also commits to meeting RG 1.115, "Protection Against Low-Trajectory Turbine Missile," Revision 1, which specifies that the probability of unacceptable damage from turbine missiles be maintained to less than 10^{-7} per reactor-year.

The probability of unacceptable damage from turbine missiles is expressed as the product of (1) the probability of turbine missile generation resulting in the ejection of turbine disk (or internal structure) fragments through the turbine casing (P_1); (2) the probability of ejected missiles perforating intervening barriers and striking safety-related SSCs (P_2); and (3) the probability of struck structures, systems, or components failing to perform their safety functions (P_3).

In previous reviews of the probability calculation, the staff found that the mathematical models that were used to calculate P_2 and P_3 require numerous approximations and simplifying assumptions in order to incorporate available data in the analysis. As a result, the calculations of P_2 and P_3 were not accurate; however, the calculation of P_1 was considered more accurate and precise. Therefore, in recent years, the staff placed emphasis on reviewing the probability of turbine missile generation (P_1). The staff believes that if P_1 is controlled to a minimum, the final failure probability of unacceptable damage can be met. Consistent with the staff position taken in recently licensed nuclear plants, the probability of turbine missile generation should be kept to no greater than 10^{-5} per reactor-year for an unfavorably oriented turbine and 10^{-4} for a favorably oriented turbine. The staff also recommended certain actions for situations when the probability does not meet the minimum values required (see Table 3.1 of this report). The staff recommended in its DSER (SECY-91-153) that GE include the minimum requirement for the probability of turbine missile generation in SSAR Table 3.5-1. (This was not identified as an open item.) GE included this table in the SSAR and it is acceptable.

GE prepared a proprietary report in January 1984 entitled, "Probability of Missile Generation in General Electric Nuclear Turbines." The staff reviewed the report and found the methodology acceptable for calculating the

Table 3.1 Criteria pertaining to the probability of turbine missile generation for a favorably oriented turbine

Probability Criterion ($yr.^{-1}$)	Required Licensee Action
$P_1 < 10^{-4}$	This is the general, minimum reliability requirement for loading the turbine and bringing the system on line.
$10^{-4} < P_1 < 10^{-3}$	If this condition is reached during operation, the turbine may be kept in service until the next scheduled outage, at which time the licensee is to take action to reduce P_1 to meet the first criterion before returning the turbine to service.
$10^{-3} < P_1 < 10^{-2}$	If this condition is reached during operation, the turbine is to be isolated from the steam supply within 60 days, at which time the licensee is to take action to reduce P_1 to meet the first criterion before returning the turbine to service.
$10^{-2} < P_1$	If this condition is reached at any time during operation, the turbine is to be isolated from the steam supply within 6 days, at which time the licensee is to take action to reduce P_1 to meet the first criterion before returning to service.

probability of turbine missile generation (NUREG-1048, Supplement 6). Various parameters are considered in the methodology that influence the outcome of the turbine missile generation probability, such as turbine disk design, disk material properties, turbine speed control systems, postulated crack growth rate, and inspection intervals. On the basis of parametric variations, the methodology can optimize turbine disk inspection intervals so that the turbine missile generation probability will be kept to the allowable limit. The staff believes that by emphasizing on the turbine disk inspection program, which includes a specified inspection interval, and an effective turbine control system maintenance program, the turbine will be operated in a safe manner.

Consistent with the staff position taken in recently licensed nuclear plants, the COL applicant should submit for NRC approval, within 3 years of obtaining a COL, a turbine system maintenance program, including probability calculations of turbine missile generation based on the methodology approved by the NRC, or commit to volumetrically inspect all low-pressure turbine rotors at the second refueling outage and every other (alternate) refueling outage thereafter until a maintenance program is approved by the staff. This position is also discussed in Section 10.2.2 of this report. In the DSER (SECY-91-153), the staff classified this item as an interface requirement. Upon further review the staff reclassified this item in the DFSER as COL Action Item 3.5.1.3-1.

GE addressed this item in SSAR Section 3.5.1.1.1.3, stating that the COL applicant shall submit for NRC approval, within 3 years of obtaining a COL, a turbine system maintenance program, including probability calculations of turbine missile generation based on the methodology approved by the NRC, or commit to volumetrically inspect all low-pressure turbine rotors at the second refueling outage and every other (alternate) refueling outage thereafter until a maintenance program is approved by the staff. This is acceptable.

3.5.1.4 Missiles Generated by Natural Phenomena

The staff reviewed the design of the facility for protecting SSCs important to safety from missiles generated by natural phenomena in accordance with SRP Section 3.5.1.4. The SRP acceptance criteria specify that the design meet GDC 2 and 4. GDC 2 requires that SSCs important to safety be protected from the effects of natural phenomena; GDC 4 requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents (LOCAs). The design is considered to be in compliance with GDC 2 and 4 if it meets RG 1.76, "Design Basis Tornado for Nuclear Power Plants," Revision 0, Positions C.1 and C.2, and RG 1.117, "Tornado Design Classification," Revision 1, Positions C.1 through C.3.

GE considered tornado-generated missiles as the most limiting hazard resulting from natural phenomena in the design of SSCs important to safety. According to the guidelines of SRP Section 3.5.1.4, this is an acceptable design basis for the standard plant. In the early SSAR amendments, the missiles considered in the ABWR design were taken from ANSI/ANS Standard 2.3. They corresponded to a design-basis tornado (DBT) with a maximum tornado wind speed of 418 km/hr (260 mph) for Region I and with a probability of occurrence of 10E-6 per year. As stated in the DSER (SECY-91-153), the staff disagreed with GE's choice of the DBT for the ABWR standard design and identified this as part of DSER Outstanding Issue 8.

The staff's position on the DBT is provided in RG 1.76. For the eastern United States, RG 1.76 specifies tornado parameters with probability of occurrence of 10E-7 per year at up to a 90-percent confidence level and maximum wind speeds in tangential and translational velocities of 531 and 113 km/hr (330 and 70 mph), respectively. However, the regulatory position in RG 1.76 has been reevaluated recently by the staff using more recent tornado data. The reevaluation is documented in NUREG/CR-4461, "Tornado Climatology of the Contiguous United States," dated May 1, 1986. The reevaluation found that the tornado strike probabilities range from near 10E-7 per year for much of the western United States to about 10E-3 per year in the central United States. The wind speeds associated with a tornado having a strike probability of 10E-7 range from less than 246 km/hr (153 mph) to 534 km/hr (332 mph). These wind speed estimates are 48 to 161 km/hr (30 to 100 mph) lower than the speed estimates presented in WASH-1300 and RG 1.76 for most of the United States. From the reevaluation, the staff concluded that it would be reasonable to reduce DBT wind speeds to 322 km/hr (200 mph) for the United States west of the Rocky Mountains and to 483 km/hr (300 mph) for the United States east of the Rocky Mountains. As a result, the Commission in its SRM of July 21, 1993, approved the staff-recommended position that a maximum tornado wind speed of 483 km/hr (300 mi/hr) be used in the DBT for ALWRs. The revised tornado parameters are also discussed in Section 2.3.1 of this report. Subsequently, GE used the modified RG 1.76 guidelines in identifying the DBT as 483 km/hr (300 mph) with a strike probability of 10^{-7} per year. The staff finds this acceptable.

In the early SSAR amendments, GE stated that the ABWR design will withstand an 1,800-kg (4000-lb) automobile, a 125-kg (280-lb), 20-cm (8-in.) armor-piercing artillery shell, and a 2.54-cm (1-in.) solid steel sphere, all impacting at 35 percent of the maximum horizontal windspeed of the DBT. GE identified these missiles as Spectrum I. GE's preliminary evaluation of the design revealed that the reactor building (RB) superstructure and roof should be thickened and the roof purlins strengthened. As a result, the seismic model should also be modified. The structural design aspects of this issue were discussed in Section 3.8.4 of the DFSER. This was DFSER Open Item 3.8.4-4. GE's resolution of DFSER Open Item 3.8.4-4 and the staff's evaluation of GE's responses to this open item are discussed in Section 3.8.4 of this report. In the DFSER, the staff noted that, as a result of these structural design changes, several SSAR sections and appendices that were affected need to be revised to reflect the aforementioned DBT and missile spectra. This was DFSER Confirmatory Item 3.5.1.4-1. SSAR Sections 3.3.2 and 3.5.1.4, specify the design basis tornado wind speed of 483 km/hr (300 mph) and missile Spectrum I of SRP Section 3.5.1.4. On the basis of this information, Confirmatory Item 3.5.1.4-1 is resolved.

Based on this information, the staff concludes that GE has adequately identified and characterized the design basis natural phenomena in accordance with the modified guidelines of RG 1.76.

Positions C.1, C.2, and C.3 of RG 1.117 state that SSCs important to safety that should be protected from the effects of a DBT are:

(1) those necessary to ensure the integrity of the RCPB,

(2) those necessary to ensure the capability to shut down the reactor and maintain it in a safe-shutdown condition (this includes both standby and cold shutdown capability), and

(3) those whose failure could lead to radioactive releases resulting in calculated offsite exposures greater than 25 percent of the guideline exposures of 10 CFR Part 100.

Compliance with Positions C.1 through C.3 of RG 1.117 was identified by the staff as a part of Outstanding Issue 8 in the DSER (SECY-91-153). In the SSAR Section 3.5.1.4 revisions that followed, GE discussed Positions C.1 and C.2. These discussions are acceptable. Subsequent to this, GE informed the staff that Position C.3 of RG 1.117 would also be met. This was DFSER Confirmatory Item 3.5.1.4-2. Systems meeting the protection guidelines of RG 1.117 have been identified in Table 3.2-1. These systems, with the exception of the standby gas treatment system (SGTS) charcoal adsorbers and certain instrumentation, are located in seismic Category I structures which are designed to withstand winds and missiles generated by the above-mentioned tornado.

As discussed in SSAR Section 10.1, safety-related instrumentation is provided in the turbine building to detect the fast closure of the turbine main steam stop and control valve oil pressure, stop valve position, turbine first-stage pressure, and main condenser pressure. This instrumentation is designed to fail safe should it be damaged due to fire, flood, missiles, or pipe failures. The staff finds this acceptable.

GE described the plant layout that provides external missile protection for the SGTS charcoal adsorber beds and the offgas system charcoal adsorber beds. The SGTS is located in the seismic Category I reactor building and is therefore protected from a tornado-generated missile. The offgas system charcoal beds are located deep within the turbine building, which is not designed to seismic Category I requirements. However, based on its review of the layout drawings of the turbine building, the staff concludes that the intervening barriers (walls and floors) between the beds and an external tornado missile provide adequate protection for the beds.

Based on this additional design information, the staff finds that the ABWR design meets the guidelines of Positions C.1, C.2, and C.3 of RG 1.117 and is acceptable. Therefore, DSER Outstanding Issue 8 and DFSER Confirmatory Item 3.5.1.4-2 are resolved.

In the DFSER, the staff noted that the COL applicant should identify missiles generated by other site-specific natural phenomena that might be more limiting than those considered in the ABWR design and should provide protection for the SSCs against such missiles. These were DFSER COL Action Items 3.5.1.4-1 and 3.5.1.4-2, respectively. To address this first COL action item, GE revised SSAR Section 3.5.4.2 by stating that the COL applicant shall identify missiles generated by other site-specific natural phenomena that may be more limiting than those considered in the ABWR design and shall provide protection for the SSCs against such missiles. This is acceptable. To address the second COL action item, GE revised SSAR Section 3.5.4.5 by stating that a turbine system maintenance program, including probability calculations of turbine missile generation meeting the minimum requirement for the probability of missile generation, shall be submitted by the COL applicant for staff review. This is also acceptable.

As a result of its review of the tornado and design information, the staff concludes that GE has identified the worst-case natural phenomena and has provided adequate protective design features to ensure that SSCs important to safety will be protected. Furthermore, GE has identified the SSCs important to safety which require protection from natural phenomena. Therefore, the staff concludes that GE meets the guidelines of RGs 1.76 and 1.117. As a result, the requirements of GDC 2 and GDC 4 are satisfied.

Verification of adequate separation of redundant divisions of safety-related systems and location of these systems in structures designed to withstand tornadic winds and missiles generated by these winds ensure that these systems will not be rendered inoperable as a result of missiles generated by natural phenomena. In the DFSER, this verification was identified as Open Item 3.5.1.4-1. Subsequently, GE submitted design descriptions and ITAAC for the buildings within the ABWR design scope. These include verifications related to missile protection for safety-related SSCs and location of these systems in structures designed to withstand tornado missiles. The adequacy and acceptability of the design description and ITAAC are evaluated in Section 14.3 of this report; therefore, DFSER Open Item 3.5.1.4-1 is resolved.

SSCs important to safety are designed to withstand the effects of natural phenomena without loss of the capability to perform their safety functions. The basis for acceptance in the staff review is the conformance of the ABWR design and design criteria for the protection from the effects of natural phenomena to the Commission's regulations as set forth in the GDC and to applicable regulatory guides. The staff concludes that GE's assessment of possible hazards due to missiles generated by the design basis tornado is acceptable and conforms to the requirements of GDC 2 and GDC 4 as related to tornado-generated missiles. This conclusion is based on the ABWR design meeting the requirements of GDC 2 and GDC 4 by meeting (a) RG 1.76, Positions C.1 and C.2 and (b) RG 1.117, Positions C.1 through C.3. Therefore, the ABWR design meets the guidelines of SRP Section 3.5.1.4 and is acceptable.

3.5.1.5 Site Proximity Missiles (Except Aircraft)

In SSAR Section 3.5.1.5, GE originally stated that "external missiles other than those generated by tornados are not considered design basis . . .," since the resultant event probability is $\leq 10^{-7}$. In the DFSER, the staff identified it as a site-specific action item for the COL applicant to address. This was DFSER COL Action Item 3.5.1.5-1. Subsequently, in SSAR Section 3.5.4.3, Amendment 33, GE stated that the COL applicant shall provide an analysis that demonstrates that the probability of missiles impacting the ABWR standard plant and causing consequences greater than those permitted in 10 CFR Part 100 guidelines is $\leq 10^{-7}$. This is acceptable.

3.5.1.6 Aircraft Hazards

In the SSAR Section 3.5.1.6, GE originally stated that "aircraft hazards are not a design-basis event . . .," since the resultant event probability is $\leq 10^{-7}$. In the DFSER, the staff identified it as a site-specific action item for the COL applicant to address. This was DFSER COL Action Item 3.5.1.6-1. Subsequently, in SSAR Section 3.5.4.3, Amendment 33, GE stated that the COL applicant shall provide an analysis that demonstrates that the probability of aircraft impacting the ABWR standard plant and causing consequences greater than those permitted in 10 CFR Part 100 guidelines is $\leq 10^{-7}$. This is acceptable.

3.5.2 Structures, Systems, and Components To Be Protected From Externally Generated Missiles

The staff reviewed the design of the facility for protecting SSCs important to safety (within the ABWR design scope) against externally generated missiles in accordance with SRP Section 3.5.2. The SRP acceptance criteria specify that the design must meet GDC 2, "Design Bases for Protection Against Natural Phenomena," and 4, "Environmental and Dynamic Effects Design Bases." The design is considered to be in compliance with GDC 2 and 4 if it meets RG 1.13, Revision 1, "Spent Fuel Storage Facility Design Basis," as it relates to the capability of the spent fuel pool systems and structures to withstand the effects of externally generated missiles and prevent missiles from contacting stored fuel assemblies; RG 1.27, "Ultimate Heat Sink for Nuclear Plants," Revision 2, as it relates to the capability of the ultimate heat sink and connecting conduits to withstand the effects of externally generated missiles; RG 1.115, "Protection Against Low-Trajectory Turbine Missiles," Revision 1, as it relates to the protection of the SSCs important to safety from the effects of turbine missiles; and RG 1.117, "Tornado Design Classification," Revision 1, as it relates to the protection of SSCs important to safety from the effects of tornado missiles. Protection from low-trajectory turbine missiles, including compliance with RG 1.115, is discussed in Section 3.5.1.3 of this report.

GE identified safety-related SSCs in SSAR Table 3.2-1. It considered the tornado-generated missiles as the limiting externally generated missiles for the ABWR design. Therefore, all the safety-related systems and components listed in the table, with the exceptions discussed in Section 3.5.1.4 of this report, are located in tornado-resistant buildings or structures. On the bases of this information, the staff concludes that the ABWR design meets the guidelines of RG 1.117.

The new and spent fuel storage systems are located in the tornado-resistant RB. Therefore, the guidelines of RG 1.13 are met.

The ultimate heat sink (UHS) and its connecting conduits are not considered in this section. GE identified the UHS as being outside the scope of the ABWR design and defined it in Section 4 of the CDM as an interfacing system for COL applicants referencing the ABWR design. Applicants referencing the ABWR design will provide the design details of the UHS for staff review at the combined license stage. At that time, the staff will evaluate the UHS design for compliance with the guidelines of RG 1.27 and the requirements of GDC 2 and GDC 4.

SSAR Section 3.5.1.4 states that the offgas system charcoal delay beds are located deep within the turbine building and consequently will be unlikely to rupture as a result of a design basis tornado missile. From the layout drawings for the turbine building, the staff concludes that the intervening walls and barriers between the delay beds and an external tornado missile provide adequate protection for the beds.

As was stated in Section 3.5.1.1 of this report, GE provided information in SSAR Section 3.5.1.1.1.3 and Figure 3.5-2 that shows the orientation of the turbine-generator in relation to the in-scope buildings which house safety-related equipment and identifies the low-trajectory missile strike zone. Based on this information, the staff concludes that no safety-related equipment is located within the turbine-generator missile strike zone. As a result, this equipment is protected from a missile ejected from the turbine-generator. Therefore, the staff concludes that the ABWR design meets the guidelines of RG 1.115.

As discussed in SSAR Section 10.1, safety-related instrumentation is provided in the turbine building to detect the fast closure of the turbine main steam stop and control valve oil pressure, stop valve position, turbine first-stage pressure, and main condenser pressure. This instrumentation is designed to fail safe should it be damaged due to fire, flood, missiles, or pipe failures. The staff finds this acceptable.

Based on the review of this information, the staff concludes that the ABWR design meets the guidelines of RGs 1.13, 1.115, and 1.117. Furthermore, the staff concludes that GE has provided adequate guidance to ensure that an applicant referencing the ABWR design will provide sufficient design information to ensure that the UHS will meet the guidelines of RG 1.27 and the requirements of GDC 2 and GDC 4. Therefore, the staff finds that the portion of the design within the ABWR scope meets the requirements of GDC 2 and GDC 4.

In the DSER (SECY-91-153), the staff identified the need for an interface requirement to protect safety-related equipment from failures of non-safety-related SSCs not housed in tornado-resistant buildings or structures (Outstanding Issue 9). Upon further evaluation, the staff determined that this action did not meet the definition of an interface requirement. This issue was reclassified as DFSER Open Item 3.5.2-1 related to ITAAC. Verification of adequate separation of redundant divisions of safety-related systems and location of these systems in structures designed to withstand externally generated missiles (including those that result from the failure of non-safety-related SSCs not housed in tornado-resistant buildings) ensure that these systems will not be rendered inoperable as a result of the missiles. In the DFSER, this verification was identified as Open Item 3.5.2-1. Subsequently, GE submitted design descriptions and ITAAC for the buildings within the ABWR design scope. These include verifications related to missile protection for safety-related SSCs and location of these systems in structures designed to withstand externally generated missiles. The adequacy and acceptability of the design description and ITAAC are evaluated in Section 14.3 of this report; therefore, DFSER Open Item 3.5.2-1 is resolved.

In the DFSER, the staff noted that the COL applicant should design SSCs outside the ABWR design scope. Any failure of these SSCs which may result in external missile generation should not prevent safety-related SSCs from performing their intended safety function. This was DFSER COL Action Item 3.5.2-1. GE has included this action item in the SSAR. This is acceptable.

In the DSER (SECY-91-153), the staff noted that GE had not listed the design-basis tornado-generated missiles considered in the ABWR design. Subsequently, GE stated that the ABWR design will withstand an 1,800-kg (4,000-lb) automobile, a 125-kg (280-lb), 20-cm (8-in.) armor-piercing artillery shell, and a 2.54-cm (1-in.) solid steel sphere, all impacting at 35 percent of the maximum horizontal windspeed of 483 km/hr (300 mph) of the DBT. These missiles are identified as Spectrum I. As stated in Section 3.5.1.4 of this report, the RB superstructure and roof were thickened, the roof purlins were strengthened, and the seismic model was modified.

As was stated earlier, verification of adequate separation of redundant divisions of safety-related systems and location of these systems in structures designed to withstand externally generated missiles ensure that these systems will not be rendered inoperable as a result of the missiles. In the DFSER, this verification was identified as Open Item 3.5.2-2. Subsequently, GE submitted design descriptions and ITAAC for the buildings within the ABWR design scope. These include verifications of divisional separation for safety-related SSCs and location of these systems in structures designed to withstand externally generated missiles. The adequacy and acceptability of the design description and ITAAC are evaluated in Section 14.3 of this report; therefore, DFSER Open Item 3.5.2-2 is resolved.

The review of the SSCs to be protected from externally generated missiles included all safety-related SSCs within the ABWR design scope provided to support the reactor facility. From review of the SSAR design criteria, design bases, and safety classifications for SSCs necessary for safe reactor shutdown, the staff concludes that the SSCs to be protected from externally generated missiles conform to the guidelines of SRP Section 3.5.2 and the requirements of GDC 2 and 4.

3.5.3 Barrier Design Procedures

Missile barriers and protection structures are designed to withstand and absorb missile impact loads to prevent damage to safety-related SSCs based on the relevant requirements of GDC 2 and 4. The staff reviewed the design of seismic Category I SSCs to determine if they are shielded from, or designed for withstanding, various postulated missiles using the guidance of SRP Section 3.5.3. SSAR Section 3.5.3 contains information on procedures used in the design of the structures, shields, and barriers to resist the effects of missiles.

For the prediction of local damage from missiles, GE provided, in SSAR Amendment 32, information on the procedures used in the design of concrete and steel structures. GE applied the empirical equations such as the modified Petry formula or U.S. Army Technical Manual TM 5-855-1 formula analytically for missile protection in concrete. The staff finds that use of the Petry formula or U.S. Army Technical Manual TM 5-855-1 formula, as verified by impact tests, and with the thickness equal to or greater than the minimum required as specified for Region II listed in Table 1 of SRP Section 3.5.3 will result in sufficient concrete barrier thickness to prevent barrier perforation and, when necessary, prevent spalling or scabbing. This is acceptable.

GE used the Stanford equations for missile penetration in steel. As discussed in SRP 3.5.3, the staff finds the use of this formula acceptable. Composite barriers are not used in the ABWR design and were, therefore, not discussed.

Regarding overall damage prediction, GE assumed that missile impact is plastic and that all of the missile's initial momentum is transferred to the structure or barrier, with only a portion of the kinetic energy absorbed as strain energy within the structure or barrier. GE evaluated the equivalent static load on the impacted area using an analysis for rigid missiles similar to the Williamson and Alvy analysis, "Impact Effect of Fragments Striking Structural Elements," November 1973. As stated in SRP 3.5.3, the staff finds the assumption of plastic collisions and use of the Williams and Alvy analytical procedure, together with the use of the permissible ductility ratio as indicated in Appendix A to SRP Section 3.5.3 to determine equivalent static loads, acceptable.

The staff finds acceptable the procedures used for determining the effects and loadings on seismic Category I structures and missile shields and barriers induced by design-basis missiles selected for the plant because they provide a conservative basis for engineering design to ensure that the structures or barriers will adequately withstand the effects of such forces.

The use of these procedures provides reasonable assurance that if a designbasis missile should strike seismic Category I structures or other missile shields and barriers, the structures, shields, and barriers will not be impaired or degraded to an extent that will result in a loss of required protection. Seismic Category I systems and components protected by these structures will, therefore, be adequately protected against the effects of missiles and will be capable of performing their intended safety functions. Conformance with these procedures is an acceptable basis for satisfying, in part, the requirements of GDC 2 and 4.

As previously discussed, GE used acceptable procedures in its barrier design. Thus, the staff finds that the barrier design is acceptable and meets SRP Section 3.5.3 and GDC 2 and 4, with respect to the capabilities of the structures, shields, and barriers to provide sufficient protection to SSCs that must withstand the effects of natural phenomena (tornado missiles), and the environmental effects of missiles, pipe whipping, and discharging fluids.

In SSAR Section 3.5.4, GE identified the responsibilities of the COL applicant for the design of barriers and protective structures to withstand the impact of postulated missiles. The staff discusses these actions in Sections 3.5.1 and 3.5.2 of this report.

SSAR Section 3.5.4.1 identifies a COL action for the barrier design of the UHS. For this action, COL applicants should meet RG 1.27, "Ultimate Heat Sink for Nuclear Power Plants," by demonstrating that the UHS and the connecting conduits are capable of withstanding the effects of externally generated missiles. This is acceptable.

3.6 Protection Against Dynamic Effects Associated With the Postulated Rupture of Piping

3.6.1 Plant Design for Protection Against Postulated Piping Failure in Fluid Systems Outside Containment

The staff reviewed the ABWR design regarding protection of SSCs important to safety against postulated piping failures in fluid systems outside the containment, but within the ABWR design scope, in accordance with SRP Section 3.6.1. Specifically, the SRP acceptance criteria specify that the design meet GDC 4, "Environmental and Dynamic Effects Design Bases," as it relates to accommodating the dynamic effects of postulated pipe rupture, including the effects of pipe whipping and discharging fluids. The design is considered to comply with GDC 4 if it conforms to Branch Technical Position (BTP) ASB 3-1, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment," with regard to high- and moderate-energy fluid systems outside the containment.

The staff will evaluate on a case-by-case basis (1) protection against postulated piping failures outside the containment for any SSCs not within the scope of the ABWR design for each application referencing the ABWR design and (2) the systems or features added to the ABWR structures by these applications.

GE discussed the analysis methodology and the effects of postulated pipe breaks in high-energy fluid systems for pipe whip, jet impingement, flooding, room pressurization, and environmental parameters such as temperature, pressure, humidity, and radiation. However, the staff stated in the DSER (SECY-91-153) that GE did not consider pipe breaks and the resulting dynamic effects in the postulation of piping failures in main steam and feedwater systems inside the main steam tunnel (MST). GE justified their exclusion by stating that the piping in these systems met the LBB criterion. The staff expects that a bona fide LBB analysis should use plant-specific data such as piping geometry, materials, fabrication procedures, loads, degradation mechanisms, and pipe support locations. In its evaluation of the LBB exclusion, which is discussed in Section 3.6.3 of this report, the staff concluded that LBB could not be considered in the analysis of pipe failures. GE subsequently provided an analysis of an MSL and a main feedwater line pipe failure in SSAR Section 6.2.3; the staff's review of this analysis is found in Section 6.2.1.7 of this report. In a meeting with the staff, GE committed to removing references to LBB from the SSAR. This was identified as Confirmatory Item 3.4.1-1 in the DFSER and is discussed in Section 3.4.1 of this report.

GE discussed pipe leakage crack events involving moderate-energy fluid systems for wetting from spray, flooding, and other environmental effects, addressing the methods for protecting the systems against the effects of piping failures. Physical separation is used to the extent practicable to prevent the loss of redundant safety-related systems (including auxiliary systems) as a result of any single postulated event. If the required spatial separation (based on specific breaks) between redundant trains or systems cannot be maintained, barriers, enclosures, shields, or restraints will be provided. Protection also includes ensuring that the equipment and components important to safety are environmentally qualified for the environment to which they may be exposed as a result of postulated piping failures, as discussed in SSAR Section 3.11, "Equipment Qualification Environmental Design Criteria," and Appendix 3I. The staff's evaluation of the protection provided against adverse environmental effects resulting from postulated piping failures is discussed in Section 3.11 of this report.

GE provided lists of safety-related systems, components, and equipment both inside and outside the containment that are protected against the effects of moderate- and high-energy piping failures by the methods previously described.

GE stated that actual piping dimensions, material properties, and the equipment and associated piping stresses and regionalized environmental conditions will be the responsibility of the referencing applicants. In earlier SSAR amendments, GE included the following as interface requirements for the COL applicant:

(1) a summary of the dynamic analyses applicable to high-energy piping systems, including sketches and data on calculated stress intensities, cumulative usage factors, and stress ranges for selecting break locations

(2) descriptions showing how safety-related systems are protected from jets, flooding, and other adverse environmental effects resulting from failures of moderate energy piping systems

(3) identification of specific features provided for protecting each of the systems listed in the ABWR SSAR tables

(4) details on protection provided against the effects of piping failures to ensure MSL feedwater and isolation valve functional capability

(5) typical examples, if any, where protection for safety-related systems and components against the dynamic effects of pipe failures includes enclosures in suitably designed structures or compartments, drainage systems, and equipment environmental qualifications

In the DSER (SECY-91-153), the staff stated that these requirements were acceptable. Upon further evaluation, the staff determined that these actions did not meet the definition of a 10 CFR Part 52 interface requirement and that GE should identify these requirements as COL action items in the SSAR stating that the applicant referencing the ABWR design should provide this information. As a result, these requirements were reclassified in the DFSER as COL Action Items 3.6.1-1, 3.6.1-2, 3.6.1-3, 3.6.1-4, and 3.6.1-5, respectively. SSAR Section 3.6.5 includes this information. This is acceptable.

In Section 3.6.2 of this report, the staff reviewed the criteria and methodology proposed by GE for the COL applicant to use to analyze the effects that breaks in high-energy fluid systems would have on adjacent safety-related SSCs with regard to pipe whip and jet impingement loads. The criteria and methodology discussed in SER Section 3.6.2 will be used by the COL applicant to ensure adequate protection against the dynamic effects of postulated ruptures of piping in the ABWR standard design.

The staff stated in the DSER (SECY-91-153) that it was unable to conclude that the ABWR design complies with GDC 4 as it relates to protection of SSCs important to safety against postulated failures outside containment. The staff's concerns were identified as Outstanding Issue 10, which consisted of the following:

(1) The staff concluded that SSAR Table 3.6-2 did not include all the systems, components, and equipment that have to be protected against piping failures. For example, the reactor service water (RSW) system and the equipment and components used to supply reactor building cooling water (RCW) system water to the residual heat removal (RHR) system, fuel pool cooling (FPC) heat exchangers, and the heating, ventilation, and air conditioning emergency chilled water (HECW) system refrigerators were not included in the SSAR table. Also, GE did not explain why certain systems such as the high-pressure core flooder (HPCF), the reactor core isolation cooling (RCIC) process sampling, and the standby liquid control (SLC) systems were not listed in the SSAR Table 3.6-4. Subsequently, GE revised these tables to include all of these systems or provided justification for not including them in the tables. This was found to be acceptable and this issue was resolved as reflected in the DFSER.

(2) In the DSER (SECY-91-153), the staff stated that GE had not provided the results of an analysis of the postulated worst-case piping failure of a moderate- or high-energy line (including total failure of non-seismic Category I piping systems) for the RCIC compartment, equipment and valve room, and other applicable areas outside the containment (e.g., RHR piping areas). GE subsequently provided subcompartment pressure analyses for compartments inside and outside of containment. The staff's review of these analyses is found in Section 6.2.1.7 of this report.

(3) As discussed above, the staff also identified that GE had not provided a subcompartment analysis for the MST. The staff was concerned that the main steam and feedwater lines would be routed in a tunnel through the control building. A steam or feedwater line failure may render the control room uninhabitable and compromise the safe-shutdown capability. Further, postulated leakage cracks in the RSW system piping may adversely affect the control room habitability systems. Subsequently, GE provided subcompartment pressure analyses (including pipe failures in the MST) in SSAR Section 6.2.3. Evaluation of the analysis can be found in Section 6.2.1.7 of this report. The vulnerability of the main control room (MCR) as a result of flooding in the MST and failure of RSW piping in the control building basement was evaluated and discussed in Section 3.4.1 of this report. Other environmental effects of pipe failures are discussed in SSAR Section 3.11. The safety evaluation can be found in Section 3.11 of this report. Based on its review of flood protection for the MCR included in Section 3.4.1 of this report, the staff concludes that the ABWR design provides adequate assurance that the MCR can be protected from the adverse effects associated with a pipe failure. This issue was resolved as reflected in the DFSER.

The staff expressed a concern about the pressure values in SSAR Table 3I.3-15. The staff noted that the values appear to reflect the zone pressures that result during accident conditions, assuming that the blowout panels function properly. A more conservative scenario would assume failure of the blowout panels. In the DFSER, the staff noted GE's commitment to modify the table to ensure that it reflects the highest anticipated pressures resulting from accident conditions, assuming failure of the blowout panels. This was DFSER Confirmatory Item 3.6.1-2. Subsequently, after discussions with GE, the staff concluded that the blowout panels fail in a safe condition. Specifically, the panels fail open. Therefore, the values in the Appendix 3I tables accurately reflect conservative design conditions; therefore, DFSER Confirmatory Item 3.6.1-2 is resolved.

Another concern expressed during meetings with GE involved the ABWR design's ability to provide adequate physical separation of safety-related systems and equipment; safety-related systems must be protected from the effects of fire, flood, missiles, pipe failures, and adverse environments to ensure their ability to shut down the reactor and mitigate the consequences of an accident. Protection is normally ensured by providing redundant safety-related systems and physical separation of the redundant systems. The ABWR provides physical separation by either spacial separation or by housing redundant divisions of each safety-related system in a separate compartment. Each compartment is designed to withstand the effects of the events previously identified. By providing adequate physical separation for each redundant safety-related division, at least one division will be available to perform its safety-related function assuming a single active failure in a division. GE included SSAR Section 3.13 that provides detailed information on divisional separation. All redundant divisions of safety-related systems are housed in separate divisional spaces inside the secondary containment. Locating these systems in these areas ensures that safety-related systems will be available to perform their safety functions given a fire, flood, missile, or pipe failure event and the adverse environmental consequences associated with such events. Each divisional space has walls, doors, floors, ceilings, and penetrations that are designed to prevent or accommodate the effects of these events and ensure system safety functions. In some cases, flood water and environments associated with failed systems in one division are allowed to reach redundant divisions. For instance flooding in the upper floors of the reactor building which may result from the failure of water systems may flow under the divisional room doors and enter other divisional areas. However, GE has evaluated the subsequent effects of these events and has shown that in no case will the water level rise to such an extent as to adversely affect equipment in the redundant division. In all cases, there is sufficient floor space and drainage capability via stairways, floor, drains, etc. to ensure that the water accumulation in any area is minimal. A second example involves the secondary containment HVAC. Unlike other plant systems, this system has common inlet and outlet headers which serve all three plant divisional areas. The staff has reviewed this arrangement and has concluded that the arrangement is acceptable for the following reasons:

(1) The interconnecting ductwork can be automatically and manually isolated during plant events.

(2) The ductwork is located high in the divisional spaces so that flood water cannot propagate from one division to another.

(3) Should a fire occur in a division, the smoke-removal mode of the HVAC system will minimize the amount of smoke which can propagate to redundant divisions (see SSAR Sections 9.4 and 9.5.1).

(4) The environmental effects of a postulated pipe failure outside containment (CUW or RCIC) are well within the environmental qualification limits for the equipment in each division.

As discussed in SSAR Section 3.13, HVAC ductwork is expected to leak or possibly fail during a pipe break event. However, failure of the ductwork will actually assist in minimizing the pressurization effects of the break by providing increased blowdown volume and providing additional vent pathways. In addition, as mentioned previously, safety-related equipment in each division is environmentally qualified for conditions worse than those expected during these pipe failures. Based on this information, the staff concludes that the interdivisional connection provided by the secondary containment HVAC system does not prevent the full safety function capability of any safety-related systems.

As discussed in SSAR Section 10.1, safety-related instrumentation is provided in the turbine building to detect the fast closure of the turbine main steam stop and control valve oil pressure, stop valve position, turbine first-stage pressure, and main condenser pressure. This instrumentation is designed to fail safe should it be damaged due to fire, flood, missiles, or pipe failures. The staff finds this acceptable.

As a result of its review of this information, the staff concludes that GE has provided sufficient design information and supporting analyses to ensure that the guidelines of ASB BTP 3-1 are met. Specifically, the ABWR design provides protection of safety-related equipment from the effects of high- and moderate-energy pipe failures by separation of redundant divisions of these systems, by distance, or by locating safety-related systems in enclosures that provide the necessary protection. Furthermore, all safety-related systems and components meet the guidelines of RG 1.29 and are designed to seismic Category I standards. Piping failures are postulated in accordance with BTP MEB 3-1, pipe restraints are provided in accordance with the guidelines in SRP Section 3.6.2 (see Section 3.6.2 of this report), and the effects of the postulated piping failures, including those from nonseismic systems, have been considered. Based on this information, the staff concludes that the guidelines of BTP ASB 3-1 are met, and therefore, the requirements of GDC 4 are met.

GE originally submitted the design descriptions and the ITAAC for systems within the ABWR design scope as well as for piping. These included verifications related to protection of safety-related equipment from pipe failures and their effects. During the DFSER preparation, the staff review was in progress. This was DFSER Open Item 3.6.1-1. Subsequently, GE provided revised design descriptions and ITAAC. The adequacy and acceptability of the design description and the ITAAC are evaluated in Section 14.3 of this report; therefore, DFSER Open Item 3.6.1-1 is resolved.

The review of the plant design for protection against postulated piping failures outside containment included all high- and moderate-energy piping systems located outside containment. The review of these high- and moderate-energy systems for the ABWR included layout drawings, P&IDs, and descriptive information.

The staff concludes that the ABWR design as it relates to the protection of safety-related structures, systems, and components from the effects of piping failures outside containment meets the requirements of GDC 4 with respect to accommodating the effects of postulated pipe failures and the guidelines of SRP Section 3.6.1. As a result, the staff concludes that the ABWR design is acceptable.

3.6.2 Determination of Rupture Locations and Dynamic Effects Associated With the Postulated Rupture of Piping

GDC 4 requires that SSCs important to safety be designed to be compatible with and to accommodate the effects of the environmental conditions resulting from normal operations, maintenance, testing, and postulated accidents, including loss-of-coolant accidents (LOCAs). It also requires that they be adequately protected against 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 plant. In accordance with SRP Section 3.6.2, the staff reviewed GE's proposed criteria and methodology for the COL applicant to analyze and ensure adequate protection against and the dynamic effects that breaks in high-energy fluid systems would have on adjacent safety-related SSCs with regard to pipe whip and jet impingement loadings. Pipe whip need only be considered for those high-energy piping systems having fluid reservoirs with sufficient capacity to develop a jet stream. GE correctly used the criteria for determining high- and moderate-energy lines in SRP Section 3.6.1, Branch Technical Position (BTP) ASB 3-1, in SSAR Section 3.6.2.1. GE listed all high-energy systems in SSAR Tables 3.6-3 and 3.6-4.

One of the guidelines in SRP 3.6.2, Branch Technical Position (BTP) MEB 3-1, states that the analyses for the maximum stresses, stress ranges, and usage factors to be used for determining postulated high- and moderate-energy pipe break and crack locations should be based on loads that include the OBE. In SECY-93-087, the staff recommended the elimination of the OBE in the design process on the basis that it would not result in a significant decrease in the overall plant safety margin. The detailed basis for the staff's recommendation is discussed in Section 3.1.1 of this report. In Section 3.1.1.3 of that discussion, the staff includes acceptable deviations from SRP 3.6.2 that result from pipe break postulation without the OBE. As stated in Section 3.1.1.9, "Conclusions," of this report, GE has incorporated these acceptable deviations in Sections 3.6.1 and 3.6.2 of the SSAR. Therefore, the staff's evaluation of these sections of the SSAR have been based on the Commission-approved staff recommendations.

For ASME Class 1 piping, the staff position for postulating pipe breaks is delineated in SRP 3.6.2, BTP MEB 3-1. Before NRC issued Revision 2 of BTP MEB 3-1 in June 1987, breaks were postulated at intermediate locations between terminal ends of a pipe run if the maximum stress range as calculated by Eq.(10) $> 2.4\ S_m$ and if the maximum stress range as calculated by either Eq. 12) or (13) $> 2.4\ S_m$, where Eqs. (10), (12), and (13) and S_m are as defined in ASME Section III, Subsection NB 3653. This staff position was implemented in many plants operating today. In Revision 2 of BTP MEB 3-1, the same criteria were maintained for break exclusion in the containment penetration areas. However, for other areas, the criteria were revised to require that breaks be postulated at any intermediate locations when only Eq. (10) $> 2.4\ S_m$. The use of Eqs. (12) and (13) was eliminated. This resulted in an inconsistency in the Revision 2 criteria in that they allow higher limits in the containment penetration areas than in other areas. The break exclusion areas should provide a margin greater than (or at least equal to) the margin for areas outside the break exclusion area. To determine the impact of this inconsistency, the staff obtained several independent analyses for both BWRs and PWRs that compared the number of postulated pipe breaks resulting from the use of Revisions 1 and 2 criteria. These analyses indicated that the Revision 2 criteria will result in a significant increase in the number of postulated breaks, which may be counterproductive in terms of improving plant safety. Therefore, the staff recommended that SRP 3.6.2 be revised to reinstate the Revision 1 criteria related to allowing the use of Eqs. (12) and (13) for the postulation of intermediate pipe breaks in ASME Class 1 piping systems. Sections 3.6-1 and 3.6-2 in the SSAR contains criteria that are consistent with the Revision 1 criteria as previously discussed. Therefore, the staff concludes that this is an acceptable deviation from SRP 3.6.2, Revision 2.

In the ABWR standard plant, breaks are not postulated in those portions of high-energy piping between the containment isolation valves outside and inside the containment that are designed to meet ASME Code, Section III, Article NE-1120 and the additional design guidelines in SRP Section 3.6.2, including BTP MEB 3-1, (Rev. 2). Section 3.6.2.1.4.2 of the SSAR describes acceptable actions to address all of the applicable guidelines in SRP Section 3.6.2. Two of these guidelines are discussed below.

1. SRP 3.6.2 recommends that an augmented inservice inspection program be implemented for those portions of piping within the break exclusion region. An augmented inservice inspection was DFSER COL Action Item 14.1.3.3.7.2-1. In a letter dated March 17, 1993, GE revised SSAR Section 3.6.2.1.4.2, Item (7) and added SSAR Section 3.6.5.3 to state that the COL applicant shall perform a 100-percent volumetric examination of circumferential and longitudinal pipe welds in the break exclusion region during each inspection interval as defined in Article IWA-2400, ASME Code, Section XI. The staff finds that this action for the design and examination of high-energy piping in the containment penetration area meets SRP Section 3.6.2 and is acceptable. GE has included this action item in Section 3.6.2.1.6.2 of the SSAR. The staff's resolution of this DFSER COL action item is also discussed in Section 3.12.7.2 of this report.

2. SRP 3.6.2 contains design, testing, and examination guidelines for guard pipes in the containment penetration areas. SSAR Section 3.6.2.4 states that the ABWR primary containment does not require guard pipes. This is because the ABWR design does not contain guard pipes as defined in Section 3.6.2.4 of RG 1.70, "Standard Format and Content of Safety Analysis Report for Nuclear Power Plants," Revision 3: "a guard pipe is a device to limit pressurization of the space between dual barriers of certain containments to acceptable levels." The staff notes that SRP 3.6.2 uses the term "guard pipe" in a broader context than that in RG 1.70 to include all applicable sleeves in the containment penetration area. Section 3.6.2.1.4.2(6) in the SSAR provides design, testing, and examination requirements for such sleeves. These requirements are consistent with the guidelines in SRP 3.6.2 and are acceptable.

For ASME Code Class 1, 2, and 3 and non-seismic Category I high- and moderate-energy lines that are not in the containment penetration area, SSAR Section 3.6.2 presents the criteria for determining postulated rupture and crack locations and the methodology used to evaluate the dynamic effects of pipe whip, jet thrust, and jet impingement that result from such breaks.

In the DFSER, the staff noted that according to SRP Section 3.6.2, the SSAR should include the following:

- sketches of applicable piping systems showing the location, size, and orientation of postulated pipe breaks and the location of pipe whip restraints and jet impingement barriers
- a summary of the data developed to select postulated break locations, including calculated stress intensities, cumulative usage factors, and stress ranges

This was DFSER COL Action Item 14.1.3.3.7-1. In SSAR Section 3.6.5.1, "Details of Pipe Break Analysis Results and Protection Methods," Amendment 31, GE stated that the COL applicant shall provide this information. This is acceptable. Resolution of this item is also discussed in Section 3.12.7 of this report.

In the DFSER, the staff identified an Open Item 3.6.2-1 regarding the computer programs for pipe whip analyses and the design methodology for pipe whip restraints applicable to the ABWR plant design. This open item was identified during an audit of high-energy line break criteria and pipe whip analyses. The audit was performed at the offices of GE in San Jose, California on March 23 through 27, 1992, in which GE disclosed that their pipe whip sample analysis was not yet complete but provided descriptions of the methods of analysis and design requirements for pipe whip restraints to be utilized in the sample analysis. The staff found these methods of analyses and design requirements were not in accordance with the corresponding methods and requirements in SSAR Amendment 17, Sections 3.6.2.2.2 and 3.6.2.3.3, respectively. Subsequently, in response to follow-up audits performed at the offices of GE in San Jose, California on July 28 through 31, 1992, and November 16 and 17, 1992, GE submitted Amendment 26 to the SSAR and provided

SSAR markups in a letter dated May 26, 1993. These changes revised SSAR Section 3.6.2 extensively to provide comprehensive pipe whip analysis methods and whip restraint design requirements. In the May 26, 1993 letter, GE also submitted sample analysis GE-NE-123-E070-0493, "Sample Analysis for the Effects of Postulated Pipe Break, ABWR Main Steam Piping." All of the above information provided, in aggregate, the following responses to DFSER Open Item 3.6.2-1:

1. Section 3.6.2.2.2 of the SSAR was extensively modified to permit alternative analytical approaches in accordance with ANSI/ANS 58.2 Paragraphs 6.3.1 through 6.3.5. These alternative approaches include the use of the ANSYS and PDA computer programs which were used in the GE pipe whip sample analyses. This response adequately addresses the staff's original concern about the use of the PDA program alone for pipe whip analyses.

2. Section 3.6.2.3.3 of the SSAR was modified to include design requirements for pipe whip restraints with crushable material and rigid restraints in addition to U-bar pipe whip restraints. This response results in the SSAR being consistent with the information presented to the staff during the audits mentioned, and is acceptable.

3. Appendix 3L "Procedure for Evaluation of Postulated Ruptures in High Energy Pipes" was added to the SSAR. This appendix defines an acceptable procedure for evaluation of dynamic effects of fluid dynamic forces resulting from postulated ruptures in high energy piping systems. The four major steps in the evaluations included (1) the identification of rupture locations and rupture geometry, (2) the design and selection of pipe whip restraints, (3) the procedure for dynamic time-history analysis with simplified models, and (4) the procedure for dynamic time history analysis using detailed piping models.

4. The sample analysis, GE-NE-123-E070-0493, is intended to be illustrative of the GE pipe break analysis method and, although not a part of the SSAR, it supplements Appendix 3L, which is discussed above. The analysis documented the GE procedures for (1) calculating thrust forces at break locations, (2) performing nonlinear time-history pipe whip analyses, (3) demonstrating compliance with the SRP Section 3.6.2 stress requirements for piping near containment penetration areas, and (4) demonstrating that the GE PDA computer program was adequate for selecting the size of pipe whip restraints. The staff's review found that these GE procedures are acceptable, and the intended objectives of the sample analysis were achieved.

GE included the applicable information in Section 3.6.2 of the SSAR, and therefore, DFSER Open Item 3.6.2-1 is resolved.

In the DFSER, the staff identified Open Items 3.6.2-2 and 14.1.3.3.7-1 about the edition of the ANSI/ANS-58.2 standard referenced in Section 3.6.2.2.1 of the SSAR, Amendment 17, and inconsistencies between the criteria for evaluating the effects of fluid jets on essential SSCs specified in Section 3.6.2.3.1 of the SSAR, Amendment 17, and corresponding criteria specified in SRP 3.6.2 and ANSI/ANS-58.2, 1988 Edition. The staff's evaluation of this issue is discussed in Section 3.12.7 of this report.

SRP Section 3.6.2 states that if a structure separates a high-energy line from an essential component, the separating structure should be designed to withstand the consequences of the pipe break in the high-energy line which (could) produce the greatest effect at (to) the structure. This is irrespective of the fact that the pipe rupture criteria in SRP Section 3.6.2 might not require such a break location to be postulated.

For the ABWR, the structures are designed to withstand the dynamic effects of pipe breaks where the pipe rupture criteria require break locations to be postulated. In addition, for areas where physical separation of redundant trains is not practical, a high-energy line separation analysis (HELSA) is performed to determine which high-energy lines meet the spatial separation requirement and which lines require further protection. For the HELSA evaluation, which is discussed in Section 3.6.1.3.2.2 of the SSAR, no particular break points are evaluated. Breaks are postulated at any point in all of the high-energy piping systems listed in SSAR Tables 3.6-3 and 3.6-4, and any structure identified as necessary by the HELSA evaluation are designed for worst-case loads. This was DFSER COL Action Item 14.1.3.3.7.3-1. SSAR Section 3.6.5.1, Item (8) of the SSAR states that the HELSA will be performed by the COL applicant as described, which is acceptable. The staff's resolution of this action item is also discussed in Section 3.12.7.3 of this report.

Using the above HELSA evaluation, the staff finds that an adequate level of protection is provided to ensure that the safety-related function of components, systems, and equipment will not be adversely affected by a postulated break in any ABWR high-energy piping systems. Plant arrangement provides physical separation to the extent practical and the HELSA evaluation ensures that no more than one redundant train can be damaged. If damage could occur to more than one division of a redundant safety-related system within 9.14 m (30 ft) of any high-energy piping, other protection devices such as barriers, shields, enclosures, deflectors, or pipe whip restraints are used. When necessary, the protection requirement are met through the use of walls, floors, columns, abutments, and foundations. Thus, the staff finds that the HELSA criteria satisfy the intent of the SRP 3.6.2 guideline by ensuring that structures are adequately designed to withstand the consequences of a worst-case pipe break with no adverse impact on the safety-related function of systems, components, and equipment and are acceptable.

3.6.2.1 Conclusions

From these evaluations, the staff concludes that the criteria for postulating pipe rupture and crack locations and the methodology for evaluating the subsequent dynamic effects resulting from these ruptures comply with SRP Section 3.6.2 and meet GDC 4. The staff's conclusion is based on the following.

The proposed pipe rupture locations will be adequately determined using the previous staff-approved criteria and guidelines. GE has sufficiently and adequately defined the design methods for high-energy mitigation devices and the measures to deal with the subsequent dynamic effects of pipe whip and jet impingement to provide adequate assurance that, upon completion of the high-energy line break analyses as part of the ITAAC process, the ability of safety-related SSCs to perform their safety functions will not be impaired by the postulated pipe ruptures.

The provisions for protection against the dynamic effects associated with pipe ruptures of the RCPB inside the containment and the resulting discharging fluid provides adequate assurance that design-basis LOCAs will not be aggravated by the sequential failures of safety-related piping and that the performance of the emergency core cooling system (ECCS) will not be degraded as a result of these dynamic effects.

The arrangement of piping and restraints and the final design considerations for high- and moderate-energy fluid systems inside and outside the containment, including the RCPB, will be the responsibility of the COL applicant. These staff-approved high-energy line break criteria and guidelines will be used to assure that the SSCs important to safety that are in close proximity to the postulated pipe ruptures will be protected. GE has developed an ITAAC to verify that the safety of the plant will not be adversely affected by the dynamic effects resulting from the postulated pipe break, as discussed in Section 3.12.7 of this report. Using these criteria and guidelines will ensure that the consequences of pipe ruptures will be adequately mitigated so that the reactor can be safely shut down and can be maintained in a safe-shutdown condition in the event of a postulated rupture of a high- or moderate-energy piping system inside or outside the containment.

3.6.3 Leak-Before-Break Evaluation Procedures

SSAR Section 3.6.3 describes the evaluation procedures for the ABWR LBB methodology. The application of the LBB methodology to piping systems is permitted by GDC 4, which states, in part:

> . . . dynamic effects associated with postulated pipe ruptures in nuclear power units may be excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of a fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping.

The analyses referred to in GDC 4 should be based on such plant-specific data as piping geometry, materials, piping loads, and pipe support locations. The staff must review the LBB analyses for specific piping designs before an applicant can exclude the dynamic effects from the design basis for the piping system.

In the DFSER, the staff noted that COL applicants seeking approval of the LBB approach for high-energy piping systems in the ABWR plant should submit an LBB plant-specific analysis in accordance with GDC 4 and include the information stated in the previous discussion. This was DFSER COL Action Item 3.6.3-1. In SSAR Section 3.6.5.2, GE stated that the COL applicant shall prepare a plant-specific LBB analysis report and submit the report to the NRC staff for approval. This is acceptable.

Although the staff is currently using the methodology and acceptance criteria provided in SRP Section 3.6.3 and NUREG-1061, Volume 3, the staff recognizes that the LBB technology is continually evolving. Therefore, the staff will review LBB requests for the ABWR plant on a case-by-case basis using the staff's methodology and acceptance criteria in effect at the time of the submittal.

3.7 Seismic Design

The staff reviewed the seismic design adequacy of the ABWR standard plant using SRP Sections 3.7.1 through 3.7.4 as the basis, and considered GE's responses to the open items, confirmatory items and COL action items identified in the DSER (SECY-91-153) and the DFSER. In addition, the staff conducted two design calculation audits at GE's office in San Jose, California, and two design calculation audits at Bechtel Power Corporation (consultant to GE) in San Francisco, California. The first design calculation audit, conducted November 28 through 30, 1989, covered the design of the containment and RB. The staff's findings and concerns from the first design calculation audit and GE's subsequent resolutions are contained in Appendix 3.7A to the DSER (SECY-91-153). The second design calculation audit, conducted on March 30 through April 2, 1992, covered primarily the design of the RB, control building, and radwaste building substructure. The staff's findings and concerns from the second design calculation audit were documented in the audit report issued on May 15, 1992. The third and forth audits covered three major areas: (1) seismic reanalysis of the safety-related structures, (2) GE's response to the DFSER open and confirmatory items, and (3) designs of the safety-related structures, including the RB, the containment structure, and the control building. These two design calculation audits were conducted at the Bechtel (San Francisco) office on October 12 through 15, 1992 and February 22 through 25, 1993, respectively. These audits were to determine if the structures of the ABWR standard plant are adequately designed and if the commitments documented in the SSAR are properly implemented.

In the DFSER, the staff's evaluation covered both OBE and SSE. However, as discussed in Section 3.1.1 of this report, by implementing the Commission-approved staff recommendations on OBE elimination, unless otherwise denoted, only the SSE seismic design is evaluated on the following.

3.7.1 Seismic Design Parameters

In the SSAR, the input seismic design response spectra for the SSE are defined at plant finished grade in the free field. These design response spectra comply with the ground motion response spectra recommended in RG 1.60, "Design Response Spectra for Seismic Design of Nuclear Power Plants," Revision 1. The peak horizontal as well as the peak vertical ground acceleration (PGA) is 0.3*g*. For the standard plant design, SSE (i.e., RG 1.60 response spectra anchored to 0.3*g*), was employed to calculate the responses of the SSCs. The staff's evaluation of the design ground motion for the SSE is discussed in Section 2.6.3 of this report.

In SSAR Section 3.7.1, Amendment 33, GE stated that the three components of the synthetic SSE ground motion time history used for the seismic analysis and design of the ABWR seismic Category I SSCs are adjusted in amplitude and frequency to obtain response spectra for damping ratios of 2-, 3-, 4-, 5-, and 7-percent of critical damping to envelop the SSE design ground response spectra at a sufficient number of frequency points as recommended by SRP Section 3.7.1 plus three additional frequencies at 40, 50, and 100 Hz. The power spectral density function (PSDF) of the two horizontal components of the synthetic SSE ground motion time history envelops the target PSDF specified in Appendix A to SRP Section 3.7.1 for a frequency range of 0.3 to 24 Hz. For the vertical component of the ground motion time history, GE followed the guideline of Appendix A to SRP 3.7.1 and developed a target PSDF for the vertical ground motion. GE also showed that the vertical component of the ground motion time history (synthetic time history) satisfied the PSDF enveloping criteria. The details of developing the target PSDF and the demonstration of PSDF enveloping process are described in a GE submittal dated May 7, 1993, (GE's responses to Item 3 of the audit report dated February 22, 1993). In addition, GE showed that the three components of synthetic time history are statistically independent of each other in that the cross-correlation coefficients at zero-time lag between these components are less than 0.16.

From this evaluation above, the staff concludes that the SSE input ground motion (the design ground response spectra and the ground motion time history) as documented in SSAR Section 3.7.1 meets the guidelines of SRP 3.7.1 and RG 1.61 and, therefore, is acceptable.

To define the design ground motion (ground response spectra and the associated synthetic time history) at plant-finished grade in the free field, in accordance with the SRP guidelines, is acceptable for the purpose of developing the envelope of seismic responses (both structural member forces and FRS) for the design of the standard plant. However, when confirming the adequacy of the standard plant design for a specific shallow soil site, the COL applicants should define the site-specific ground response spectra and associated design time history as the free field motion at a level that complies with the guidelines of SRP Section 3.7.1.I.1. The details of the staff's review of GE's compliance with these guidelines and the staff's conclusion are discussed under "Confirmation of Plant-Specific Seismic Design Adequacy" in Section 3.7.2 of this report.

The damping ratios used in the analysis of the ABWR seismic Category I structures comply with the SSE damping ratios specified in RG 1.61, "Damping Values for Seismic Design of Nuclear Power Plants." For soils, GE determined damping values (soil material damping and energy dissipation as a result of wave propagation) on the basis of the soil shear strains induced in the free field. The approach for considering the soil damping complies with the SRP Section 3.7.2 guidelines and is, therefore, acceptable.

The ABWR seismic Category I structures have reinforced concrete mat foundations that are designed to be supported on soil, rock, or compacted backfill. Appendix 3H to the SSAR, Amendment 33, specifies the following embedment depths from the plant finished grade to the bottom of the basemat for each seismic Category I structure:

- 25.9 m (85 ft) for RB complex (shield building, containment structure, drywell, and reactor pedestal)
- 23.2 m (76.1 ft) for control building
- 16.0 m (52.5 ft) for radwaste building substructures

These three buildings are designed to have independent foundations. During the design calculation audits, GE also discussed the use of these embedment depths for the seismic analysis to determine the seismic soil-structure interaction (SSI) effects for all seismic Category I structures except the radwaste building that is assumed to be surface-founded in the seismic analysis. The staff's evaluation regarding the adequacy of the SSI analysis, including the consideration of these embedment depths, is discussed in Section 3.7.2 of this report. However, because the depth of the embedment is an important aspect of the seismic design, the variation in the depth of embedment of an as-built plant beyond the tolerance limit of 0.3 m (1 ft) should be verified by calculations to reconcile the difference. The embedment depths verification is part of building-specific ITAAC. The adequacy and acceptability of the ABWR design description and ITAAC are evaluated in Section 14.3 of this report.

Bounding Site Parameters

In the DFSER, the staff identified COL Action Item 3.7.1-1 to anchor the design basis OBE to a peak ground acceleration (PGA) of 0.15*g*. Recently, the Commission approved the staff recommendation of eliminating OBE from the ABWR design. The staff's evaluation of this issue is discussed in Section 3.1 of this report, and COL Item 3.7.1-1 is withdrawn and resolved.

In SSAR Section 2.3.1.2, Amendment 33, GE stated that the COL applicant will use the following bounding site conditions to confirm the adequacy of the ABWR standard plant seismic design for a specific site:

- The site-specific ground motion response spectra are bounded by the RG 1.60 design response spectra anchored to 0.3*g*.
- For a shallow soil site, the site-specific ground response spectra and the associated time history should be specified as the free field motion at a level that complies with the guidelines of SRP Section 3.7.1.
- The site soil static bearing capacity at the foundation level of the reactor and control building is 0.72 MPa (15 kip/ft^2) minimum.

The bounding site conditions are consistent with those discussed in SSAR Section 3.7.1 and are acceptable.

Certain site-specific SSCs for ABWR plants that are not part of the certified design may be designed using site-specific spectra. To develop these spectra, the horizontal and vertical free field ground surface site-specific ground motion spectra for a controlling earthquake for a site should be obtained using the procedures of SRP 2.5.2. The maximum spectral amplitude of these spectra in the frequency range 5 to 10 Hertz should be obtained. Both the horizontal and vertical design response spectra for the ABWR certified design and the RG 1.60 shapes anchored to 0.3*g* peak ground accelerations should be scaled throughout their entire frequency range in such a manner that the minimum spectral amplitudes of the certified design spectra are equal to the maximum amplitudes of the horizontal and vertical site-specific ground motion spectra, respectively, in the 5- to 10-Hertz frequency range. The resulting design response spectra should be used as the minimum for the design of site-specific SSCs for ABWR plants.

The staff concludes that GE meets the relevant requirements of GDC 2 and Appendix A to 10 CFR Part 100 by appropriate consideration of the most severe earthquake, SSE, to which the ABWR seismic Category I SSCs are expected to be subjected. GE meets these requirements by the use of (1) SSE design response spectra that comply with RG 1.60, (2) synthetic ground motion time histories that comply with the design response spectrum and PSDF enveloping criterion of SRP Section 3.7.1, and (3) specific percentage of critical damping values in the seismic analysis of ABWR seismic Category I SSCs that conforms to RG 1.61. This ensures that the seismic inputs to the ABWR seismic Category I SSCs are adequately defined to form a reasonable basis for the design of such SSCs to withstand seismic loadings.

3.7.2 Seismic System Analysis

The staff's review of the seismic analysis of the seismic Category I SSCs includes the seismic analysis methods and acceptance criteria used for the ABWR seismic Category I structures, the reactor pressure vessel (RPV) and containment internal structures design. The radwaste building, which is not a seismic Category I structure and does not house any safety-related equipment and systems, is included in this review because GE elected to design this structure for the SSE seismic loads to ensure that the embedded portion of the building retains its structural integrity during and after an SSE, and to prevent unacceptable leakage of the radwaste material outside the building. The seismic analyses of the turbine building and the condenser, which is located inside the turbine building but has a separate foundation, are also included in this review although these two structures are not seismic Category I. They are within the scope of this review because a portion of the MSL, which is required to be designed for the SSE based on the discussion in Section 3.2.1 of this report, is inside the turbine building, and one end of this portion of the MSL system and branch lines is anchored at the turbine building and the condenser.

GE originally analyzed all seismic Category I structures for the OBE ground motion only and obtained the SSE responses, including the FRS of the structures by multiplying the corresponding OBE responses by a factor of two. The results of the staff review were documented in the DSER (SECY-91-153), DFSER, and the design calculation audit reports. In these reports, a number of open, confirmatory, and COL action items were identified.

In response to the DSER (SECY-91-153) and DFSER open, confirmatory, and COL action items and to address the concerns identified in the audit reports, GE initiated a seismic reanalysis for all seismic Category I structures. GE then used the results of this reanalysis to replace the original analysis results as part of the basis for the design certification application. GE explained the reasons for performing this analysis as follows:

(1) As a result of changing the design basis tornado wind speed from 418 km/hr (260 mph) to 483 km/hr (300 mph) and adopting the use of tornado missile Spectrum I per SRP Section 3.5.1.4, the RB superstructure and roof were thickened and the roof purlins strengthened. These structural design upgrades resulted in a need to modify the seismic models.

(2) The original SSE responses of the ABWR seismic Category I structures, including the FRS, were obtained by multiplying the corresponding OBE analysis results by a factor of two. As a result, a significant design margin was included in the SSE responses. To eliminate the excessive design margin, GE initiated the seismic reanalyses for the RB complex and the control building to take advantage of the Commission-approved staff recommendations on OBE elimination.

(3) The original building responses used for the seismic input to subsystems such as piping were obtained by enveloping the analysis results from two dynamic models, namely, models with and without the steel stabilizer truss between the RSW and reinforced concrete containment vessel (RCCV). Because of the protection to be provided for the personnel during reactor hot standby and shutdown for maintenance and refueling, the RSW height was raised to the height of 15 cm (6 in.) below the RCCV top slab. This change eliminated the steel stabilizer truss from design and, thus, the stabilizer truss was deleted to result in the revised model for seismic reanalysis.

(4) To address the NRC staff's concern about the use of a two-dimensional (2-D) dynamic model for the seismic analysis of seismic Category I structures, only three dimensional (3-D) soil-structure system models were used in the seismic reanalyses.

In the seismic reanalysis of the seismic Category I structures, GE regenerated the three components of the ground motion time history. In SSAR Section 3.7.1 and Figures 3.7.6 through 3.7.20 and Figures 3.7.24 through 3.7.26, GE compared the design ground response spectra and the target PSDFs with the respective response spectra and the PSDFs obtained from these three components of the ground motion time history. As shown in these figures, the three components satisfy both the response spectrum enveloping criterion for all damping ratios to be used in the analysis and the PSDF enveloping criterion provided in SRP Section 3.7.1. Therefore, they are acceptable to the staff.

Regarding the seismic analysis of the ABWR plant structures including all seismic Category I structures, the staff found the procedures used for structural modeling, seismic soil-structure interaction analysis, development of FRS, inclusion of the effects of variation in parameters on FRS, inclusion of torsional effects, evaluation of dynamic stability of buildings (such as overturning and sliding), and determination of composite damping acceptable. The staff's basis for the acceptance of these techniques and the review of the analysis and design for each building are discussed in detail latter in the building-specific evaluations of this section.

As discussed in SSAR Section 3.7.2 and Appendix 3A and Appendix 3H to the SSAR, Amendment 33, GE performed dynamic analyses of the seismic Category I structures to generate the SSE responses (structural member forces and FRS) on a linear elastic basis. The seismic responses were calculated for the two horizontal directions and the vertical direction. GE did not use the technique of constant static factors for computing the vertical responses. The structural damping ratios used by GE complied with those specified in RG 1.61. For the structure of structural materials with different damping ratios and for the modal response method of analysis used, GE calculated a composite modal damping ratio using the strain energy technique. The techniques used for the dynamic analyses of structures discussed above are consistent with the guidelines of SRP Section 3.7.2 and, therefore, are acceptable.

The 3-D time history method applied in the frequency domain forms the basis for the dynamic analyses of all major seismic Category I structures. The technique used for performing the dynamic analysis in the frequency domain meets the guidelines of SRP Section 3.7.2 and, therefore, is acceptable. GE generated the FRS, which serve as inputs for the seismic analysis, design, and test verifications of systems and components, from the time history method of dynamic analysis. To account for variation in the structural frequencies as a result of uncertainties in the material properties of the structures and in the modeling and analysis techniques, GE broadened the peaks of FRS. To develop the FRS envelopes for the subsystem design of the standard plant, GE, as discussed in SSAR Section 3A.10.2, Amendment 33, (1) calculated the FRS for the damping ratios of 2-, 3-, 5-, and 10-percent from all 3-D analysis cases, using computer code SASSI, (2) developed the FRS envelopes from the FRS for all site conditions at the required locations in each of the three directions, and (3) at each location, developed the bounding horizontal FRS envelopes from the FRS envelopes in the two horizontal directions. In addition, GE applied a peak broadening of 15 percent to the bounding horizontal FRS envelopes and the vertical FRS envelopes to account for the uncertainties caused by structural modeling, material properties, and soil modulus. In SSAR Section 3.7.2, GE also discussed two options for the design of seismic Category I components regarding the peak broadening: (1) if a detailed parametric variation study is made, the minimum peak broadening ratio can be 10 percent, and (2) in lieu of peak broadening, the peak shifting method included in Appendix N to ASME Section III can be used. GE's process for developing the FRS envelopes meets the guidelines of RG 1.122, "Development of Floor Design Response Spectra for Seismic Design of Floor Supported Equipment or Components," Revision 1, and SRP Section 3.7.2, and is acceptable. The acceptability of Appendix N to ASME Section III is discussed in Section 5.2.1.2 of this report.

From comparing the natural frequencies obtained from a 2-D fixed base building model and a 3-D fixed base building model with the embedment effect included, GE found that the torsional effect resulting from the eccentricity between the center-of-mass and center-of-rigidity of the seismic Category I structures (RB, control building, and embedded portion of radwaste building) on the seismic responses is negligible because of the symmetry in the geometrical layout of the buildings. Therefore, in SSAR Section 3.7.2.11, Amendment 33, GE stated that for the ABWR seismic Category I structures, the actual eccentricities are negligible and the torsional moments are due to accidental torsion only. From its review, the staff agreed that the effect of the building eccentricity is negligible. For the seismic design of structures, GE followed the approach and the procedures that comply with the guidelines of SRP Section 3.7.2 and applied an accidental eccentricity equal to 5 percent of the maximum building dimension at each floor to calculate the seismic shear for distribution to the lateral load resisting structural elements. GE evaluated the stability of the structure against seismic overturning by requiring a minimum factor of safety of 1.1 between the potential energy needed to overturn the structure and the maximum kinetic energy of the structure during the SSE. These approaches comply with the guidelines of SRP Section 3.7.2 and are acceptable. In early SSAR amendments, GE did not describe the procedure for determining the stability of the structure against seismic sliding. This was DFSER Open Item 3.7.2-1. In Appendix 3H to the SSAR, Amendment 33, GE provided the analysis procedures used for the dynamic overturning of the RB, control building, and radwaste building and the evaluation results (safety factors against overturning, sliding, and flotation) of these three buildings. From the review of Appendix 3H to the SSAR and the design calculation audit conducted on February 22 through 25, 1993, the staff concludes that the reactor, control, and radwaste buildings will be dynamically stable under the specified SSE. Therefore, DFSER Open Item 3.8.5-1 is resolved. This concern is also discussed in Section 3.8.5 of this report.

From evaluation of the general approach for the seismic analysis of the seismic Category I structures discussed above, the staff concludes that the procedures used by GE to (1) combine the modal responses, (2) combine the effects of the three earthquake components, (3) account for the effects of variation in parameters on FRS envelopes, (4) include torsional effect in the seismic design of structures, (5) evaluate stability of structures against seismic overturning and sliding, and (6) determine composite damping ratios for structures comply with the guidelines of SRP Section 3.7.2 and the applicable regulatory guides and are, thus, acceptable.

During the design calculation audit conducted on February 22 through 25, 1993, the staff was concerned about that, in its calculation of seismic loads from the live loads, GE reduced the live load on a global basis but did not provide the basis for the reduction or provide the criteria for the design of local structural elements such as slabs, beams, and columns. In SSAR Sections 3.8.1.3.1, 3H.1.4.3.1, 3H.2.4.3.1, and 3H.3.4.3.1, Amendment 33, GE stated that the floor area live load shall be omitted from areas occupied by equipment whose weight is specifically included in dead load and that the live load shall not be omitted under equipment where access is provided. For the computation of global seismic loads, only a portion of the live load designated as "Lo" is used because of the overall light occupancy of power plants. The "Lo" loads are established in accordance with the layout and mechanical requirements. However, the live loads used in the load combinations for the design of local structural elements such as slabs and beams are the full values. In the design calculations, GE used 25 percent of the live load for computing the seismic loads owing to an SSE and used the full value of the live load for the design of structural elements. The basis for using 25 percent of the live load for computing the SSE seismic loads is provided in SSAR Section 3.8.1. As a result of its review, the staff concludes that the reduction factor of 75 percent used by GE for calculating the contribution of the live load to the overall seismic loads is consistent with the guidelines of ASCE 7-88 Standard (formerly ANSI A58.1) and the common industry practice, and is, therefore, acceptable.

On the subject associated with the interaction of non-seismic Category I SSCs with the seismic Category I SSCs, SSAR Section 3.7.2.8 states that all non-seismic Category I SSCs will meet one of the following criteria:

(1) the collapse of any non-seismic Category I structure, system, or component will not cause the non-seismic Category I structure, system, or component to strike any seismic Category I structure, system, or component.

(2) the collapse of any non-seismic Category I structure, system, or component will not impair the integrity of seismic Category I SSCs. This may be demonstrated by showing that the impact loads on the seismic Category I structure or system or component resulting from collapse of an adjacent non-seismic Category I structure or system or component, because of its size and mass are either negligible or smaller than those considered in the design (e.g., loads associated with tornado, including tornado missiles).

(3) the non-seismic Category I SSCs will be analyzed and designed to prevent their failure under SSE conditions in such a manner that the margin of safety of these SSCs is equivalent to that of seismic Category I SSCs.

Criteria (1), (2), and (3) are acceptable to the staff.

Although these criteria are used for the design of the standard plant, during the construction phase, interferences from field run commodities and field modifications can lead to adverse interaction between seismic Category I and other non-seismic Category I SSCs. To identify and correct such potentially adverse interaction in accordance with these described criteria, SSAR Section 3.7.5.4, Amendment 33, stated that the COL applicant will describe the process for completion of the design of balance-of-plant and non-safety related systems and propose procedures for an inspection of the as-built plant to verify that the interaction of non-seismic Category I SSCs with seismic Category I SSCs does not cause failure of the seismic Category I SSCs to perform their intended safety function. This is acceptable. The staff will review the process and procedures as part of the COL application and the ITAAC for the COL stage.

To demonstrate that the as-built plant structures (primary containment structure, internal structures, RB, control building, radwaste building, and turbine building) are able to withstand the structural design basis loads as defined in SSAR Section 3.8, Amendment 33, GE, in SSAR Section 3H.5, Amendment 33, stated that when the construction is complete, a structural analysis report will be prepared to document the results of the review of construction records for material properties used in construction (i.e., in-process testing of concrete properties and procurement specifications for structural steel and reinforcing bars) and the inspection of as-built building dimensions. In this report, according to GE, construction deviations and design changes, if any, will be assessed to determine appropriate disposition. The as-built plant structures are considered acceptable "as-they-are," if the structural design meets the acceptance criteria and load combinations defined in SSAR Section 3.8, and the dynamic responses (i.e., FRS, shear forces, axial forces and moments) of the as-built plant structures are bounded by the responses documented in Appendices 3A, 3G, and 3H to the SSAR, Amendment 33. GE also stated that depending upon the extent of the deviations and design changes, compliance with the acceptance criteria can be determined by either:

a. Analyses of evaluations of construction deviations and design changes, or

b. The design basis analyses will be repeated using the as-built condition.

The staff considers that the reconciliation analysis procedures to be used by GE will ensure that the as-built plant structures are able to withstand the structural design basis loads and load combinations defined in SSAR Section 3.8, Amendment 33, and are, thus, acceptable.

The staff's evaluations of the seismic analyses and design of the ABWR seismic Category I buildings and other structures are discussed in the following paragraphs.

Reactor Building

GE originally developed the seismic response envelopes (shear forces, moments and FRS) for the RB considering 14 generic site conditions with soil profile depths ranging from 25.9 m (85 ft) to 91.4 m (300 ft) and averaged soil layer shear wave velocities varying from 303 m/sec (994 ft/sec) to 3048 m/sec (10,000 ft/sec). These site conditions represent a range of soft-soil site, medium-soil site, stiff-soil site, and hard-rock site. The 14 site conditions with various soil profiles and the associated shear wave velocity along the soil depth were listed in SSAR Table 3A.3-6, Amendment 16. A total of 42 cases of soil-structure interaction (SSI) analyses were performed in the original seismic analysis. In the seismic reanalysis, the site conditions (as shown in Table 3A-6 of SSAR, Amendment 33) used are basically the same as those used in the original analyses, except that Profile VP2 was eliminated, and Profiles VP6, HR, and EH were replaced by rock and hard-rock conditions that represent all site conditions with soil shear wave velocities above 1058 meters/second (3300 ft/sec). With the revised site conditions, a total of 22 SSI analysis cases were considered. The 14 site conditions and the 22 analysis cases were listed in Table 3A-7 of SSAR Appendix 3A, Amendment 33. The structural stick models, which represent the RB, RCCV, internal structures and reactor pressure vessel (RPV) were used in the reanalyses. In addition, one condition of the reinforced concrete structural model with the properties of cracked concrete was also considered. The Bechtel-version of SASSI computer code was used for the SSI analysis. The staff reviewed the validation documents and approved the use of this code. This code was also reviewed by the staff for the structure-to-structure interaction analysis. The staff found (1) the site conditions and the SSI analysis cases covered a wide range of soil properties and site geometry, (2) the modeling technique used for the structural stick model is consistent with the guidelines of SRP Section 3.7.2, and (3) the version of SASSI Computer Code previously accepted by the staff for other nuclear power plant licensing applications was used for the SSI analyses of the ABWR plant structures, and, therefore, the seismic design analysis of the RB is acceptable.

As discussed in SSAR Appendix 3A, Amendment 33, the SSI analysis cases were categorized into two groups. In the first group, 3-D SSI analyses of the RB and the control building were performed individually without considering of structure-to-structure interaction effects (Cases 1 through 16 in SSAR Table 3A-7). In the second group, 2-D SSI analyses of the RB and control building were performed considering individual buildings as well as multiple buildings, including the turbine building, to evaluate the structure-to-structure interaction effects (Cases 17 through 22 in SSAR Table 3A-7). For all these 22 SSI analyses, an embedment of 25.9 m (85 ft) from the finished grade to the bottom of the basement was used.

A 2-D structural model used in the original analyses combined with a 3-D soil-structure foundation model was used in the seismic reanalysis. The SSE damping ratios recommended in RG 1.61 were assigned to the structural elements in the reanalysis. In response to Outstanding Issue 4 of the DSER (SECY-91-153), GE increased the maximum tornado wind speed from 418 km/hr (260 mph) to 483 km/hr (300 mph) and adopted Missile Spectrum I per SRP Section 3.5.1.4 for the seismic Category I structure design. As a result, the upper portion of the RB model was modified because the building roof and the super structures were strengthened. From the design calculation audits, the staff concludes that GE's modeling techniques meet the guidelines of the SRP Section 3.7.2 and are, therefore, acceptable and that DSER Outstanding Issue 4 is resolved.

To obtain the input for the SASSI analyses, GE used computer code SHAKE to perform a computer analysis of the free-field soil column to obtain the shear modulus and material damping of soils compatible with the seismic strains induced in the free field for each site condition. In the SSI analysis, the structural model of the RB (consisting of the enclosure structure, RCCV, RPV pedestal, RPV, and internals) did not include the eccentricity of the structure. For verifying the symmetry of the RB and demonstrating the adequacy for not considering the building eccentricity in the analyses, GE compared the frequencies and modal participating factors of the fixed-base RB model both with and without the calculated eccentricity and found that the effect of eccentricity is negligible. GE also considered the effect of separation between the foundation soil and embedded wall on the structural response. During the design calculation audits conducted on October 12 through 15, 1992, and February 22 through 25, 1993, the staff found that GE compared the FRS obtained from a fixed base finite element model (unsymmetrical model) and a fixed base stick model (symmetrical model) and showed good agreement of two sets of horizontal FRS. However, in the frequency range between 20 Hz and 30 Hz, the vertical FRS at the building walls generated from the finite element model significantly exceeded the vertical FRS generated from the stick model. In SSAR Section 3.7.2.1.5.1.1 and 3A.10.2, Amendment 33, GE stated that to include these exceedances in the structural models responses, the results of the finite element analysis were used as an additional case to obtain the site enveloping results. From this review, the staff concludes that GE's modeling of the RB and supporting soil medium, use of the 2-D structural model, and SSI analysis methods comply with the guidelines of SRP Section 3.7.2 and the final seismic response envelopes, including the FRS envelopes, are reasonable and acceptable.

To determine the seismic loads for the structural design, the analysis results of the 16 three-dimensional analysis cases (Cases 1 through 16) based on the computer code SASSI were enveloped to account for the effects of the 14 site conditions and the uncertainties of the relevant parameters. The enveloping maximum shears and moments along the RB walls, RCCV shell, reactor shield wall (RSW), reactor pedestal and key internal structural elements for horizontal excitation were given in Tables 3A-19a through 3A-19d of SSAR Appendix 3A, Amendment 33. Because this building is nearly symmetrical about the two horizontal axes, the torsional moments are obtained using the enveloping shear force at each floor multiplied by an accidental eccentricity equal to 5 percent of the respective maximum floor dimension. These forces, moments and torsional moments were used for the design of various structural elements. For the design of the structural elements below the ground surface, such as exterior walls and RCCV shell (which is structurally tied with the exterior walls), GE did not consider the reduction of the horizontal shear forces at the structural elements below the ground surface as calculated from the analyses but did consider the largest shear force above the ground surface for the design. In the vertical direction, GE expressed the loads in terms of the enveloping absolute acceleration to simplify the analyses and design of floor slabs and components. The staff reviewed the final reanalysis results (accelerations, displacements, forces and moments) as provided in SSAR Appendix 3A, Amendment 33. GE has met the guidelines of SRP Section 3.7.2 for developing the seismic design load envelopes for the RB structure and the design loads calculated are acceptable.

In the DFSER, the staff stated that the SSAR did not completely describe the procedure for calculating the FRS and for developing the revised FRS envelopes. This was DFSER Confirmatory Item 3.7.2-1. During the design calculation audits, GE agreed to provide the basis for applying the uncertainty factors to the FRS (horizontal and vertical). This was DFSER Confirmatory Item 3.7.2-2. Also, GE agreed to include the seismic structural displacement profiles, which are needed for the seismic design of piping systems, in the SSAR. This was DFSER Confirmatory Item 3.7.2-3. These three confirmatory items are resolved.

From the audit of the design calculations and reanalysis results, and the review of Appendix 3A to the SSAR, Amendment 33, the staff found that the structural response envelopes (member forces, bending moments, and FRS) are dominated by the responses obtained from the RB SSI analysis with hard-rock site condition (Case RZU). In other words, the structural responses of the RB with a hard-rock foundation envelop most of the responses calculated from the SSI analyses for other site conditions. The ABWR standard plant structures are designed using the envelopes of seismic forces and moments. The use of the structural response envelopes for the design of the plant structural elements, systems, and components might yield unnecessarily high loads for a plant founded on a soil site. As an acceptable alternative, GE or the COL applicant may group the 14 generic site conditions into different generic categories of site condition (such as soft-soil site, medium-soil site, soft-rock site, hard-rock site, etc.) and develop the FRS envelopes for each generic category. The resulting FRS envelope for the particular category of site condition most representative of a specific site may be used for the subsystem design for this site. The staff will review the development of the category-based FRS envelope and the site-specific design of subsystems on a case-by-case basis.

Originally, GE did not consider the flexibility effect of the drywell equipment and piping support structure (DEPSS) when generating the FRS for the seismic input to the design of subsystems supported by the DEPSS. Because of the exclusion of the DEPSS' flexibility effect, which might cause additional amplification of the FRS, the staff believed that such subsystems supported on the DEPSS as piping and equipment could be underdesigned based on the existing FRS. This was DFSER Open Item 3.7.2-2. SSAR Section 3.8.3.4.4, Amendment 33, described the procedure for checking the applicability of the deformation criteria of frame-type pipe supports given in SSAR Section 3.7.3.3.4, from which one can determine whether the DEPSS can be considered rigid or not. The staff's review and acceptance of the deformation criteria of frame-type pipe supports are discussed in Section 3.12.6.7 of this report. If these criteria can not be met, SSAR Section 3.7.3.3.4 further stated that the COL applicant will generate the FRS at piping attachment points by considering the DEPSS as part of the structure and using the dynamic analysis methods described in SRP Section 3.7.2 or by analyzing the piping systems treating the DEPSS as a part of pipe support. This is acceptable, and DFSER Open Item 3.7.2-2 is resolved.

The staff concludes that the SSI analyses performed for all site conditions and analysis cases, the development of the seismic load envelopes for the structural design of the ABWR RB, and the generation of the FRS envelopes for the design of the subsystems located in the RB are acceptable.

Control Building

On the basis of the seismic design audits, the staff found that the modeling technique and analysis method used for the control building are essentially the same as those used for the RB. Therefore, the staff considers the results obtained adequate and acceptable. The details of this staff review are discussed in the following paragraphs.

In the original SSI analysis, GE did not consider the effect of structure-to-structure interaction between the control building and adjacent buildings such as the RB and turbine building. The energy feedback from the adjacent buildings during an earthquake could significantly affect the seismic response of the control building because these adjacent buildings are much heavier. GE should consider the effect of structure-to-structure interaction in the SSI analysis of the control building. This was DFSER Open Item 3.7.2-3.

GE used a 2-D SSI model in the original SSI analysis. As shown in SSAR Amendment 23, GE's parametric studies for the RB indicated that the 2-D SSI analysis typically underestimated both the horizontal and vertical spectral peak accelerations at higher elevations of the building for medium-stiff-soil sites and hard-rock sites. During the design calculation audits, the staff was concerned about the significance of the difference between 2-D and 3-D SSI analyses of the control building. This was DFSER Open Item 3.7.2-4.

GE originally considered three generic site conditions to generate the envelopes of structural seismic loads and FRS. GE also applied uncertainty factors of 1.5 and 1.0, respectively, to the horizontal and vertical FRS envelopes of the control building. During the design calculation audits, GE said that a part of the uncertainty factor of 1.5 for the horizontal FRS envelope was to account for the uncertainty resulting from using only three site conditions in the standard design. The staff was concerned with the basis of the uncertainty factors and with the sufficiency of considering only three site conditions in the standard design. GE agreed to provide the basis for the uncertainty factors. This was DFSER Confirmatory Item 3.7.2-4. Also, the original SSAR did not describe the procedure to generate FRS envelopes. GE committed to document this procedure in a future amendment of the SSAR. This was DFSER Confirmatory Item 3.7.2-5.

In response to these DFSER open and confirmatory items, GE, following the same procedures that were applied to the RB, conducted a seismic reanalysis of the control building. In this seismic reanalysis, the site conditions and the SSI analysis cases used are the same as those used in the RB seismic reanalysis. The 2-D structural dynamic model shown in Figures 3A-27 through 3A-29 of SSAR Appendix A, Amendment 33, was combined with a 3-D soil-structure foundation model for the seismic reanalysis. The site conditions and the soil profiles for the 22 SSI analysis cases are documented in Table 3A-7 of SSAR Appendix 3A, Amendment 33. For all these 22 SSI analyses, an embedment of 23.2 m (76 ft) was included. In addition, the same approach, analysis procedures and ground motion time history that were applied to the RB were adopted for the control building analyses, and the SSI analysis computer code SASSI was also used in this analysis. The staff concludes that the consideration of the soil site conditions and SSI analysis cases, the modeling technique used for structure and soil foundation, and the ground motion time history used in the control building seismic reanalysis are acceptable, and DFSER Open Item 3.7.2-4 is resolved.

To generate the FRS envelopes, GE (1) followed the same procedures that were applied to the RB, (2) generated 2-, 3-, 5-, and 10-percent damping FRS for all SASSI analysis cases shown in SSAR Table 3A-7, (3) developed the envelopes of the FRS at all required locations in each of the three directions, (4) developed the envelope FRS in the two horizontal directions at each location to form the bounding horizontal FRS, and (5) broadened the peaks of the FRS by ± 15 percent. The uncertainty factors of 1.5 and 1.0, respectively, to the horizontal and vertical FRS envelopes used in the original seismic analyses were not applied in this reanalysis. The staff reviewed the process for developing the FRS envelopes and the resulting FRS envelope plots provided in SSAR Appendix 3A, Amendment 33, and found them acceptable, and both DFSER Confirmatory Items 3.7.2-4 and 3.7.2-5 are resolved.

In response to DFSER Open Item 3.7.2-3, GE used the 2-D soil-foundation model combined with the 2-D control building, RB, and turbine building models, and the SASSI computer code to evaluate the significance of the effects of structure-to-structure interaction (Cases 20 through 22 in SSAR Table 3A-7). The analysis results, as documented in Tables 3A-15 through 3A-18 and Figures 3A-122 through 3A-127 of SSAR Appendix 3A, Amendment 33, showed that the effect of structure-to-structure interaction is significant for the control building. However, these results are bounded by the structural response envelopes, which is acceptable, and DFSER Open Item 3.7.2-3 is resolved.

From the review of Appendices 3A and 3G of SSAR Chapter 3 and the certified design material (CDM) for the GE ABWR design dated March 1992, the staff observed that the building dimensions are inconsistently specified in these documents. For example, the dimensions of the control building are specified to be 16 m x 45 m (52 ft x 147 ft) in plan and 12.2 m (40 ft) in embedment depth according to SSAR Section 3A.2, 22 m x 56 m (72 ft x 184 ft) in plan and 25.9 m (85 ft) in embedment depth according to SSAR Section 3G.3.2, and 24 m x 56 m (79 ft x 184 ft) in plan and 23.1 m (75 ft and 9 in.) in embedment depth according to the CDM. In the DFSER, the staff noted that GE should verify the accuracy of all dimensions of the control building, including the embedment depth, used in the final seismic analysis of the seismic Category I structures shown in the SSAR and the CDM. This concern also applied to the dimension of all other seismic Category I building structures, including the RB. This was DFSER Open Item 3.7.2-5.

In SSAR Sections 3.7 and 3.8, Amendment 33, GE corrected all inconsistent building dimensions, including building embedment depths, for the seismic Category I structures. This is acceptable, and DFSER Open Item 3.7.2-5 is resolved.

Radwaste Building Substructure

During an earlier design calculation audit, GE indicated that the radwaste building does not house any safety-related equipment and components and hence, there is no need to generate FRS for the subsystems. To ensure that the building maintains its structural integrity during and after an SSE and to prevent unacceptable leakage of the radwaste material outside the embedded portion of the building, GE elected to analyze the radwaste building by the response spectrum analysis method and to design the radwaste building structure for the SSE seismic loads. When the modal response spectrum method was used for the analysis of the radwaste building structure and the subsystems, GE combined the modal responses by the method delineated in RG 1.92, "Combining Modal Responses and Spatial Components in Seismic Response Analysis," Revision 1. The effects of high frequency modes were considered in accordance with the guidelines of Appendix A to SRP Section 3.7.2. The co-directional responses of the radwaste building structure to the three components of the earthquake ground motion were combined using the square-root-of-sum-of-squares (SRSS) rule according to the guidelines of RG 1.92. The methods used for the combination of the modal responses and the co-directional responses to the three components of the ground motion comply with the guidelines of SRP Section 3.7.2 and are, thus, acceptable.

The seismic analysis was performed using a fixed-base standing stick model to represent the structure. This simplified analysis model excludes the effects of both the structural embedment and site soil conditions. The resulting fundamental horizontal frequency is within the frequency range of the maximum amplification of the input ground response spectrum. This ensures that the resulting seismic loads for the design of the structure are sufficiently conservative to preclude the need for considering the effects of structural embedment and site soil conditions. In the DFSER, the staff reported that GE had not completed the implementation of the QA program for the seismic analysis of this building. In addition, the SSAR did not include the analysis results such as the structural frequencies, seismic shear forces, and seismic moments. This was DFSER Confirmatory Item 3.7.2-6.

As a result of discussions during the audit on February 22 through 26, 1993, in SSAR Section 3H.3, Amendment 33, GE provided the analysis methods and results of the radwaste building. As a result of its review, the staff concludes that they are acceptable. Also in a letter dated September 15, 1993, GE certified that the implementation of the QA program for the design calculations of the radwaste building had been completed. Therefore, DFSER Confirmatory Item 3.7.2-6 is resolved.

Turbine Building

In the DFSER, the staff noted that GE had committed to perform a dynamic analysis for the portion of the MSL inside the turbine building, but neither the FRS for use as the seismic input for the MSL analysis nor the procedure to generate the FRS had been provided in the SSAR. On the basis of SRP Sections 3.7.2 and 3.7.3, the staff requested that GE perform a dynamic analysis of the turbine building and condenser to generate a set of FRS as the seismic input for the MSL analysis. The staff also noted that, according to GE, the FRS generated at the containment shell are used as the input motion at the end of MSL anchored to the containment, and the ground motion response spectrum multiplied by an amplification factor serves as the input at the end of MSL (including branch lines) anchored to the turbine building and condenser. Use of the containment FRS as the input at the containment side for the MSL analysis is acceptable.

However, the staff noted in the DFSER about the adequacy of using some multiple of the ground response spectrum as the input for the MSL analysis at the turbine building side anchor. This was DFSER Open Item 3.7.2-6. In SSAR Section 3.2.5.3, Amendment 33, GE stated that the dynamic input loads for the design of the MSLs in the turbine building are derived as follows: (1) for locations on the basemat, the amplified response spectra (ARS) (FRS in this report) shall be based upon the RG 1.60 response spectra normalized to 0.6g (i.e., 2 times the SSE ground response spectra) and (2) for locations at the operating deck level (either operating deck or turbine deck), the ARS used shall be the same as those used at the RB end of the MST. SAMs shall be similarly calculated. The staff concludes that the dynamic input loads for the design of the MSLs inside the turbine building are acceptable because (1) a comparison of the response spectra at the RB foundation level with the RG 1.60 response spectra anchored to 0.6*g* ZPA shows that the RG 1.60 response spectra anchored to the same 0.6*g* ZPA envelop the response spectra at the RB foundation level and (2) the turbine operating deck is located at approximately the same elevation as the anchor point of the main steamline at the RB side and the response spectra at the RB end were generated using an acceptable analysis approach as discussed. DFSER Open Item 3.7.2-6 is resolved.

In the markup of SSAR Section 3.7.3.16, dated May 21, 1992, GE described the seismic design of the turbine building using the UBC approach for seismic Zone 2A. However, the seismic design, based on UBC Zone 2A rules, does not necessarily ensure that the turbine building will be structurally capable of withstanding the SSE for the standard plant design to protect the safety function of the portion of the MSL inside the turbine building. This was DFSER Open Item 3.7.2-7. In SSAR Section 3.7.3.16, Amendment 33, GE stated that for the design of non-seismic Category I structures which are required to withstand an SSE without losing the structural integrity, the procedures described in the UBC seismic design criteria shall be followed with the following limitations:

(1) The seismic zone shall be "Zone 3."

(2) For dual systems (i.e., shear wall with braced steel frame), one of the two systems must be designed to be capable of carrying all of the seismic loading without collapse. No credit will be given for the other system for resisting lateral loads.

The seismic zone factor Z for UBC Zone 3 is 0.3, which is equivalent to a peak ground acceleration (ZPA) of 0.3*g*. Therefore, the staff concludes that GE's use of the UBC Zone 3 requirements with these restrictions as discussed for the seismic design of non-seismic Category I structures, including the turbine building, can ensure that the turbine building will retain its structural integrity under the specified SSE and is, thus, acceptable. DFSER Open Item 3.7.2-7 is resolved.

Confirmation of Plant-Specific Seismic Design Adequacy

To confirm the site-specific seismic design adequacy of the standard plant, SSAR Section 2.3.1, Amendment 9, stated that the COL applicant shall demonstrate, according to the procedure specified therein, that it has satisfied the eight site-dependent conditions specified in SSAR Section 3A.1. These eight site-dependent conditions were:

(1) The peak ground acceleration is less than 0.30*g* SSE.

(2) The site design response spectra are less than or equal to those given in RG 1.60 normalized to the peak ground accelerations in Condition 1.

(3) There is no potential for liquefaction at the plant site as a result of an SSE as reviewed and concurred with by the NRC staff (the liquefaction potential of the foundation and site soils will be investigated and reported for a long duration, New Madrid-type earthquake).

(4) There is no potential for fault movement at the plant site as reviewed and concurred with by the NRC staff.

(5) The embedment depth of the reactor building is 25.9 m (85 ft). The excavation tolerance is ±15 cm (± 0.5 ft).

(6) The average shear wave velocity for the top 9 m (30 ft) of soil is 305 m/sec (1000 ft/sec) minimum. The upper bound shear wave velocity is 3048 m/sec (10000 ft/sec).

(7) For layered soil sites with parameters that have very abrupt variations with depth, an analysis with site-unique properties will be performed to confirm the applicability of the generic analysis.

(8) The soil-bearing capacity at the site is adequate to accommodate plant design loads.

In a letter dated August 19, 1991, GE described an evaluation procedure for the site-specific confirmation of the seismic design adequacy of the ABWR standard plant and, in SSAR Section 2.3.1.2, Amendment 18, GE revised the site-dependent conditions and the standard design adequacy confirmation procedure. The staff reviewed these submittals and identified four issues in the DFSER:

(1) GE should provide the criteria and the confirmation procedure for the site condition classified as shallow soil site in the SSAR. This was DFSER Open Item 3.7.2-8.

(2) When a site-specific SSI analysis is performed for the SSE case, the three components of the ground motion time history should satisfy not only the PSDF enveloping criterion, but also the response

spectrum enveloping criterion for all damping values to be used with response analysis. This was DFSER Confirmatory Item 3.7.2-7.

(3) In its letter dated August 19, 1991, and in SSAR Section 2.3.1, Amendment 18, GE stated that COL applicants shall consider site-dependent bounding Conditions 6 and 7 above as two individual evaluation parameters when confirming the adequacy of the standard plant design for a specific site. The effect of soil layer depth was not considered or included in the evaluation. The staff was concerned that to compare Conditions 6 and 7 with the site-specific design parameters separately is not sufficient to confirm the design adequacy of the standard plant. The staff believed that these two conditions should be considered together with the depth of soil layers. In addition, the site-specific responses (structural member forces and FRS) should be compared to the response envelopes used for the standard plant design unless the site-specific parameters (shear wave velocity, number of soil layers, and depth of soil layers) can be demonstrated to be comparable to one of the 14 generic site conditions. This was DFSER Open Item 3.7.2-9.

(4) In SSAR Section 2.3.1.2, Amendment 18, GE stated that the FRS comparison can be made for one damping value only when confirming the seismic design adequacy of piping and equipment. According to the guidelines of SRP Section 3.7.1.I.1.b, the FRS comparison should be performed for all damping values assigned to different piping systems and equipment. This was DFSER Confirmatory Item 3.7.2-8.

To respond to these four items, in SSAR Amendment 30, GE revised the eight site-dependent conditions by (1) moving these conditions from Appendix 3A to Section 2.3, "COL License Information," (2) eliminating Conditions 6 and 7, and (3) adding the condition for the shallow soil site. GE also revised the confirmation procedure by only comparing the site-specific SSE ground response spectrum of 5-percent damping at plant grade in the free-field with the design ground response spectrum (i.e., RG 1.60 response spectrum anchored to 0.3*g* PGA). The staff reviewed these revisions and determined that it is acceptable to compare the ground response spectra at plant grade in the free-field for the confirmation of the design adequacy and that there is no need to specifically confirm the adequacy of the ground motion time history, local soil layering effects, and FRS for the subsystem analysis and design. The basis for the staff conclusion is that when the site-specific ground motion response spectrum is developed and the free-field surface ground motion is calculated, the local geotechnical properties (soil layering, shear module, etc.) are all considered. As far as the adequacy of site-specific ground motion time history and the generation of the FRS, the staff will review it on a case-by-case basis. Therefore, DFSER Open Items 3.7.2-8 and 3.7.2-9 and DFSER Confirmatory Items 3.7.2-7 and 3.7.2-8 are resolved.

The staff concludes that the ABWR standard plant design is acceptable and meets the requirements of GDC 2 and Appendix A to 10 CFR Part 100. This conclusion is based on the following: the applicant has met the requirements of GDC 2 and Appendix A to 10 CFR Part 100 with respect to the capability of the structures to withstand the effects of the earthquakes so that their design reflects:

1. Appropriate consideration for the most severe earthquake recorded for the most sites east of the Rocky Mountains with an appropriate margin (GDC 2);

2. Appropriate combination of the effects of normal and accident conditions with the effect of the natural phenomena; and

3. The importance of the safety functions to be performed (GDC 2). The use of a suitable dynamic analysis to demonstrate that SSCs can withstand the seismic and other concurrent loads.

The applicant has met the requirements of Item 1 by using seismic design parameters that meet the guidelines of SRP Section 3.7.1. The combinations of earthquake-induced loads with those resulting from normal and accident conditions in the design of seismic Category I structures meet the guidelines of SRP Sections 3.8.1 through 3.8.5 and are in conformance with Item 2.

The scope of review of the seismic system and subsystem analysis for the plant included the seismic analysis methods for all seismic Category I SSCs. It included review of procedures for modeling, seismic soil-structure interaction, development of envelope response spectra, inclusion of torsional effects, evaluation of seismic Category I structure overturning and sliding, and determination of composite damping. The review also included design criteria and procedures for evaluation of the interaction of non-seismic Category I structures with Category I structures, and the effects of parameter variations on FRS. In addition, the review included criteria and seismic analysis procedures for seismic Category I buried piping outside containment.

GE performed the system and subsystem analyses on an elastic and linear basis. Time history methods form the bases for the analyses of all major seismic Category I structures except the radwaste building. The seismic analysis of the radwaste building and the seismic analysis of seismic Category I systems and components are based on the response spectrum analysis method. When the modal response spectrum method was used, modal responses were combined in conformance with RG 1.92. GE's consideration of the high-frequency mode contribution to the overall structural responses met the guidelines of Appendix A to SRP Section 3.7.2. The SRSS of the maximum codirectional responses was used in accounting for three components of the earthquake motion for both the time history and response spectrum methods. FRS inputs to be used for analysis and design of structural elements, systems, and components were generated from the time history method, and they are in conformance with RG 1.122, "Development of Floor Design Response Spectra for Seismic Design of Floor-Supported Equipment or Components," Revision 1. A vertical seismic system dynamic analysis was employed for all SSCs where analyses had shown significant structural amplification in the vertical direction. Torsional effects and stability against overturning and sliding were considered.

A coupled structure and soil model was used to evaluate soil-structure interaction effects upon seismic responses. Appropriate nonlinear stress-strain and damping relationships for the soil were considered in the analysis. The staff concludes that the use of seismic structural analysis procedures and criteria delineated by GE provide an adequate basis for the seismic design, which is in conformance with the requirements of Item 3 listed.

3.7.3 Seismic Subsystem Analysis

The staff review of the seismic subsystem analyses included the seismic analysis methods for ABWR seismic Category I SSCs that were not explicitly included in the structural models when seismic analyses of the seismic Category I structures were performed. Such items include all seismic Category I cable trays and supports, conduit and supports, above-ground tanks, buried piping and tunnels, and structural elements that support other seismic Category I items, the dynamic effects of which could affect the seismic response of the supported items (e.g., the steel platforms, the radial steel beams, and the DEPSS in the RB). Seismic qualification of seismic Category I mechanical equipment and instrumentation and electrical equipment is evaluated, respectively, in Sections 3.9 and 3.10 of this report. The seismic analysis of piping systems is discussed in Section 3.12 of this report.

GE performed the subsystem analysis on a linear elastic basis. The modal response spectrum method and equivalent static load method formed the basis for the analyses of all major seismic Category I subsystems in both the horizontal and vertical directions. When the modal response spectrum method was used, the analysis model was established and the procedures used for its development complied with the guidelines of SRP Section 3.7.3 for dynamic analyses. The modal responses and the spatial components of responses were combined according to the methods delineated in RG 1.92. These methods used for combining seismic responses comply with the guidelines of SRP Section 3.7.2, and are, thus, acceptable. GE used the FRS envelopes obtained from the time history analyses of the structures for the seismic input for the subsystem analyses. GE considered torsional effects of eccentric masses. These modeling techniques and analysis methods for the subsystems meet the guidelines of SRP Section 3.7.3 and, therefore, are acceptable.

Seismic Category I cable trays and conduit supports were analyzed using the modal response spectrum method of analysis. The analysis procedure and design criteria were described in more detail in SSAR Section 3.8.4, Amendment 33. The staff's evaluation of the subject is given in Section 3.10.2 of this report.

Buried seismic Category I piping systems and tunnels were analyzed using the techniques that account for the effects of seismic wave travel, differential movements of pipe anchors, bent geometry and curvature changes, local soil settlements or soil arching. The early SSAR amendments did not describe in detail the procedure for the analysis of buried piping and tunnels or provide any description of the procedure for the dynamic analysis and evaluation of the above-ground tanks. This was DFSER Open Item 3.7.3-1. In SSAR Section 3.7.3.12, Amendment 33, GE provided the design procedures for buried piping and pipe tunnels, and in SSAR Section 3.7.3.17, Amendment 33, GE provided the analysis procedures for the above-ground tanks. The procedures for the analysis and design of buried piping, pipe tunnels, and the above-ground tanks conform with the guidelines of SRP Section 3.7.3, and are, therefore, acceptable. DFSER Open Item 3.7.3-1 is resolved.

The staff concludes that the design of the subsystems of the ABWR standard plant is acceptable and meets GDC 2 and Appendix A to 10 CFR Part 100 with regard to the capability to withstand the effects of the earthquakes. Evaluation findings for SSAR Section 3.7.3 have been combined with those for SSAR Section 3.7.2 and are given in Section 3.7.2 of this report.

3.7.4 Seismic Instrumentation

The seismic instrumentation system specified for the ABWR plant in SSAR Section 3.7.4 is acceptable and meets GDC 2 and 10 CFR Part 100 Appendix A. The applicant met these requirements by requiring installation of instrumentation that is capable of adequately measuring the effects of an earthquake.

GE meets the requirements of 10 CFR Part 100, Appendix A, and GDC 2 by providing instrumentation that is capable of measuring the effects of an earthquake. The installation of the specified seismic instrumentation as specified in SSAR Section 3.7.4.1 in the reactor containment structure, other ABWR Category I structures, and the free field constitutes an acceptable program for recording data on seismic ground motion as well as data on the amplitude, frequency, and phase relationship of the seismic response of major structures and systems in the event of an earthquake. A readout of pertinent data from the various seismic instruments will yield sufficient information to guide the operator on a timely basis to determine if the level of earthquake motion ground requiring plant shutdown has been exceeded. Data obtained from such installed seismic instrumentation will be sufficient to ascertain that the seismic analysis assumptions and the analytical models used in the seismic design of the ABWR were adequate and that allowable stresses have not been exceeded under conditions when continuity of operation is intended. The staff finds the design for seismic instrumentation acceptable.

In view of the time and effort required to determine if the level of earthquake ground motion requiring plant shutdown has been exceeded, COL applicants should establish plant operating procedures that define specifically what constitutes a significant exceedance of the level of earthquake ground motion requiring plant shutdown. This was DFSER COL Action Item 3.7.4-1. GE has included this information in Section 3.7.4.4 of the SSAR. This is acceptable.

Because of the continuous enhancement in the state of the art of seismic instrumentation and the revisions to Appendix A of 10 CFR Part 100 and to RG 1.12 (Rev. 1) (currently in progress), conformity with instrumentation guidelines in existence at the time of an individual license application will be required. This was DFSER COL Action Item 3.7.4-2. GE has also included this information in Sections 3.7.4.1 and 3.7.4.2 of the SSAR. This is acceptable.

3.8 Design of Seismic Category I Structures

3.8.1 Concrete Containment

The containment is designed as a reinforced-concrete cylindrical shell structure with an internal steel liner made of carbon steel, except for wetted surfaces where stainless steel or carbon steel with stainless steel cladding will be used. It is divided by the diaphragm floor and the reactor pedestal into an upper drywell chamber, a lower drywell chamber and a suppression chamber. The containment is surrounded by and structurally integral with the RB through the RB floor slabs and the spent fuel pool structures. The containment wall is 2.0 m (6 ft 7 in.) thick with an inside radius of 14.5 m (47 ft 7 in.) and height of 29.5 m (96 ft 9 in.). The containment design pressure is 310.3 kPa (45 psig). The containment is designed to resist various combinations of dead loads; live loads; environmental loads, including those resulting from wind, tornados, and earthquakes; normal operating loads; and loads generated by a postulated LOCA. The design, fabrication, construction, and testing of containment are in accordance with Subsection CC of ASME Code, Section III, Division 2. In the DFSER, the staff stated a concern about the use of ASME Code edition. This was DFSER Open Item 3.8.1-1. In SSAR Section 1.8, Amendment 33, GE stated that the 1989 edition of ASME Code, Section III, will be used for the containment design. The staff has reviewed the adequacy of the 1989 Edition of ASME Code, Section III, Division 2, for reinforced concrete containment design and found it acceptable with the following clarification of the criteria for the tangential shear design. With regards to the edition of the ASME code used for the ABWR containment design, Open Item 3.8.1-1 is considered closed. According to RG 1.136 (Rev. 2), the staff has not yet endorsed Subsection CC-3000 of ASME Code, Section III, Division 2, "Code for Concrete Reactor Vessels and Containments." Specifically, the staff raised a concern that the code specified allowable tangential shear stress of $0.2f'c$ is too high for the design of reinforced concrete containments. Subsequently, SSAR Section 3.8.1.5 revised the allowable tangential shear from $0.2f'c$ to $2.4\sqrt{f'c}$ (3.92 MPa (569.2 psi) for the conrete strength $f'c$=27.6 MPa (4000 psi) to be used for containment vessel and foundation mat). The results of GE's analysis show that the maximum tangential shear stress is 3.60 MPa (522.3 psi) and the total calculated shear strain is 0.000295, under the factored load combinations. In addition, GE provided the staff its backup calculations on December 3, 1993, for review. As a result of its review, the staff found that the revised allowable tangential shear stress of $2.4\sqrt{f'c}$ is within the limit of in-plane shear strength $2.65\sqrt{f'c}$ for shear walls specified in ACI 318 Code and the calculated total shear strain of 0.000295 is smaller than the limit of tangential shear strain based on the tests performed for the reinforced concrete cylindrical shell. Based on this discussion, the staff concludes that the proposed allowable tangential shear stress is acceptable, and DFSER Open Item 3.8.1-1 is resolved. To follow the 10 CFR Part 52 design certification process, any change to the use of the ASME Code (1989 Edition) for the design and construction of reinforced concrete containment structural elements (including the tangential shear stress limit discussed above), would involve an unreviewed safety question and, therefore, require NRC review and approval prior to implementation. Furthermore, any requested change to the use of this code must either be specifically described in the COL application or submitted for license amendment after COL issuance.

The major steel components of the concrete containment, as stated in SSAR Section 3.8.2, consist of personnel air locks, equipment hatches, penetrations, and the drywell head. These components are designed for the same loads and load combinations as those used in the design of the concrete containment shell to which these components will be attached. These components will be fabricated, and tested as Class MC components in accordance with the 1989 Edition of Subsection NE of ASME Code, Section III, Division 1. Also in SSAR Section 3.8.1.7.1, GE stated that the COL applicant will perform the structural integrity test (SIT) of the ABWR containments in accordance with the provisions of Article CC-6000 and Subarticle CC-6230 of the ASME Code, Section III, Division 2 (1989 Edition). Similar to that for concrete containment, the staff raised, in the DFSER, a concern about the edition of ASME Code for the design of major steel components. This was DFSER Open Item 3.8.1-2. In Amendment 33 of the SSAR, GE stated that the 1989 Edition of ASME Code, Section III, will be used for the design and test. This edition of ASME Code is referenced in 10 CFR 50.55a and is, therefore, acceptable. DFSER Open Item 3.8.1-2 is resolved.

In SSAR Section 3.8.2.4.1.4, Amendment 33, GE stated that the drywell head, which consists of shell, finger pin closure, and drywell-head anchor system, was analyzed using a finite-element stress analysis computer program. The stresses, including discontinuity stresses induced by the combination of external pressure or internal pressure, dead load, live load, thermal effects and seismic loads, were evaluated. GE's analyses and limits for the resulting stress intensities are in accordance with Subarticles NE-3130 and NE-3200 of ASME Code Section III, Division 1 (1989 Edition). The staff found that the analysis and design approach and the use of design code are consistent with the guidelines of SRP Section 3.8.1. This is acceptable. GE also stated in the SSAR that the compressive stress within the knuckle region caused by the internal pressure and the compression in other regions caused by other loads are limited to the allowable buckling stress values in accordance with Subarticle NE-3222 of ASME Code Section III, Division 1 (1989 Edition). GE's evaluation of the potential buckling of the drywell head and the buckling criteria used are consistent with the staff position for shell buckling described in Appendix E of this report. This is acceptable. To follow the 10 CFR Part 52 design certification process, change to the use of ASME Code (1989 Edition) for the design, fabrication, and construction of drywell head against buckling would involve an unreviewed safety question and, therefore, require NRC review and approval prior to implementation. Furthermore, any requested change to the use of this code must either be specifically described in the COL application or submitted for license amendment after COL issuance.

As for the ultimate capacity of the concrete containment, including the steel drywell head, the staff evaluation is discussed in Section 19 of this report.

During the first two design calculation audits, the detailed design calculations for the containment shell were not available for review. This was DFSER Confirmatory Item 3.8.1-1. During the audit dated February 22 through 25, 1993, GE provided its final containment shell design calculations for staff review. The staff found the design calculations acceptable. Therefore, Confirmatory Item 3.8.1-1 is resolved.

GE considers the first ABWR containment as a prototype and requires its SIT to be performed as such. Therefore, the COL applicant is required to provide the details of the test and the instrumentation as required by the ASME Code, Section III, Division 2, Subsection 6212, for such a test to the staff for review and approval. In the DFSER, the COL applicant's action to provide the details of the test and the instrumentation as required for such a test was DFSER COL Action Item 3.8.1-1. In SSAR Section 3.8.1.7.1, Amendment 33, GE stated that the COL applicant shall perform the SIT according to Article CC-6600 of ASME Code, Section III, Division 2 (1989 Edition) and RG 1.136, "Materials, Construction, and Testing of Concrete Containments," Revision 2, after completing the containment construction. This is acceptable.

The staff review of the performance of the containment structure under the loads and load combinations beyond the design limits is discussed in Section 19 of this report.

The staff concludes that the design of the concrete containment is acceptable and meets the relevant requirements of 10 CFR Part 50, 50.55a, and GDC 1, 2, 4, 16, and 50. This conclusion is based on the following:

1. GE has met the requirements of Section 50.55a and GDC 1 with respect to assuring that the concrete containment is designed, fabricated, erected, contracted, tested, and inspected to quality standards commensurate with its safety function to be performed

by meeting the guidelines of regulatory guides and industry standards indicated below.

2. GE has met the requirements of GDC 2 by designing the concrete containment to withstand a 0.3*g* SSE with sufficient margin, and the combinations of the effects of normal and accident conditions with the effects of environmental loadings such as earthquakes and other natural phenomena.

3. GE has met the requirements of GDC 4 by assuring that the design of the concrete containment is capable of withstanding the dynamic effects associated with missiles, pipe whipping, and discharging fluids.

4. GE has met the requirements of GDC 16 by designing the concrete containment so that it is an essentially leaktight barrier to prevent the uncontrolled release of radioactive effluent to the environment.

5. GE has met the requirements of GDC 50 by designing the concrete containment to accommodate, with sufficient margin, the design leakage rate, calculated pressure and temperature conditions resulting from accident conditions, and by assuring that the design conditions are not exceeded during the full course of the accident condition. In meeting these design requirements, GE has used the recommendations of regulatory guides and industry standards indicated below. GE has also performed an appropriate analysis that demonstrates that the ultimate capacity of the containment will not be exceeded and establishes acceptable margin of safety for the design.

The criteria used in the analysis, design, and those proposed for construction of the concrete containment structure to account for anticipated loadings and postulated conditions that may be imposed upon the structure during its service lifetime are in conformance with established criteria, and with codes, standards, guides, and specifications acceptable to the regulatory staff. These include meeting the positions of RG 1.94, "Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants," Revision 1, and RG 1.136, "Materials, Construction, and Testing of Concrete Containments," Revision 2, and industry standard ASME Boiler and Pressure Vessel Code, Section III, Division 2 (1989 Edition).

The use of these criteria as defined by applicable codes, standards, guides, and specifications; the loads and loading combinations; the design and analysis procedures; the structural acceptance criteria; the materials, quality control programs, and special construction techniques; and the testing and inservice surveillance requirements, provide reasonable assurance that, in the event of winds, tornados, earthquakes and various postulated accidents occurring within and outside the containment, the structure will withstand the specified design conditions without impairment of structural integrity or safety function of limiting the release of radioactive material.

3.8.2 Steel Components of the Reinforced Concrete Containment

The staff's evaluation of the design adequacy of the major steel components of the reinforced concrete containment is discussed in Section 3.8.1 of this report.

3.8.3 Concrete and Steel Internal Structures of Steel or Concrete Containment

In the ABWR design, the internal structures inside the containment include the reinforced-concrete diaphragm floor, the reactor pedestal, the reactor shield wall, and other structural components. The diaphragm floor separates the upper drywell from the suppression pool. The reactor pedestal consists of a ledge on a cylindrical shell that forms the reactor cavity, extending from the bottom of the diaphragm to the top of the containment foundation slab. The space enclosed by the cylindrical shell under the reactor is the lower drywell, which is connected to the suppression pool through a series of vertical and horizontal vents in the shell wall. A steel equipment platform is located in the lower drywell and is accessible through a steel personnel tunnel and a steel equipment tunnel from outside the containment. Other internal structures include the DEPSS and the miscellaneous floors. The major code used in the design of concrete internal structures is American Concrete Institute (ACI) Standard 349 (1980 Edition). The use of this code for the design of the seismic Category I reinforced concrete structures is acceptable, except that the staff position on the design requirements for the steel embedments should be satisfied. The staff position on steel embedment design is described in Appendix F of this report. As described in early SSAR Amendments, GE used American National Standards Institute/American Institute of Steel Construction (ANSI/AISC) Standard N-690 (1984 Edition) for the design of all steel internal structures. In the DFSER, the staff noted that this standard had not been approved and accepted by the staff, the use of ANSI/ASCE N-690 was DFSER Open Item 3.8.3-1. In SSAR Section 3.8.3 and Tables 3.8-4 and 3.8-9, Amendment 33, GE provided limitations of using the ANSI/AISC N-690 Standard (1984 Edition) in the design of steel internal structures. The use of this standard with these limitations complies with the staff position described in Appendix G of this report. This is acceptable, and DFSER Open Item 3.8.3-1 is resolved. To follow the 10 CFR Part 52 design certification process, any change to the use of ANSI/AISC Standard N-690 (1984 Edition) and ACI Standard 349 (1980 Edition) for the design and construction of containment internal structural elements would involve an unreviewed safety question and, therefore, require NRC review and approval prior to implementation. Furthermore, any requested change to the use of these codes must either be specifically described in the COL application or submitted for license amendment after COL issuance.

The containment concrete and steel internal structures are designed to resist various combinations of dead and live loads, accident-induced loads (including pressure and jet loads), and seismic loads. The load combinations used cover those cases likely to occur and include all loads that may act simultaneously. The containment internal structures are designed and proportioned to remain within the limits in accordance with SRP Section 3.8.3 for the various load combinations. These limits are based on ASME Code, Section III, Division 2, ACI Standard 349 (1980 Edition), and ANSI/AISC Standard N-690 (1984 Edition) for concrete and steel structures, respectively, modified as appropriate for load combinations that are considered extreme.

For the analysis and design, the diaphragm floor, reactor pedestal, and reactor shield wall were included in the finite element model of the RCCV. The computer code STARDYNE was used with all design basis loads considered as static loads. The analysis and design procedures are essentially the same as those used for the RCCV analysis and design. As discussed in Section 3.8.1 of this report, they are acceptable.

The DEPSS, according to SSAR Section 3.8.3.4.4, Amendment 33, is a two-level three-dimensional space frame structure that consists of columns, radial beams, circumferential beams and steel grating and is designed to support the deadweight of non-safety-related equipment and safety-related and non-safety-related piping systems. For the analysis and design of the DEPSS, GE stated in SSAR Section 3.8.3.4.4, Amendment 33, and during the design audits, that the finite element method was used for the analysis and the design was accomplished in accordance with the ANSI/AISC N-690 Standard (1984 Edition). This is also acceptable to the staff. The staff's review and evaluation of the FRS generated through the DEPSS for the input to the piping analysis are discussed in Section 3.7.2 of this report.

The materials of construction and their fabrication, construction, and installation are in accordance with ACI Standard 349 (1980 Edition) and ANSI/AISC Standard N-690 (1984 Edition) for the concrete and steel structures, respectively. In the DFSER, the staff stated the concern about the use of the edition of ASME Code. This was DFSER Open Item 3.8.3-2. In Amendment 33 of the SSAR, GE specified that the 1989 version of ASME Code will be used for the design. This is acceptable, and DFSER Open Item 3.8.3-2 is resolved.

The criteria specified for the analysis, design, and construction of the containment internal structures to account for anticipated loadings and postulated conditions that may be imposed on the structures during their lifetime conform to the established criteria, codes, standards, and specifications acceptable to the staff. These include meeting the guidelines of RG 1.94, Revision 1, RG 1.136, Revision 2, and RG 1.142, Revision 1.

The use of these criteria as defined by the applicable codes, standards, guides, and specifications (on the loads and loading combinations, the design and analysis procedures, the structural acceptance criteria, the materials, the quality control programs, and the testing requirements) provides reasonable assurance that, in the event of earthquakes and various postulated accidents occurring within the containment, the internal structures will withstand the specified design conditions without impairment of the structural integrity or the performance of required safety functions.

During the first two audits, the detailed design calculations for these structures were not available for review. After the staff requested these calculations, GE committed to provide such design information on the containment internal structures. This was DFSER Confirmatory Item 3.8.3-1. Subsequently, during the audit dated February 22 through 25, 1993, the staff reviewed GE's design calculations of the internal structures and found them acceptable. Therefore, DFSER Confirmatory Item 3.8.3-1 is resolved.

The staff concludes that the design of the containment internal structures are acceptable and meets the relevant requirements of 10 CFR Part 50, Section 50.55a, and GDC 1, 2, 4, and 50. This conclusion is based on the following:

1. GE has met the requirements of Section 50.55a and GDC 1 with respect to assuring that the containment internal structures are designed, fabricated, erected, constructed, tested and inspected to quality standards commensurate with its safety function to be performed by meeting the guidelines of regulatory guides and industry standards indicated below.

2. GE has met the requirements of GDC 2 by designing the containment internal structure to withstand the 0.3g SSE with sufficient margin and the combinations of the effects of normal and accident conditions with effects of environmental loadings such as earthquakes and other natural phenomena.

3. GE has met the requirements of GDC 4 by assuring that the design of the internal structures is capable of withstanding the dynamic effects associated with missiles, pipe whipping, and discharging fluids.

4. GE has met the requirements of GDC 50 by designing the containment internal structures to accommodate, with sufficient margin, the design leakage rate, calculated pressure, and temperature conditions, resulting from accident conditions, and by assuring that the design conditions are not exceeded during the full course of accident conditions. In meeting these design requirements, GE has used the recommendations of RGs and industry standards indicated below in the next paragraph. GE has also performed an appropriate analysis that demonstrates the ultimate capacity of the structures and establishes acceptable margin of safety for the design.

The criteria used in the design, analysis, and those proposed for construction of the containment internal structures to account for anticipated loadings and postulated conditions that may be imposed upon structures during their service lifetime conform with established criteria and with codes, standards, and specifications acceptable to the regulatory staff. These include meeting the positions of RG 1.57, Revision 0, RG 1.94, Revision 1, and RG 1.142, Revision 1, and industry standards ACI-349, ASME Boiler and Pressure Vessel Code, Section III, Division 2, "Code for Concrete Reactor Vessels and Containments," Boiler and Pressure Vessel Code, Section III, Subsection NE, ANSI/AISC N-690, "Specification for the Design, Fabrication, and Erection of Steel Safety-Related Structures for Nuclear Facilities," and ANSI N45.2.5.

The use of these criteria as defined by applicable codes, standards, and specifications, the loads and loading combinations; the design and analysis procedures; the structural acceptance criteria; the materials, quality control programs, and special construction techniques; and the testing and in-service surveillance requirements provide reasonable assurance that, in the event of earthquakes and various postulated accidents occurring within the containment, the interior structures will withstand the specified design conditions without impairment of the structural integrity or the performance of required safety functions.

3.8.4 Other Seismic Category I Structures

Other seismic Category I structures within the ABWR design scope are the balance of the RB, which is integral with the RCCV, and the control building. Because GE elected to design the radwaste building substructure to remain structurally intact during an SSE to help contain liquid from a possibly ruptured tank, the radwaste building substructure also is included in this safety evaluation, although it does not house any safety-related systems and components, and hence, is not seismic Category I.

As discussed in Section 3.7.2 of this report, the turbine building is not seismic Category I, but must be capable of withstanding the SSE so as not to impair the safety function of the portion of the MSL and condenser (when used as an alternative leakage path) housed within the turbine building. On May 21, 1992, GE submitted its justification for demonstrating that the turbine building will not fail during and after an SSE. In the DFSER, the staff stated the concern about the design adequacy of the turbine building. This was DFSER Open Item 3.8.4-1. In SSAR Section 3.7.3.16, Amendment 33, GE revised the design procedures for the turbine building. The staff reviewed these procedures and found them acceptable. The details of staff's evaluation is discussed in Section 3.7.2 of this report. DFSER Open Item 3.8.4-1 is resolved.

In SSAR Section 3.8.4, Amendment 33, GE describes the method of dynamic analysis, loads and load combinations, and design procedures and criteria for the design of seismic Category I cable tray, conduit, and their supports. The staff's safety evaluation of the design procedures and criteria for the seismic Category I cable tray and conduit supports is discussed in Section 3.10.2 of this report.

Seismic Category I structures within the ABWR standard plant design are constructed of structural steel or concrete, or both. The structural components consist of slabs, walls, beams, and columns. The major code used in the design of concrete seismic Category I structures is ACI Standard 349 (1980 Edition). For steel seismic Category I structures, ANSI/AISC Standard N-690 (1984 Edition) was used. As discussed in Section 3.8.3 of this report, the use of ANSI/AISC Standard N-690 (1984 Edition) for the design of the seismic Category I structures is acceptable to the staff with the restrictions discussed in Section 3.8.3 of this report. To follow the 10 CFR Part 52 design certification process, any change to the use of ACI Standard 349 (1980 Edition) and ANSI/AISC Standard N-690 (1984 Edition) for the design and construction of the seismic Category I structural elements, would involve an unreviewed safety question and, therefore, require NRC review and approval prior to implementation.

Furthermore, any requested change to the use of these codes must either be specifically described in the COL application or submitted for license amendment after COL issuance.

The concrete and steel seismic Category I structures in the ABWR are designed and proportioned to resist various combinations of dead loads; live loads; environmental loads, including winds, tornados, and SSE; and loads generated by postulated ruptures of high-energy pipes (such as reaction and jet impingement forces, compartment pressures, and impact effects of whipping pipes). GE performed static finite-element analyses of the structures to determine the distribution of structural forces and moments for the various loads and load combinations. The design and analysis procedures used for these seismic Category I structures are acceptable because they are the same as those approved for previous license applications and are in accordance with SRP Section 3.8.4.

According to SSAR Section 3.8.4.2, Amendment 33, the materials of construction and their fabrication, construction, and installation are in accordance with ACI 349 Code (1980 Edition) and ANSI/AISC Standard N-690 (1984 Edition), respectively, for the reinforced concrete and structural steel in the RB, control building, and radwaste building substructure. The use of these codes complies with SRP Section 3.8.4 and is acceptable.

During the design calculation audit conducted on October 12 through 15, 1992, the staff was concerned about the design adequacy of the reactor subcompartment walls and the removable walls under the pressure and thermal loads owing to a high-energy line (CUW lines and reactor core isolation lines) break (HELB). In response to this staff concern, GE provided the evaluation procedures, criteria, and results for the reactor subcompartment walls and removable walls in SSAR Section 3H.4, Amendment 33. From the audit results of the design calculations and the review of the SSAR, the staff concludes that the evaluation procedures and criteria used are reasonable and the design of these walls, including the removable walls are adequate.

In the early SSAR amendments, GE did not account for the effect of the hydrodynamic load on the RB resulting from a safety/relief valve (SRV) discharge or a LOCA in the containment. Because the RB encloses and is structurally integral with the containment shell at each floor level and at the spent fuel grider, the effect of the hydrodynamic load on the RB as a result of a SRV discharge or a LOCA in the containment should be factored into the design. This was DFSER Open Item 3.8.4-2. In SSAR Appendix 3G, Amendment 33, GE specified that the safety-related SSCs as applicable are also analyzed and designed for the dynamic excitations originating from the event of operational transients and LOCA. The input loads considered in the structural dynamic response analysis include condensation oscillation (CO), pool chugging (CH), horizontal vent chugging (HV), SRV discharge, and annulus pressurization (AP). The staff's evaluation about the adequacy of these loads described in SSAR Appendix 3B, Amendment 33, is discussed in Section 6 of this report. In the structural analyses, these loads were classified as pipe nozzle break loads, symmetric loads, and asymmetric loads. A multi-stick lumped-mass model that represents the RB, RCCV, reactor shield wall and pedestal, and RPV was used for the analysis. The damping values recommended in RG 1.61 were considered for SRV and LOCA loads. For the pipe nozzle break loading cases, multi-input excitation modal time-history analyses were performed. For the symmetric and asymmetric loading cases, a frequency response method was used. In SSAR Appendix 3G, GE provided the structural responses (accelerations and displacements) and the FRS owing to the hydrodynamic loads applied on the RB as a result of an SRV discharge and a LOCA. From review of the SSAR and the audit conducted on February 22 through 24, 1993, the staff concludes that the structural responses, including the FRS generated and the structural design (RB) against these loads, are reasonable. DFSER Open Item 3.8.4-2 is resolved.

During the early design calculation audits, the staff reviewed the static analyses that calculated the structural element forces and moments resulting from the various loads and load combinations acting on the control building and radwaste building substructure. The detailed design calculations for the RB, control building, and radwaste building substructure, however, were not available to the staff for review. In addition, GE did not complete the implementation of the QA programs for the static analyses and detailed design calculations for both the control building and the radwaste building substructure. GE committed to provide the detailed design calculations to the staff for review and to complete the implementation of the QA programs for the control building and the radwaste building substructure. These two issues were DFSER Confirmatory Items 3.8.4-1 and 3.8.4-2, respectively. During the audit conducted on February 22 through 25, 1993, the staff reviewed GE's final design calculations for these buildings and the procedures for implementing the QA program and found them acceptable. Therefore, DFSER Confirmatory Item 3.8.4-1 is resolved. In a letter dated September 15, 1993, GE certified the implementation of the QA program for the design calculations of the reactor, control, and radwaste buildings. Therefore, DFSER Confirmatory Item 3.8.4-2 is resolved.

During the audit conducted on October 12 through 15, 1992, in the Bechtel San Francisco office, the staff was concerned about the design of the embedded portion of the exterior walls of the seismic Category I structures. GE used the methods described in Bechtel Power Corporation Topical Report, BC-TOP-4A, "Seismic Analysis of Structures and Equipment for Nuclear Power Plants," Revision 4, to consider static soil pressure and the soil pressure induced by the earthquake and the design of these embedded exterior walls. The staff noted that these BC-TOP-4A methods for calculating the dynamic lateral earth pressures on the embedded walls had not been reviewed and accepted by the staff. As a result, GE agreed to follow the guidelines documented in the staff position for the embedded wall and retaining wall design (Appendix H of this report) and to design the embedded portion of the exterior walls of all seismic Category I structures. During the staff audit on February 22 through 25, 1993, in the Bechtel San Francisco office, the staff found the design methods and the design results of the exterior embedded walls acceptable.

The detailed design calculations of roof structures against severe weather phenomena, such as heavy rainfall and snow loading were unavailable during the earlier design audits. This was DFSER Open Item 3.8.4-3. In SSAR Appendix 3H, Amendment 30, GE stated that the design rainfall is 493 mm/hr (19.4 in./hr) and the roof of the seismic Category I structures is designed to have parapets with scuppers to supplement roof drains or designed without parapets so that excessive ponding of water cannot occur. The design rainfall of 493 mm/hr (19.4 in./hr) and the drainage system provisions form a reasonable design to prevent excessive roof ponding. This is acceptable, and DFSER Open Item 3.8.4-3 is resolved.

The seismic Category I structures for the ABWR standard plant were initially designed to withstand a maximum tornado wind speed of 418 km/hr (260 mph). The staff raised in the DSER (SECY-91-153) its concern with the acceptance of this design tornado wind speed (Outstanding Issues 4, 8, and 9). In response, GE increased the design tornado wind speed to 483 km/hr (300 mph). GE also revised the tornado-generated missile spectrum, specified in ANSI/ANS 2.8, to the Spectrum I specified in SRP Section 3.5.1.4. In a letter dated May 29, 1992, GE informed the staff that, based on its preliminary evaluation of the effect of the revised tornado wind and tornado missile loadings, the RB superstructure and roof design required additional thickness and the roof purlins required strengthening. These structural changes would affect the seismic model and hence the seismic response of the RB. In the DFSER, the staff was concerned that the resulting modification of the seismic model would affect the seismic analysis and design results contained in several sections and appendices in Chapter 3 of the SSAR. This was DFSER Open Item 3.8.4-4. During the audit on February 22 through 25, 1993, the staff reviewed GE's seismic reanalysis results and design calculations and found that the resolution for this open item is acceptable. DFSER Open Item 3.8.4-4 is resolved.

GE used the NASTRAN computer code to perform the static analysis and to calculate the structural element forces and moments of the control building and radwaste building subjected to the various loads and load combinations. A finite element analysis model for each building was used in the analysis. The load combinations for the reinforced concrete structures are in accordance with ACI Standard 349 (1980 Edition).

In the DFSER, the staff noted that in analyzing the control building, GE had not considered the effects of winds, tornados, and tornado missiles and, in analyzing the radwaste building, GE had not considered the effect of winds and had used incorrectly calculated soil pressure loads. These were DFSER Confirmatory Items 3.8.4-3 and 3.8.4-4, respectively. In SSAR Appendix 3H, Amendment 33, GE documented its design results of the control building and radwaste building for the wind, tornado and tornado missile loadings, and soil pressure loads, which is acceptable; therefore, DFSER Confirmatory Items 3.8.4-3 and 3.8.4-4 are resolved. The static analysis methods and the analysis results for the element forces and moments for the control building and radwaste building are acceptable.

Sufficient descriptive and design information for the seismic Category I structures should be provided in the SSAR to meet guidelines in Section 3.8.4.1 of RG 1.70, "Standard Format and Content of Safety Analysis Report for Nuclear Power Plants," Revision 3. Such information typically includes the floor plans, roof plans, vertical sections, and structural models used in the static analysis to calculate element forces and moments, configurations of major structural components, and arrangements of reinforcements in major concrete structural members. For the RB structure, SSAR Figures 3.8-1 through 3.8-9 and Section 3H.3 show the design information that meets the SRP guidelines. For the control building and radwaste building substructure, however, the early SSAR did not provide the descriptive and design information similar to that provided for the RB. This was DFSER Open Item 3.8.4-5. In SSAR Appendix 3H and Chapter 21, Amendment 33, GE provided the design description and design drawings for the control and radwaste buildings. The staff reviewed this information and found it acceptable; therefore, DFSER Open Item 3.8.4-5 is resolved.

In the DFSER, the staff noted that the COL applicant should identify and describe any other seismic Category I structures that are not within the ABWR scope as a part of its application. The staff will review its analysis and design on a case-by-case basis. The staff also noted that the COL applicant should ensure that the settlement of adjacent buildings will be such that the integrity of underground piping or tunnel will not be jeopardized. This was DFSER COL Action Item 3.8.4-1. In SSAR Sections 3.8.4 and 3.8.6.4, Amendment 33, GE stated that the COL applicant shall identify all seismic Category I structures, and in SSAR Section 2.3.2.36, Amendment 33, GE stated that the COL applicant shall perform stability evaluation of all safety-related facilities, including foundation rebound, settlement, differential settlement, and bearing capacity. These COL actions are acceptable.

In conclusion, the design of safety-related structures other than containment or containment interior structures are acceptable and meets the relevant requirements of 10 CFR Part 50, Section 50.55a, and GDC 1, 2, and 4. This conclusion is based on the following:

1. GE has met the requirements of Section 50.55a and GDC 1 with respect to assuring that the safety-related structures other than containment are designed, fabricated, erected, and constructed to quality standards commensurate with its safety function to be performed by meeting the guidelines of regulatory guides and industry standards indicated below.

2. GE has met the requirements of GDC 2 by designing the safety-related structures other than containment to withstand the 0.3*g* SSE with sufficient margin and the combinations of the effects of normal and accident conditions with the effects of environmental loadings such as earthquakes and other natural phenomena.

3. GE has met the requirements of GDC 4 by assuring that the design of the safety-related structures is capable of withstanding the dynamic effects associated with missiles, pipe whipping, and discharging fluids.

4. GE has met the requirements of Appendix B to 10 CFR Part 50 because their QA program provides adequate measures for implementing guidelines relating to structural design audits.

The criteria used in the analysis, design, and construction of all the plant seismic Category I structures to account for anticipated loadings and postulated conditions that may be imposed upon each structure during its service lifetime are in conformance with established criteria, codes, standards, and specifications acceptable to the regulatory staff. These include meeting the guidelines of RG 1.69, Revision 0, RG 1.91, Revision 1, RG 1.94, Revision 1, RG 1.115, Revision 1, RG 1.142, Revision 1, and RG 1.143, Revision 1, and industry standards ACI-349 and ANSI/AISC N-690, "Specifications for the Design, Fabrication, and Erection of Steel Safety-Related Structures for Nuclear Facilities."

The use of these criteria as defined by applicable codes, standards, and specifications, the loads and loading combinations; the design and analysis procedures; the structural acceptance criteria; the materials, quality control, and special construction techniques; and the testing and inservice surveillance requirements provide reasonable assurance that, in the event of winds, tornados, earthquakes, and various postulated accidents occurring within the structures, the structures will withstand the specified design conditions without impairment of structural integrity or the performance of required safety functions.

3.8.5 Foundations

The ABWR design employs separate reinforced-concrete mat foundations for major seismic Category I structures. The RB foundation, which is integral with the containment foundation, supports the containment structure, reactor pedestal, other internal structures, and the balance of RB structure. Even though the containment structure foundation is integral with the RB foundation, it is a portion of the foundation within the perimeter of the containment structure. Therefore, the foundation was designed as a part of the containment boundary. The concrete foundations were designed to resist various combinations of dead loads, live loads, environmental loads (including winds, tornados, and SSE), and loads generated by postulated ruptures of high-energy pipes. The original detailed design information such as the factor of safety against overturning for the RB was calculated and provided in SSAR Appendix 3H, Amendment 4. However, in the DFSER, the staff noted that no such information was given in the SSAR for the control building and the radwaste building substructure. This was DFSER Open Item 3.8.5-1. In addition, the staff also noted that if foundation waterproofing is used, the COL applicant should evaluate the capability of the foundations to transfer seismic shear forces. This was DFSER COL Action Item 3.8.5-1. In SSAR Appendix 3H, Amendment 33, GE provided the loading condition used for the dynamic overturning, sliding and flotation analyses of the RB, control building, and radwaste building and the evaluation results (safety factors against overturning, sliding, and flotation) of these three buildings. From review of the SSAR Appendix 3H and the design calculation audit conducted on February 22 through 25, 1993, the staff concludes that the reactor, control, and radwaste buildings will be dynamically stable under the specified SSE and DFSER Open Item 3.8.5-1 is resolved. As for COL Action Item 3.8.5-1, which deals with the potential for the foundation mat to slide over the plastic sheets that are generally used as waterproofing material, SSAR Sections 3.8.5.4 and 3.8.6.1, Amendment 33, stated that the COL applicant will evaluate the capability of the foundation to transfer shear loads where foundation waterproofing material is used. This is acceptable. In addition, based on the current practice of foundation construction and the staff's understanding, a layer of gravel will be placed on the excavated foundation surface for the soil site and the excavated rock foundation surface will be roughened before pouring concrete and placing the waterproofing material. The treated foundation surface will increase the fiction between the structural foundation and the supporting foundation surface. Therefore, the staff concludes that the treated foundation surface will be capable to transfer the seismic shear loads.

In the DFSER, the staff noted that GE had not provided the editions of the ASME and ACI Codes. This was DFSER Open item 3.8.5-2. Subsequently, GE, in SSAR Section 3.8.5.2, Amendment 33, identified code editions. The major code used in the design of concrete mat foundations is the 1980 Edition of the ACI 349 Standard, except for the portion of the foundation within the containment boundary for which the 1989 Edition of ASME Code, Section III, Division 2, was used. The design and analysis procedures, the materials of construction and their fabrication, construction code, and installation used for the seismic Category I foundations, are in accordance with the procedures in ACI 349 (1980 Edition) and ASME Code, Section III, Division 2 (1989 Edition). The seismic Category I foundations are designed and proportioned to remain within the limits of these design codes for the applicable load combinations, including those that are considered extreme. The staff noted that this edition is referenced in 10 CFR 50.55a and is, therefore, acceptable. GE has also included this information in the SSAR; therefore, DFSER Open Item 3.8.5-2 is resolved.

To follow the 10 CFR Part 52 design certification process, any change to the use of ACI 349 Code (1980 Edition) and ASME Code (1989 Edition) for the design and construction of the seismic Category I building foundations would involve an unreviewed safety question and, therefore, require NRC review and approval prior to implementation. Furthermore, any requested change to the use of these codes must either be specifically described in the COL application or submitted for license amendment after COL issuance.

During the first two design calculation audits, the staff found that the detailed design calculations for the foundations of all seismic Category I structures were not available for review, and in addition, GE did not complete the implementation of the QA programs for the design of the foundations of the control building and radwaste building substructure. In the DFSER, the staff noted that GE had committed to complete the detailed design calculations for the foundations of all seismic Category I structures and to complete the implementation of the QA programs for the foundations of the control building and radwaste building substructure. These were DFSER Confirmatory Items 3.8.5-1 and 3.8.5-2. During the audit conducted on February 22 through 25, 1993, GE provided the detailed design calculations of the seismic Category I structure foundations for review, which was acceptable. Therefore, DFSER Confirmatory Item 3.8.5-1 is resolved. In a letter dated September 15, 1993, GE certified that the implementation of the QA program for the design calculations of the seismic Category I building foundations. This is acceptable. Therefore, DFSER Confirmatory Item 3.8.5-2 is resolved. The staff concludes that the design of the seismic Category I foundations are acceptable and meets the relevant requirements of 10 CFR Part 50, section 50.55a, and GDC 1, 2, and 4. This conclusion is based on the following:

1. GE has met the requirements of Section 50.55a and GDC 1 with respect to assuring that the seismic Category I foundations are designed, fabricated, erected, constructed, tested and inspected to quality standards commensurate with its safety function to be performed by meeting the guidelines of regulatory guides and industry standards indicated below.

2. GE has met the requirements of GDC 2 by designing the seismic Category I foundation to withstand the 0.3*g* SSE with sufficient margin and the combinations of the effects of normal and accident conditions with the effects of environmental loadings such as earthquakes and other natural phenomena.

3. GE has met the requirements of GDC 4 by assuring that the design of seismic Category I foundations is capable of withstanding the dynamic effects associated with missiles, pipe whipping, and discharging fluids.

The criteria used in the analysis, design, and construction of all the plant seismic Category I foundations to account for anticipated loadings and postulated conditions that may be imposed upon each foundation during its service lifetime are in conformance with the established criteria, codes, standards, and specifications acceptable to the regulatory staff. These include meeting the guidelines of RG 1.142 and industry standards ACI 349 and ANSI/AISC N-690, "Specification for Design, Fabrication and Erection of Steel Safety-Related Structures for Nuclear Facilities."

The use of these criteria as defined by applicable codes, standards, and specifications; the loads and loading combinations; the design and analysis procedures; the structural acceptance criteria; the materials, quality control, and special construction techniques; and the testing and in-service surveillance requirements provide reasonable assurance that, in the event of winds, tornados, earthquakes, and various postulated events, seismic Category I foundations will withstand the specified design conditions without impairment of structural integrity and stability or the performance of required safety functions.

3.8.6 Certified Design Material

In GE's CDM, dated May 30, 1992, GE provided design descriptions and ITAAC for several structural systems. In the DFSER, the staff identified 11 open items related to these structural systems. In response to the open items, GE submitted system- and building-specific ITAAC for staff review. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Section 14.3 of this report.

3.9 Mechanical Systems and Components

SRP Sections 3.9.1 through 3.9.6 address review of the structural integrity and functional capability of various safety-related mechanical components. The review is not limited to ASME Code components and supports, but is extended to other components such as control rod drive mechanisms, certain reactor internals, and any safety-related piping designed to industry standards other than the ASME Code. The staff reviewed such issues as load combinations, allowable stresses, methods of analysis, summary of results, and pre-operational testing. The staff's evaluation focused on determining whether there is adequate assurance of a mechanical component performing its safety-related function under all postulated combinations of normal operating conditions, system operating transients, postulated pipe breaks, and seismic events.

3.9.1 Special Topics for Mechanical Components

The staff reviewed the information in SSAR Section 3.9.1 related to the design transients and methods of analysis used for all seismic Category I components, component supports, core support (CS) structures, and reactor internals designated as Class 1, 2, 3 and CS under ASME Code, Section III, and those not covered by the Code. It reviewed the assumptions and procedures used for the inclusion of transients in the design and fatigue evaluation of ASME Code Class 1 and CS components. It also reviewed the computer programs used in the design and analysis of seismic Category I components and their supports, as well as experimental and inelastic analytical techniques.

SSAR Table 3.9-1 lists the design transients for five plant operating conditions that are applicable to the design of systems, components, and equipment, and, as an example, the number of either plant operating events or cycles for each of the design transients used in the design and fatigue analyses of the reactor pressure vessel. The operating conditions are as follows:

(1) ASME Service Level A - normal conditions

(2) ASME Service Level B - upset conditions, incidents of moderate frequency

(3) ASME Service Level C - emergency conditions, infrequent incidents

(4) ASME Service Level D - faulted conditions, low-probability postulated events

(5) testing conditions

The number of cycles in the original list of transients in SSAR Table 3.9.1 appeared to be based on a 40-year life. In the DFSER, the staff noted that for a design life of 60 years, the number of cycles for each transient should be increased by a factor of 1.5 (DFSER Open Items 3.9.1-1 and 14.1.3.3.5.2-1). In a letter dated October 22, 1993, GE responded to this item by providing a revision to Table 3.9-1. In this revision, the original number of cycles resulting from transients for daily and weekly reduction to 50-percent power, control rod pattern change, loss-of-feedwater heaters, and turbine stop valve full closure were increased by a factor of 1.5. The number of cycles for the remaining transients in Table 3.9-1 are identical to the original numbers. GE stated that the original number of cycles were based on BWR operating experience and are conservative for a 60-year design life. The staff agrees that BWR operating experience provides an acceptable basis for estimating the number of cycles anticipated for a 60-year design life. Therefore, Open Items 3.9.1-1 and 14.1.3.3.5.2-1 are resolved.

GE used computer codes to analyze mechanical components. The computer programs used for static and dynamic analyses to determine the structural and functional integrity of seismic Category I Code and non-seismic Category I code items are included in SSAR Appendix 3D. Design control measures to verify the adequacy of the design of safety-related components are required by Appendix B to 10 CFR Part 50. SSAR Section 3.9.1.2, states that the quality of the programs and the computer results are controlled either by GE or by outside computer program developers and the programs are verified by one or more of the methods recommended in SRP Section 3.9.1.

The NRC staff's review of this issue was, in part, based on an evaluation of the adequacy of the GE computer program used in the representative ABWR piping analyses that were audited by the staff on March 23 through 26, 1992, at GE's offices in San Jose, California. This was accomplished by the staff performing an independent piping analysis to confirm the adequacy of these GE analyses. In the DFSER, the computer program adequacy was DFSER Open Items 3.9.1-2 and 14.1.3.3.4.1-1. The resolution of these issues is discussed in Section 3.12.4.1 of this report.

SSAR Section 3.9.1.3 identifies several components for which experimental stress analysis is performed in conjunction with analytical evaluation. The experimental stress analysis methods are used in compliance with Appendix II of the ASME Code, Section III. This meets the guidance of SRP Section 3.9.1 and is, therefore, acceptable.

SSAR Section 3.9.1.5 states that inelastic analysis is only applied to ABWR components to demonstrate the acceptability of three types of postulated events. Each event is an extremely low-probability occurrence and the equipment affected by these events would not be reused. A discussion of each of these three postulated events follows:

(1) Postulated Pipe Rupture

As discussed in Section 3.6.2 of this report, ruptures are postulated in certain piping systems in accordance with the conservative guidelines in SRP Section 3.6.2. For some full-diameter ruptures, pipe whip restraints may be required to protect safety-related components or equipment from the dynamic effects of a whipping pipe. Inelastic analyses are used in the design of such restraints to ensure that they will withstand this low probability loading without loss of structural integrity. In these analyses, metallic members of the restraint are limited to an allowable strain of 50 percent of the ultimate uniform strain of the impacted material. This criterion, which is discussed in SSAR Section 3.6.2.3.3, is consistent with the guidelines in SRP Section 3.6.2, and is acceptable.

(2) Postulated Blowout of a RIP Motor Casing as a Result of Weld Failure

This postulated event is discussed in SSAR Section 5.4.1.5. Each RIP is contained in an ASME Class 1 pressure boundary housing that is welded to a stub tube in the reactor pressure vessel lower head. The following low-probability failure scenario is postulated, and inelastic analysis is used for the design of one item. SSAR Figure 5.4-1 provides a sketch of the RIP cross section.

- A guillotine failure of the ASME Class 1 weld is assumed. Subsequent to this event, the stretch tube, which normally functions to hold the pump diffuser in place, is the first member to resist ejection of the pump housing. On the basis of an elastic analysis, the stresses in the stretch tube are calculated to be 85 percent of its minimum specified ultimate strength. The stretch tube could be reasonably considered to mitigate this event without failure, however, this failure scenario is continued as discussed below.

- In the event that the stretch tube also fails subsequent to the weld failure, the RIP shaft and the thrust bearing are subjected to the ejection load. The weakest link in this remaining load path is the bearing-to-shaft bolt. An elastic analysis of this bolt also results in stresses that are less than its ultimate strength. Therefore, this bolt would not be expected to fail.

- If the weld, stretch tube, and bearing-to-shaft bolt all sequentially fail, then the external vertical restraints are subjected to the ejection loads. These restraints are stainless steel rods that run from lugs on the vessel to lugs on the RIP motor cover. The structural integrity of the attachment lugs, bolts, and rod clevises under this loading condition is demonstrated by elastic analysis. The use of inelastic analysis methods is limited to the middle slender body of the rod. The allowable strain used in this analysis is identical to that used for pipe whip restraints and is acceptable as discussed in Item 1 above.

(3) Postulated Blowout of a CRD Housing as a Result of Weld Failure

This postulated event is discussed in SSAR Section 4.6.1.2.2.9. A schematic of the internal CRD blowout support is shown in Figure 4.6.7 of the SSAR. This event begins with the postulated full diameter failure of the ASME Class 1 weld that attaches the CRD housing to the stub tube in the bottom head of the reactor pressure vessel (Point A in Figure 4.6.7). In the unlikely event of such a failure, ejection of the CRD is prevented by the core support plate, the CRD guide tube, the CRD housing, and the CRD outer tube. Each of these components was demonstrated to be capable of sustaining this once-in-a-lifetime ejection load without failure. This was demonstrated by use of elastic analyses in accordance with ASME Section III, Appendix F, rules for all parts of these components with the exception of the cylindrical bodies of the guide tube, housing, and outer tube. These cylindrical portions were analyzed by inelastic analysis methods. The allowable strain used in these analyses is identical to that used for pipe whip restraints and is acceptable as discussed in Item 1 above.

On the basis of these evaluations, the staff concludes that the design transients and resulting load combinations with appropriate specific design and service limits for mechanical components and supports meet the applicable portions of GDC 1, 2, 14, and 15; Appendix B to 10 CFR Part 50; Appendix A to 10 CFR Part 100; and SRP Section 3.9.1.

GE has met GDC 14 and 15 by demonstrating that the design transients and resulting loads and load combinations with the appropriate specific design and service limits for designing ASME Code, Class 1 and CS components and supports and reactor internals provide a complete basis for the design of the RCPB for all conditions and events expected over the service lifetime of the plant.

GE has met GDC 2 and Appendix A to 10 CFR Part 100 by including seismic events in design transients that serve as the design basis for withstanding the effects of natural phenomena.

GE has met Appendix B to 10 CFR Part 50, and GDC 1 by submitting information that demonstrates the applicability and validity of the design methods and computer programs used for the design and analysis of seismic Category I structures designated as ASME Code, Class 1, 2, 3, and CS and those not covered by the Code within the present state-of-the-art limits and by having design control measures that are consistent with the applicable guidelines of SRP 3.9.1. This is acceptable for ensuring the quality of the computer programs. If the COL applicant opts to use computer programs different from those used by GE for the design of any safety-related item with the exception of piping systems, the guidelines of SRP 3.9.1 must be met for such programs. The staff's review of the requirements for piping systems is included in its evaluation of the design acceptance criteria (DAC) and ITAAC for generic piping designs in Section 3.12 of this report.

3.9.2 Dynamic Testing and Analysis of Systems, Components, and Equipment

The staff reviewed the methodology, testing procedures, and dynamic analyses that GE used to ensure the structural integrity and functionality of piping systems, mechanical equipment, and their supports under vibratory loadings. The staff's review acceptance criteria included meeting the requirements of (1) GDC 14 and 15 by conducting the piping vibration, thermal expansion, and dynamic effects testing; (2) GDC 2 by reviewing the seismic subsystem analysis methods; (3) GDC 1 and 4 by committing to testing the dynamic responses of structural components in the reactor caused by steady-state and operational flow transient conditions; (4) GDC 1 and 4 by committing to the flow-induced vibration testing of reactor internals to be conducted during the pre-operational and startup test program; and (5) GDC 2 and 4 by committing to the dynamic analysis methods to confirm the structural design adequacy and functional capability of the reactor internals and piping attached to the reactor vessel when subjected to loads from a LOCA in combination with an SSE.

3.9.2.1 Piping Pre-operational Vibration and Dynamic Effects Testing

Piping vibration, thermal expansion, and dynamic effects testing will be conducted during a pre-operational testing program. The purpose of these tests is to ensure that the piping vibrations are within acceptable limits and that the piping system can expand thermally in a manner consistent with the design intent. During the plant's pre-operational and startup testing program, which is also described in SSAR Section 14.2.12, all ABWR plants will test various piping systems for abnormal, steady-state, or transient vibration and for restraint of thermal growth. Systems to be monitored will include (1) ASME Code, Class 1, 2, and 3 piping systems; (2) high-energy piping systems inside seismic Category I structures; (3) high-energy portions of systems whose failure could reduce the functioning of seismic Category I plant features to an unacceptable safety level; and (4) seismic Category I portions of moderate-energy piping systems located outside the containment. Steady-state vibration, whether flow induced or caused by nearby vibrating machinery, could cause up to 10^{10} cycles of stress in the pipe during the 60-year design life of the plant. For this reason, the staff requires that the stresses associated with steady-state vibration be minimized and limited to acceptable levels. The test program will consist of a mixture of instrumented measurements and visual observations by qualified personnel. In SSAR Section 3.9.2.1 and Table 1.8-21, GE indicates that detailed test specifications will be prepared in accordance ANSI/OM-1987, Part 3, and ANSI/OM-1986, Part 7.

The staff finds that these criteria will provide an acceptable level of safety for a piping system to withstand the effects of vibration and thermal expansion during the plant's 60-year design life. This conforms to the applicable guidelines of SRP Section 3.9.2 and is acceptable.

The staff concludes that GE meets the relevant requirements of GDC 14 and 15 with regard to the design and testing of the RCPB. This provides reasonable assurance that there is a low probability of rapidly propagating failure and of gross rupture to ensure that design conditions will not be exceeded during normal operation, including anticipated operational occurrences, by having an acceptable vibration, thermal expansion, and dynamic effects test program that will be conducted during startup and initial operation of specified high- and moderate-energy piping, including all associated restraints and supports. The tests provide adequate assurance that the piping and piping supports are designed to withstand vibrational dynamic effects as a result of valve closures, pump trips, and other operating modes associated with the design-basis flow conditions. In addition, the tests provide assurance that adequate clearances and free movement of snubbers exist for unrestrained thermal movement of piping and supports during normal system heatup and cooldown operations. For the planned tests, loads similar to those experienced during transient and normal reactor operations will be developed.

3.9.2.2 Seismic Subsystem Analysis

The staff's review was performed in accordance with SRP Section 3.9.2 and consisted of an evaluation of SSAR Section 3.7.3. Areas reviewed were seismic analysis methods, determination of the number of earthquake cycles, basis for the selection of frequencies, the combination of modal responses and spatial components of an earthquake, criteria used for damping, torsional effects of eccentric masses, interaction of other piping with seismic Category I piping, and buried seismic Category I piping systems.

The ABWR plant is designed for an SSE ground motion defined by a RG 1.60, "Design Response Spectra for Seismic Design of Nuclear Power Plants," Revision 1, response spectrum anchored to a peak ground acceleration of 0.3*g*. Amplified building response spectra are generated for the ABWR plant to account for varying soil properties in the United States by enveloping 14 site conditions. The enveloping amplified building response spectra provided in the SSAR are used for the design and analyses of the ABWR piping systems.

The staff recognizes that the enveloping amplified building response spectra for the ABWR plant contain conservatisms that might be excessive for certain specific site conditions. Accordingly, the staff's position is that when the SSE response spectrum is defined by a RG 1.60 response spectrum anchored to a peak ground acceleration of 0.3*g*, the site-specific soil properties may be used to generate the amplified building response spectra. The method used to generate the amplified building response spectra must be consistent with the method described in SSAR Section 3.7.2, as approved by the staff. The staff's evaluation of the method used by GE for generating the amplified building response spectra is provided in Section 3.7.2 of this report.

GE performed the system and subsystem analyses on an elastic basis. Multidegree-of-freedom modal response spectrum and time history methods formed the basis for the analyses of all major seismic Category I systems and components. When the response spectrum method was used, modal responses were combined by the SRSS rule. Closely spaced modes were combined using the criteria of RG 1.92, "Combining Modal Responses and Spatial Components in Seismic Analysis," Revision 1. GE considered all modes with frequencies below 33 Hz in computing equipment and component response for seismic loadings. For seismic analysis, consideration of high-frequency modes to preclude missing mass effects should also be included. The staff's guidelines for this are provided in SRP Section 3.7.2, Revision 2, Appendix A. In the DFSER, the staff noted that the SSAR should be revised to reflect this staff position or, if an alternative method is used, then the details of its basis should be submitted to the staff for review and approval before its use. This was DFSER Open Item 3.9.2.2-1 and part of DFSER Confirmatory Item 14.1.3.3.5.6-1. In Amendment 23 to the SSAR, GE added Section 3.7.3.7.3, "Methodologies Used to Account for High Frequency Modes." The information in this new section to the SSAR is consistent with SRP 3.7.2, Revision 2, Appendix A, and is acceptable. GE has included this information in Section 3.7.3.7.3 of the SSAR. DFSER Open Item 3.9.2.2-1 is resolved. DFSER Confirmatory Item 14.1.3.3.5.6-1 is discussed in Section 3.12.5.6 of this report.

For the dynamic analysis of seismic Category I piping, each system is idealized as a mathematical model consisting of lumped masses interconnected by elastic members. The stiffness matrix for the piping system is determined using the elastic properties of the pipe. This includes the effects of torsional, bending, shear, and axial deformations as well as change in stiffness as a result of the curved members. Next, the mode shapes and the undamped natural frequencies are obtained. The dynamic response of the system is calculated by using the response spectrum method of analysis. For a piping system that is supported at points with different dynamic excitations, the response analysis is performed using an enveloped response spectrum. As an alternative to the enveloped response spectrum method, GE chose to use the multiple-support excitation analysis method. When this method is used, the staff's position is that the responses resulting from motions of supports between two or more different support groups may be combined by the SRSS method if a support group is defined by supports that have the same time history input. This usually means all supports located on the same floor, or portions of a floor in a structure. In response to RAI Q210.26 in Amendment 11 to the SSAR, GE committed to use this definition. In the DFSER, the staff requested that the SSAR be revised to include this definition. This was DFSER Confirmatory Item 3.9.2.2-1. In Amendment 23, Section 3.7.3.8.1.10, "Multiply-Supported Equipment and Components with Distinct Inputs," was added to the SSAR. This new section includes the requested definition, which is acceptable. GE has included this information in Section 3.7.3.8.1.10 of the SSAR. The staff finds this alternative to the enveloped response spectrum method acceptable, and DFSER Confirmatory Item 3.9.2.2-1 is resolved.

The staff reviewed the method for selecting the number of masses or degrees of freedom in the piping mathematical model to determine its dynamic response. On the basis of the staff's audit conducted March 23 through 26, 1992, of GE's internal documents, pipe and fluid masses are lumped at nodes that are selected to coincide with the locations of large masses (e.g., valves, pumps, and tanks) and with locations of significant geometric changes (e.g., pipe elbows, reducers, and tees). Additional mass points are selected to ensure that the spacing between any two adjacent piping nodes and masses is no greater than an idealized value. This value corresponds to the length of a simply supported beam with a uniformly distributed mass whose undamped natural frequency is equal to the cut-off frequency. Because this approach, in effect, will capture all modes up to the cut-off frequency, it is acceptable. In the DFSER, the staff noted that the SSAR should be revised to reflect this described approach. This was DFSER Confirmatory Items 3.9.2.2-2 and 14.1.3.3.4.2-1. In Amendment 23 to the SSAR, GE revised Section 3.7.3.3.1.2 to respond to this item. The staff's evaluation of this issue is discussed in Section 3.12.4.2 of this report.

The effect of pipe supports stiffness on the piping response is considered in the analytical model. Supports must be modeled in accordance with the SSAR. If supports are not modeled as stated in the SSAR, justification will be provided to validate the stiffness values used in the piping model. The justification should include verification that the generic values are representative of the types of pipe supports used in the piping system. This alternative approach to use generic stiffness values and its bases should be submitted to the staff for review and approval before its use. In the DFSER, this was identified as COL Action Item 14.1.3.3.4.2-1. The resolution of this DFSER COL Action Item is discussed in Section 3.12.4.2 of this report.

Additionally, because the amplified response spectra are generally specified at discrete building node points, any additional flexibility between these points and the pipe support (e.g., supplementary steel) should also be submitted to the staff. In the DFSER, the staff requested that the SSAR be revised to incorporate this information. This was DFSER Open Item 3.9.2.2-2 and DFSER Confirmatory Item 14.1.3.3.4.2-2. The resolution of this issue is discussed in Section 3.12.4.2 of this report.

When piping terminates at non-rigid equipment (e.g., tanks, pumps, or heat exchangers), the piping analytical model should consider the flexibility and mass effects of the equipment. In the DFSER, the staff requested that the SSAR be revised to address how the flexibility and masses of equipment attached to the piping are to be modeled. This was DFSER Open Item 3.9.2.2-3 and DFSER Confirmatory Item 14.1.3.3.4.2-3. The resolution of this issue is discussed in Section 3.12.4.2 of this report.

When analyzing piping systems, the size of the mathematical model might exceed the capacity of the computer program when large and small bore piping are included. Thus, the small bore branch lines are generally decoupled from the large bore main piping. Originally, the SSAR did not provide any criteria for the decoupling of the piping systems in the analysis model. However, in a letter to the NRC dated February 24, 1992, GE provided a decoupling criteria in a GE draft document entitled "ABWR SSAR Main Steam, Feedwater and SRVDL Piping Systems Design Criteria and Analysis Methods," Revision 0, dated February 1992. In this document, GE stated that when the ratio between pipe diameters of the branch line to main line is less than one-third, the branch line can be excluded from the piping model of the main line. Since GE is using this criterion for all piping systems in the ABWR plant, the staff requested the basis for the 1:3 ratio. In addition, GE was requested to define how the mass effect of the decoupled line is accounted for in the model of the main line and how the frequency ratio effect (or resonant amplification of the main line) is accounted for in the modeling and analysis of the branch line. In the DFSER, the staff noted that GE should revise its SSAR to include this information for the staff's review (DFSER Open Items 3.9.2.2-4 and 14.1.3.3.4.4-1). In Amendment 23 to the SSAR, GE revised Sections 3.7.3.3.1.3, 3.7.3.3.1.4, and 3.7.3.8.1.9 to respond to these open items. The staff's evaluation of this issue is discussed in Section 3.12.4.4 of this report.

In the DFSER, the staff noted that GE had not provided the staff with any specific information about the method to be used for the structural design of small bore piping systems and instrumentation lines in the ABWR plant. The staff requested that this information be included in the SSAR (DFSER Open Items 3.9.2.2-5 and 14.1.3.3.3.6-1). In Amendment 23, Section 3.7.3.8.1.9, "Design of Small Branch and Small Bore Piping," was added to the SSAR. The staff's evaluation of this issue is discussed in Section 3.12.3.6 of this report.

RG 1.61, "Damping Values for Seismic Design of Nuclear Power Plants," Revision 1, contains recommended values of damping to be used in the seismic analysis of SSCs. In addition, RG 1.84 (Revision 25, May 1988) conditionally endorses ASME Code Case N-411-1. The damping values used by GE are the same as those specified in either RG 1.61 or ASME Code Case N-411-1 as permitted by RG 1.84. These criteria are acceptable.

The staff reviewed the issue of modal damping for composite structures during the audit conducted on March 23 through 26, 1992, at GE's offices in San Jose, California. The GE SSAR did not describe the application of modal damping for composite structures in the analysis of piping systems. However, GE's internal document entitled, "Piping Systems Design Criteria and Analysis Methods," contained a table of damping values for various types of piping supports. The damping values for the piping supports (e.g., snubbers and struts) were higher than the damping values tabulated for the piping. GE indicated that these values were presented because modal damping for composite structures could be used in a response spectrum analysis as an option. In the DFSER, the staff noted that if GE plans to use the modal damping for composite structures as an option for piping analysis, then GE has to include a description and justification of the approach in the SSAR for staff review and approval before its use. This was DFSER Open Items 3.9.2.2-6 and 14.1.3.3.5.17-1. In Amendment 23 to the SSAR, GE revised Section 3.7.3.8.1.7 to respond to these open items. The staff's evaluation of this issue is discussed in Section 3.12.5.17 of this report.

SSAR Section 3.7.3.12 outlines criteria for the analysis of buried seismic Category I piping systems. In the DFSER, the staff noted that GE had not given any detailed information on how the criteria are to be applied in the design of buried piping. Also, it was not clear if the buried piping within the scope of design certification will be in contact with the soil or routed in tunnels. This information is necessary in the SSAR for the staff to complete its review. Therefore, it was identified as DFSER Open Items 3.9.2.2-7 and 14.1.3.3.3.9-1 in the DFSER. In Amendment 23 to the SSAR, GE revised Section 3.7.3.12 to respond to these open items. The staff's evaluation of this issue is discussed in Section 3.12.3.9 of this report.

SSAR Section 3.9.2.2.1 states that the minimum cut-off frequency for dynamic analysis of suppression pool hydrodynamic loads is 60 hz, based on a generic study using the missing strain energy method for representative BWR equipment under high-frequency input loadings. This cutoff frequency was previously used in the hydrodynamic analyses for currently operating BWR plants. Because the hydrodynamic load methodology used for the ABWR is the same as that used for the operating BWR plants, the cutoff frequency also is appropriate for the ABWR and acceptable.

On the basis of these discussions and the applicable evaluations in Sections 3.12.3, 3.12.4, and 3.12.5 of this report, the staff concludes that the ABWR plant design meets the relevant guidelines of 10 CFR Part 50, Appendix A, GDC 2, with respect to demonstrating design adequacy of all seismic Category I systems, components, equipment, and their supports to withstand the SSE by meeting the staff positions in RGs 1.61 and 1.92, and the applicable guidelines in SRP Section 3.9.2.

3.9.2.3 Pre-operational Flow-Induced Vibration Analysis and Testing of Reactor Internals

The configuration of reactor internals in the ABWR is different from the configuration in previous BWRs, therefore, the dynamic response of reactor internals to flow-induced vibration must be predicted analytically for the ABWR before final design approval (FDA). In addition, the staff's position is that the first ABWR plant will be tested, in accordance with the guidelines in Positions C.1.1 and C.2 of RG 1.20, "Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing," Revision 2, to evaluate the dynamic responses of reactor internals to steady-state conditions and operational flow transients. These tests are discussed in more detail in the rest of this section.

SSAR Sections 3.9.2.3 through 3.9.2.6 and 3.9.7.1 provide information on vibration testing and analysis of reactor internals. In these sections, the first ABWR plant is referred to as a "prototype plant." In the DFSER, the staff reported that this characterization is inapplicable to evolutionary LWR design certification applications as described in 10 CFR 52.47(b)(1). To address the staff's concern, GE agreed to delete references to "prototype" from its future revision to SSAR Sections 3.9.2.4 and 3.9.7.1. This was identified as DFSER Confirmatory Item 3.9.2.3-1. However, the staff subsequently concluded that the word "prototype," when used in the context of conforming to the reactor internals flow-induced vibration test guidelines of RG 1.20, does not conflict with 10 CFR 52.47(b)(1) and is acceptable. Therefore, DFSER Confirmatory Item 3.9.2.3-1 is withdrawn and resolved.

The dynamic responses of reactor internals to flow-induced vibration must be predicted for the ABWR design before FDA. One of the first steps involved in this prediction is to determine the vibration forcing functions to be used in the system and component dynamic analyses. In SSAR Section 3.9.2.3, GE outlines its approach for determining these forcing functions. Because of the complexity of the flow conditions and structures involved, these loads are not determined by detailed analysis. Instead, a combination of analytical methods and predictions based on data from previously tested reactor internals of a similar design is used. This information on forcing functions then is used in a dynamic modal analysis to predict vibration amplitudes for each dominant response mode of components in the prototype ABWR reactor internals and for interpreting the pre-operational and initial startup test results. Modal stresses are calculated and relationships between vibration measurement sensor responses and peak component stresses for each of the lower modes are obtained. The allowable amplitude in each mode is that which produces a peak stress amplitude of ± 68.95 MPa ($\pm 10{,}000$ psi). This stress is well below the allowable stress amplitude for cycles exceeding 10^6, which is defined in the design fatigue curves for austenitic stainless steels in Appendix I to ASME Code Section III.

By letter dated May 10, 1992, the staff documented a summary of this issue, which was addressed in an audit conducted at GE on February 10 through 12, 1992. As noted in the summary, GE stated that Kashiwazaki Kariwa Unit 6 (K-6), which is currently under construction in Japan, contains reactor internals that are almost identical to those in the ABWR design. During the audit, GE presented a set of documents related to the flow-induced vibration assessment program for the K-6 reactor internals. This information consists of an analysis for vibration prediction, the basis and details of instrumentation for vibration monitoring, specifications for the installation and removal of the monitoring system, and specifications for conducting the pre-operational and startup tests. In addition, GE provided a description of a full-scale, 60-degree flow test of the ABWR RIP system that was conducted in Japan.

For the vibration prediction analysis, GE took a statistical approach similar to that described in the first paragraph above to estimate the range of responses of major reactor pressure vessel internal components in their first few fundamental modes, based on correlations of measured responses of a selected group of existing BWRs with similar configurations. Parameters used in the correlation equations to estimate sample responses consisted of flow, power, stiffness, etc. As previously discussed, the acceptance criteria are more conservative than the applicable criteria in ASME Section III.

In the DFSER, the staff requested that all of the GE commitments and applicable information discussed in the staff's audit summary of May 10, 1992, be included in a future revision to the SSAR. This was DFSER Confirmatory Item 3.9.2.3-2. In the ABWR CDM, dated August 31, 1993, GE submitted its ITAAC for the reactor pressure vessel system, which provided the remaining requested information (i.e., a list of principal plant design parameters and a list of documents that were referenced during the staff's audit). GE's responses in terms of the ABWR design descriptions and ITAAC are evaluated in Section 14.3 of this report. DFSER Confirmatory Item 3.9.2.3-2 is resolved.

In SSAR Section 3.9.2.4, GE states that reactor internals flow-induced vibration measurement and inspection programs are conducted during pre-operational and initial startup testing in accordance with the guidelines of RG 1.20. These tests are conducted in the following three phases:

(1) Pre-operational tests before fuel loading - Steady-state test conditions include balanced recirculation system operation and unbalanced operation over the full range of flow rates up to rated flow. Transient flow conditions include single- and multiple-pump trips from rated flow. This subjects major components to a minimum of 10^6 cycles of vibration at the anticipated dominant response frequency and at the maximum response amplitudes. Vibration measurements are obtained during this test and a close visual inspection of internals will be conducted before and after the test.

(2) Precritical testing with fuel - This vibration measurement series is conducted with the reactor assembly complete but before reactor criticality. Flow conditions include balanced, unbalanced, and transient conditions as for the first test series. The purpose of this series is to verify the anticipated effect of the fuel on the vibration response of internals.

(3) Initial startup testing - Vibration measurements are made during reactor startup at conditions up to 100 percent of rated flow and power. Balanced, unbalanced, and transient conditions of recirculation system operation are evaluated. The primary purpose of this series is to verify the anticipated effect of two-phase flow on the vibration response of internals.

Vibration sensors may include strain gages, displacement sensors (linear variable transducers), and accelerometers. Accelerometers are provided with double-integration signal conditioning to give a displacement output. Sensor locations include the following:

- top of shroud head, lateral acceleration (displacement)
- top of shroud, lateral displacement
- control rod drive housings, bending strain
- incore housings, bending strain
- core flooder internal piping, bending strain

In addition to these components, vibration of the core flooder sparger is measured during pre-operational testing.

Only the dynamic component of strain or displacement is recorded in all vibration measurements. Data are recorded on magnetic tape, and provision is made for selective on-line analysis to verify the overall quality and level of the data. Interpretation of the data requires identification of the dominant vibration modes of each component by the test engineer using frequency, phase, and amplitude information from the component dynamic analyses. Comparison of measured vibration amplitudes to predicted and allowable amplitudes is then made on the basis of the analytically obtained normal mode that best approximates the observed mode.

The purpose of the visual inspections conducted before and following pre-operational testing is to detect evidence of vibration, wear, or loose parts. At the completion of pre-operational testing, the reactor vessel head and the shroud head are removed, the vessel drained, and major components inspected. The inspections cover the shroud, shroud head, core support structures, recirculation internal pumps, peripheral control rod drive, and in-core guide tubes. Access is provided to the reactor lower plenum for these inspections.

The analysis and test program, discussed in SSAR Sections 3.9.2.3, 3.9.2.4, and 3.9.2.6 previously summarized, conforms to applicable portions of SRP Section 3.9.2 and is acceptable. With respect to the availability of test results, the staff understands that, because the design of the reactor internals of K6 in Japan is almost identical to that of the ABWR, the test data discussed will be acquired during the pre-operational flow-induced vibration tests of the K6. GE also indicated that these results will include information specified in Positions C.2.1, C.2.2, C.2.3, and C.2.4 of RG 1.20 and will be submitted to the staff for review as noted in Position C.2.5, "Schedule," of RG 1.20. If the K6 pre-operational flow-induced test data proves insufficient for the RG 1.20 requirements, the COL applicant will develop a test plan to ensure that any additional data is obtained and submitted to the staff. This was DFSER COL Action Item 3.9.2.3-1. SSAR Section 3.9.7.1, "Reactor Internals Vibration Analysis, Measurement, and Inspection Program," states that the first COL applicant will provide, at the time of application, the results of the vibration assessment for the ABWR prototype internals in accordance with the guidelines in RG 1.20. This information will also be submitted to the staff for review and approval. As part of the action item, SSAR Sections 3.9.2.4 and 3.9.7.1 further states that ABWR plants constructed after the first plant that have similar reactor internals will be tested in accordance with the applicable RG 1.20 positions. This is acceptable.

The staff concludes that GE meets GDC 1 and 4 with regard to the reactor internals being designed and tested to quality standards commensurate with the importance of the safety functions being performed and being appropriately protected against dynamic effects (1) by meeting RG 1.20 for the conduct of pre-operational vibration tests and (2) by having a pre-operational vibration program planned for the reactor internals that provides an acceptable basis for verifying the design adequacy of these internals under test loading conditions comparable to those that will be experienced during operation. The combination of predictive analysis, pre-test inspections, tests, and post-test inspections provides adequate assurance that the reactor internals will, during their service life, withstand the flow-induced vibrations of the reactor without loss of structural integrity. The integrity of the reactor internals in service is essential for ensuring the proper positioning of reactor fuel assemblies and the incore instrumentation system to ensure safe operation and shut down of the reactor.

3.9.2.4 Dynamic System Analysis of Reactor Internals Under Faulted Conditions

GE performed dynamic system analyses to confirm the adequacy of the structural design, with no loss of function, of the reactor internals and unbroken loops of the CRD piping to withstand the loads from a LOCA in combination with the SSE. SSAR Section 3.9.2.5 describes the methodology used in developing the dynamic loads resulting from the following four significant faulted events:

- feedwater line break
- MSL break
- SSE
- safety/relief valve discharge

Analyses of other conditions existing during normal operation, abnormal operational transients, and postulated accidents show that the loads affecting safety-related reactor internals from these other conditions are less severe than the loads affecting reactor internals as a result of any of the above four events.

The dynamic systems analyses methodology described in SSAR Section 3.9.2.5 conforms to applicable guidelines in SRP Section 3.9.2 and is acceptable. The staff's evaluations of loading combinations and stress limits for reactor internals are discussed in Sections 3.9.3.1 and 3.9.5, respectively, of this report.

During an audit at GE on February 10 through 12, 1992, which is briefly discussed in Section 3.9.2.3 of this report, the staff reviewed dynamic system analyses of the K6 reactor internals under such postulated accidents as the MSL break at the RPV nozzle. In addition, the staff reviewed the subsequent analyses that evaluated the reactor internals components response to the loads resulting from the system analyses. The calculated pressure differentials during such an event were of a non-dynamic nature (slow variation). GE's assessment of the effects of these loads on the reactor internals is that amplification of loads is unlikely, and dynamic analyses are not required because of the large separation of component structural frequencies from the excitation frequencies. The staff concludes that the methodology implemented by GE in these analyses is consistent with that described in SSAR Section 3.9.2.5 and is acceptable.

The staff concludes that the ABWR dynamic system and component analysis meets the applicable portions of GDC 2 and 4 and SRP Section 3.9.2 with respect to the design of systems and components important to safety to withstand the effects of earthquakes and the appropriate combinations of the effects of normal and postulated accident conditions with the effects of the SSE by a dynamic system analysis which provides an acceptable basis for confirming the structural design adequacy of the reactor internals and unbroken piping loops to withstand the combined dynamic loads of a postulated LOCA and the SSE and the combined loads of a postulated main steamline rupture and the SSE. The analysis provides adequate assurance that the combined stresses and strains in the components of the CRD system and reactor internals will not exceed the allowable design stress and strain limits for the materials of construction and that the resulting deflections or displacements at any structural element of the reactor internals will not distort the reactor internals geometry to the extent that core cooling may be impaired. The staff finds the methods used for component analysis to be compatible with those used for the system analysis. The combination of component and system analyses is acceptable.

3.9.3 ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures

The staff's review under SRP Section 3.9.3 concerns the structural integrity and functional capability of pressure-retaining components, their supports, and core support structures that are designed in accordance with ASME Code, Section III, or earlier industrial standards. The staff reviewed loading combinations and their respective stress limits, the design and installation of pressure-relief devices, and the design and structural integrity of ASME Code, Class 1, 2, and 3 components and component supports. The staff's review acceptance criteria are based on meeting (1) 10 CFR Part 50, Subsection 50.55a and GDC 1 as related to structures and components being designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed; (2) GDC 2 as related to structures and components important to safety being designed to withstand the effects of earthquakes combined with the effects of normal or accident conditions; (3) GDC 4 as related to structures and components important to safety being designed to accommodate the effects of and to be condensible with the environmental conditions of normal and accident conditions; (4) GDC 14 as related to the reactor coolant pressure boundary being designed, fabricated, erected, and tested to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture; and (5) GDC 15 as related to the reactor coolant system being designed with sufficient margin to assure that the design conditions are not exceeded.

3.9.3.1 Loading Combinations, Design Transients, and Stress Limits

GE evaluated all ASME Code, Class 1, 2, and 3 components, component supports, core support components, control rod drive components, and other reactor internals using the load combinations and stress limits given in SSAR Section 3.9.3.1. GE's methodology and selected allowable stress values conform to SRP Section 3.9.3 and are acceptable. Additional staff positions that are not explicitly addressed in SRP Section 3.9.3 are discussed in the rest of this section.

The ASME Code, Section III, requires that the cumulative damage resulting from fatigue be evaluated for all ASME Code Class 1 piping. The cumulative fatigue usage factor should take into consideration all cyclic effects caused by the plant operating transients listed in SSAR Table 3.9-1. As noted in Section 3.9.1 of this report, the ABWR is designed for a 60-year design life. Recent test data to address fatigue concerns indicates that the effects of the reactor environment could significantly reduce the fatigue resistance of certain materials. A comparison of the test data with the Code requirements indicates that the margins in the ASME Code fatigue design curves may be less than originally intended. The DFSER noted that the staff was developing an interim position, which would be available at a later date, to account for the environmental effects in the fatigue design of the affected materials. At this time, the staff is assessing the potential generic implication of this issue on all operating plants. Depending on the severity of the issue, certain actions may be required to generically address this concern. This issue was incorrectly identified as Open Items 3.9.3.1-1 and 14.1.3.3.5.7-1 in the DFSER. The staff's evaluation of this issue is discussed in Section 3.12.5.7 of this report.

SSAR Section 3.9.3.1 states that the design life for the ABWR is 60 years. In response to a staff request, SSAR Sections 3.9.3.1 and 3.9.7.2 were revised to state that COL applicants will identify all ASME Code, Class 2, 3, and QG D components that are subjected to loadings that could result in thermal or dynamic fatigue so severe that the 60-year design life cannot be assured by required Code calculations. If similar designs have not already been evaluated, the COL applicant will either provide an appropriate analysis to demonstrate the required design life, or provide designs to mitigate the magnitude or duration of the cyclic loads (DFSER COL Action Items 3.9.3.1-1 and 14.1.3.3.5.8-1). The resolution of these action items is discussed in Section 3.12.5.8 of this report.

As stated earlier, the staff believes the margins built into the ASME fatigue design curves may not be sufficient to account for variations in the original fatigue test data as a result of various environmental effects. Therefore, the staff's position is that environmental effects should be considered in the fatigue analysis for ASME Code Class 2 and 3 piping. GE should include in its SSAR the proposed approach for accounting for the environmental effects in these fatigue analyses. In the DFSER, this was identified as a part of DFSER Open Item 3.9.3.1-1 for ASME Class 1 components. The resolution of this issue is discussed in Section 3.12.5.7 of this report.

In RAI Q210.42, the staff asked GE to provide the design basis that will be used to ensure the structural integrity of safety-related heating, ventilation, and air conditioning (HVAC) ductwork and its supports. In its response, GE stated that all of these components are designed in accordance with Section 8.2.1 of Chapter 9 of EPRI's "Advanced Light Water Reactor (ALWR) Requirements Document." Section 8.2.1.2.1 of Chapter 9 in Volume II of the Requirements Document states that all safety-related HVAC systems in the ALWR will be designed to withstand an SSE and will be capable of accomplishing their intended functions assuming a single failure of an active component and a loss of preferred power (LOPP). Section 8.2.1.2.7 of Chapter 9 in Volume II of the Requirements Document specifies that the HVAC components and supports will be designed, constructed, and installed in accordance with ANSI/ASME AG-1-1988, and ANSI/ASME N509. Portions of ANSI/ASME AG-1-1988, including rules for the design of HVAC ductwork, are still being prepared. Therefore, the staff has not yet fully endorsed this standard. Until this standard is fully endorsed, the staff's interim position is that Article AA-4000 in the 1988 revision of the AG-1 Standard provides acceptable minimum design requirements for the structural design of HVAC equipment and supports. For HVAC ductwork, in Revision 2 to RG 1.52, "Design, Testing, and Maintenance Criteria for Post Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," the staff recommends that ductwork should be designed, constructed, and tested in accordance with the provisions of Section 5.10 of ANSI N509.

The staff concludes that the design criteria required in Volume II, Chapter 9, Section 8.2.1, of the Requirements Document are consistent with current staff positions and provide an acceptable minimum design basis for ensuring that HVAC components and supports will withstand the most adverse combination of loading events without loss of structural integrity.

In accordance with NRC Bulletin 88-08, the staff is requesting that licensees and applicants review systems connected to the reactor coolant system (RCS) to determine if any sections of such piping that cannot be isolated can be subjected to stresses from temperature stratification or temperature oscillations that could be induced by leaking valves. In RAI Q210.50, the staff requested that GE review the ABWR design to determine if this phenomenon could occur. In response to this request, GE stated that the ABWR systems connected directly to the RCS or the RPV are the nuclear boiler system and the emergency core cooling systems. In the nuclear boiler system, GE reviewed the feedwater subsystem that supplies makeup water to the RPV to confirm that design requirements for temperature stratifications of feedwater piping were satisfactorily defined in system specifications and piping cycle diagrams. In the design of the ECCS, both the residual heat removal (RHR) system and high-pressure core flooder (HPCF) have piping that are directly connected to the RPV. In the unisolable sections of RHR piping, leaking toward the RPV cannot occur because the pressure will always be higher on the reactor side during normal plant operation when the upstream pumps are not operating. In the HPCF system design, the only unisolable piping connected to the RPV is the section of pipe between the reactor nozzle and the upstream isolation check valve. Cold water in this system will be upstream of the injection valve (gate valve) that is outside the primary containment. The region upstream of the injection valve will operate at a pressure lower than reactor pressure except when the HPCF safety function is required. Therefore, cold water will not flow to the unisolable pipe section and stratification will not be a problem in the HPCF system. The staff concludes that the ABWR design adequately addresses the potential problems described in NRC Bulletin 88-08, Supplements 1 and 2. However, GE did not address the potential problem described in Supplement 3 to Bulletin 88-08. The staff's evaluation of GE's response to this issue is discussed in Section 3.12.5.9 of this report. Further staff evaluations of thermal stratification are discussed as follows.

Thermal stratification is a phenomenon that can occur in long runs of horizontal piping when two streams of fluid at different temperatures flow in separate layers without appreciable mixing. Under such stratified flow conditions, the top of the pipe may be at a much higher temperature than the bottom. This thermal gradient produces pipe deflections, support loads, pipe bending stresses, and local stresses that may not have been accounted for in the original piping design. The effects of thermal stratification have been observed in both BWR and PWR feedwater piping as discussed in NRC Information Notice (IN) 84-87, and NRC IN 91-38.

During an audit conducted at GE offices in San Jose, California, on March 23 through 26, 1992, the staff asked GE to explain and demonstrate how the thermal stratification phenomenon was considered in the ABWR piping design. In response, GE stated that thermal stratification is considered as a normal design load in the ASME Code 1 stress and fatigue evaluation of the feedwater piping. In addition, GE indicated that the feedwater line will be analyzed for two thermal stratification load cases: (1) thermal stratification in the piping at the RPV nozzle and (2) thermal stratification in the feedwater header piping. The loads will be included in the piping fatigue analysis and in the evaluations of the head fitting and RPV nozzles. The temperature differences and locations for the stratification loads are defined in the feedwater piping pressure/temperature cycle diagrams. This was DFSER COL Action Item 14.1.3.3.5.10-1. The resolution of this action item is discussed in Section 3.12.5.10 of this report.

The staff reviewed and discussed the thermal stratification analysis methodology with the cognizant GE engineers and found it acceptable with the exception of an apparent discrepancy in load application. GE defined the stratified temperature profile in the pipe cross section as a constant hot temperature in the top half and cold temperature in the bottom half with a step change in the temperature at the centerline. However, in the pipe stress analysis, a linear top-to-bottom temperature profile was applied. The linear temperature profile provides lower bending moments and stresses than the step change profile. The staff asked GE to justify (1) the adequacy of the piping analysis load input and (2) the omission of the high-cycle fatigue effects of thermal striping from the analysis. In addition, the staff asked GE to provide additional justification for their methodology including test information to support their thermal stratification load definition (DFSER Open Items 3.9.3.1-2 and 14.1.3.3.5.10-1). The staff's evaluation of this issue is discussed in Section 3.12.5.10 of this report.

In the DFSER, the staff reported that GE had not provided information to the staff that would establish a minimum temperature at which an explicit piping thermal expansion analysis would be required. Unless GE provides this information in the SSAR, the staff would have required that thermal analyses be performed for all temperature conditions above ambient. This was DFSER Open Items 3.9.3.1-3 and 14.1.3.3.5.18-1. The staff's evaluation of this issue is discussed in Section 3.12.5.18 of this report.

In a letter dated January 28, 1993, GE revised Note 6 to SSAR Table 3.9-2 to state that all ASME Code Class 1, 2, and 3 piping systems that are essential for safe shutdown under the postulated events are designed to meet the requirements of NUREG-1367, "Functional Capability of

Piping Systems," dated November 1992. This report contains methodology acceptable to the staff for ensuring the functional capability of essential piping systems. GE has included this information in Table 3.9-2 of the SSAR.

The ASME Code requires that a design specification be prepared for Class 1, 2, and 3 components such as pumps, valves, and piping systems. The design specification is intended to become a principal document governing the design and construction of these components and should specify loading combinations, design data, and other design data inputs. The code also requires a design report for ASME Code, Class 1, 2, and 3 piping and components. In the SSAR, GE committed to construct all safety-related components, such as vessels, pumps, valves and piping systems, to applicable requirements of the ASME Code, Section III. During its review of the SSAR, the staff also reviewed selected documents related to design specifications and design reports. Those documents were not specifically for the ABWR, but were provided by GE and reviewed by the staff as a demonstration of how design specifications and design reports will be prepared for ABWR plants. The staff determined that the demonstration documents, with modifications, would meet code requirements. However, because the documents were not specifically for the ABWR, they would have to be modified before the staff can conclude that the design specification and design report requirements in ASME Code, Section III, Subsection NCA have been met. In the DFSER, the staff noted that the plant-specific design documentation should be prepared and made available for staff review by COL applicants referencing the ABWR design. This was DFSER COL Action Items 3.9.3.1-2 and 14.1.3.3.2.3-1. SSAR Section 3.9.7.4 states that the COL applicant will make design specifications and design reports required by the ASME Code for vessels, pumps, valves, and piping systems available to the staff for the purpose of audit. This is acceptable.

3.9.3.1.1 Intersystem LOCA Design for Piping Systems

In SECY-90-016, the staff recommended that the Commission approve the staff's resolution of the intersystem loss-of-coolant accident (ISLOCA) issue for ALW reactor plants by requiring that low-pressure piping systems that interface with the RCPB be designed to withstand full RCS pressure to the extent practicable. As noted, in its June 26, 1990, SRM, the Commission approved the staff's recommendation provided that all elements of the low-pressure system are considered. In the DFSER, the staff noted that GE had not yet submitted the details of the piping design for the full RCS pressure. This was DFSER Open Item 14.1.3.3.5.19-1. Subsequently, GE provided its implementation of the issue resolution for the ABWR in SSAR Section 3.9.3.1.

As an applicable regulation described in Section 1.6 of this report, the staff proposed that

> the standard design must minimize the effects of intersystem loss-of-coolant accidents by designing low-pressure piping systems that interface with the RCPB to withstand full reactor coolant system pressure to the extent practical.

In Section 20.2.19 of this report, under Issue 105, the staff evaluated GE's approach, in terms of the practicality for systems, components, and equipment, for implementing the ISLOCA resolution for the ABWR. In the following, the staff evaluated the minimum pressure for which low-pressure systems should be designed to ensure reasonable protection against burst failure should the low-pressure system be subjected to full RCS pressure. In establishing the minimum design pressure, the following goals were used as the basis for selection:

(1) The likelihood of rupture (burst) of the pressure boundary is low, based on a goal of approximately 10 percent failure probability for rupture;

(2) The likelihood of intolerable leakage of flange joints or valve bonnets is reasonably low although some leakage might occur;

(3) Some piping components might undergo gross yielding and permanent deformation.

Low-Pressure Piping Design

To achieve these objectives, the staff evaluated, first, on a qualitative basis, several possible ratios of the low-pressure system design pressure (P_d) to the RCS normal operating pressure (P_v) to establish the margins on burst and yield of the piping. The results of the staff's evaluation are depicted in Table 3.2 for typical carbon steel (A106 Grade B) and stainless steel (SA312 Type 304) materials and are then discussed for three ratios of the design pressure to the reactor vessel pressure (P_d/P_v). A margin of 1.0 or less represents the condition where burst or yielding is likely to occur. The higher the margin, the less likely burst or yielding is to occur. The low-pressure piping systems are assumed to be designed to the rules of the ASME Boiler and Pressure Vessel Code, Section III, Subarticle NC/ND-3600 for Class 2 and 3 piping systems.

Table 3.2 Margins for straight pipe

Material	Temp °C (°F)	P_d/P_v	S MPa (ksi)	S_v MPa (ksi)	S_u MPa (ksi)	S_y MPa (ksi)	Margins on Burst	Margins on Yield
SA 106 Grade B	37.8 (100)	1/2	103.4 (15)	206.8 (30)	413.7 (60)	241.3 (35.0)	2.00	1.34
	260 (500)	1/2	103.4 (15)	206.8 (30)	413.7 (60)	195.1 (28.3)	2.00	1.08
	37.8 (100)	1/3	103.4 (15)	310.3 (45)	413.7 (60)	241.3 (35.0)	1.33	0.89
	260 (500)	1/3	103.4 (15)	310.3 (45)	413.7 (60)	195.1 (28.3)	1.33	0.72
	37.8 (100)	1/4	103.4 (15)	413.7 (60)	413.7 (60)	241.3 (35.0)	1.00	0.67
	260 (500)	1/4	103.4 (15)	413.7 (60)	413.7 (60)	195.1 (28.3)	1.00	0.54
SA312 Type 304	37.8 (100)	1/2	129.6 (18.8)	258.6 (37.5)	517.1 (75.0)	206.8 (30.0)	1.70	0.92
	260 (500)	1/2	109.6 (15.9)	219.3 (31.8)	437.8 (63.5)	133.8 (19.4)	1.70	0.70
	37.8 (100)	1/3	129.6 (18.8)	388.2 (56.3)	517.1 (75.0)	206.8 (30.0)	1.13	0.61
	260 (500)	1/3	109.6 (15.9)	328.9 (47.7)	437.8 (63.5)	133.8 (19.4)	1.13	0.47
	37.8 (100)	1/4	129.6 (18.8)	517.1 (75.0)	517.1 (75.0)	206.8 (30.0)	0.85	0.46
	260 (500)	1/4	109.6 (15.9)	438.5 (63.6)	437.8 (63.5)	133.8 (19.4)	0.85	0.35

S = allowable stress per ASME Code, Section III for Class 2 piping

S_v = hoop stress at $P = P_v$
= $S/(P_d/P_v)$

S_u = ultimate tensile strength; from Section III, Table I-3.1 and I-3.2

S_y = yield strength; from Section III, Table I-2.1 and I-2.2

Margin on Burst Pressure = $F \times S_u \times (P_d/P_v)/S$
where F = 1.00 for SA106 Grade B
F = 0.85 for SA312 Type 304

Margin on Yield Pressure = $1.15 \times S_y \times (P_d/P_v)/S$

Piping Integrity at $P_d/P_v = 1/2$ (ASME Code Service Level D)

When P_d/P_v is equal to one-half, the margins on burst and yield are equivalent to approximately those of the ASME Boiler and Pressure Vessel Code, Section III Service Level D condition. For carbon steel pipe, this ratio will provide a margin of 2.0 on burst and 1.08 on yield for a pipe at 500 °F. For stainless steel piping, a ratio of one-half will provide a sufficient margin on burst (1.7). However, a small amount of yielding is likely to occur with a margin of 0.70 at 500 °F. No leakage of the pressure boundary is likely to occur at P_d/P_v equal to one-half.

As a result, a ratio of one-half will ensure the pressure integrity of the low-pressure piping system with ample margin.

Piping Integrity at $P_d/P_v = 1/3$

When the ratio, P_d/P_v, is reduced to one-third, the margins for carbon steel piping are lowered to 1.33 and 0.72 for burst and yield at 260 °C (500 °F), respectively. For stainless steel piping, the margins are 1.13 and 0.47 for burst and yield at 260 °C (500 °F), respectively. At these margins, it is expected that burst failure will not occur in either carbon steel or stainless steel piping. However, significant amount of yielding might occur in stainless steel piping at all temperatures and in carbon steel piping at 260 °C (500 °F). Where the carbon steel piping is at a lower temperature, some yielding might occur although to a lesser extent. The consequence of significant pipe yielding (without bursting) is that gross, permanent distortion might occur in the piping components thereby resulting in some leakage through flanges, or valve bonnets. However, it is not expected that such leakage would be uncontrollable or intolerable.

In summary, a ratio of one-third will ensure the pressure boundary of the low-pressure piping although a significant amount of pipe yielding and some leakage through flanges and valve bonnets is likely to occur.

Piping Integrity at $P_d/P_v = 1/4$

At P_d/P_v equal to one-fourth, the pressure integrity of carbon steel piping becomes questionable, and for stainless steel piping, it is likely that burst failure will occur. Prior to bursting, the piping system would undergo gross plastic deformation, experience a significant amount of leakage at flanges, valve bonnets, and pump seals, and possibly lose some pipe supports due to the radial expansion of the pipe.

Therefore, at P_d/P_v equal to one-fourth, the ability of the low-pressure piping system to withstand full RCS pressure is questionable for carbon steel piping and unlikely for stainless steel piping systems.

The staff further evaluated, on a quantitative basis, the survival probabilities of the low-pressure piping at various design pressures using the methodology described in NUREG/CR-5603, "Pressure-Dependent Fragilities for Piping Components," dated October 1990. Calculations were performed by Idaho National Engineering Laboratory (INEL) under contract with the NRC's Office of Nuclear Regulatory Research.

The INEL calculations led to results similar to the qualitative conclusions discussed above. A temperature of 176.7 °C (350 °F) was used in the calculations of the following survival probabilities. Using a temperature of 260 °C (500 °F), the survival probabilities decreases about 2 to 5 percent for the different materials and design pressures.

For carbon steel piping (SA-106 Grade B material) subjected to a pressure of 2758 kPa (400 psig) (or approximately $P_d/P_v = 0.4$), the survival probability is 99 percent. For stainless steel piping (Type 304 material), the survival probability at 2758 kPa (400 psig) (or approximately $P_d/P_v = 0.4$), was found to be about 87 percent.

Using these results, the staff finds that the GE-proposed design pressure of 0.4 times the CRD pressure 2827 kPa (410 psig) nearly achieves the staff's goal of a 90-percent survival probability under ISLOCA conditions and provides a sound basis for establishing the design pressure of low-pressure systems interfacing with the RCS pressure boundary. The minimum design pressure for the low-pressure piping systems will be identified as a certified design commitment for the ABWR.

Note, however, that the survival probabilities are based on the minimum wall thickness as calculated using Equation (3) in the ASME Boiler and Pressure Vessel Code, Section III, Subarticle NC/ND-3640. The wall thickness thus calculated does not account for manufacturing tolerances or the use of the next heavier commercial wall thickness available, which would increase the piping wall thickness and substantially increase the survival probability as well. In Section 3.9.3.1 of the SSAR, GE stated that the pipe wall thickness will be greater than or equal to schedule 40. When standard weight piping wall thicknesses are used (i.e., schedule 40 pipe up to and including 254-mm (10-in.) nominal pipe size), the survival probabilities at a design pressure 2758 kPa (400 psig) are above 99 percent for both carbon and stainless steel piping.

Therefore, for the ABWR, the staff concludes that the use of standard weight piping will provide a bounding design for the ISLOCA condition.

Valves in Low-Pressure Systems

For the valves in the low-pressure piping systems (excluding the pressure isolation valves), the selection of the valve class rating is a primary factor for designing against full RCS pressure. For example, ANSI B16.34 valves are supposed to be shop-tested to 1.5 times their 37.8 °C (100 °F) rated pressure. This would mean for a Class 300 A216 WCB (cast carbon steel) valve, the test pressure is 1.5 × 740 = 7,653 kPa (1,110 psig). For Class 150, the valve test pressure is 1.5 × 285 = 2,948 kPa (427.5 psig).

Clearly, the Class 300 valve, which is tested to a pressure of 7,653 kPa (1,110 psig), would be expected to withstand an RCS normal operating pressure of 7,171 kPa (1,040 psia) (or 7,067 kPa (1,025 psig)). However, it should not be assumed that the valve in the low-pressure system would be able to operate with this full RCS pressure across the disk.

Therefore, the staff finds that a Class 300 valve is adequate for ensuring the pressure of the low-pressure piping system under full RCS pressure (i.e., 7,067 kPa (1,025 psig)), but no credit should be taken to consider these valves operable under such conditions without further justification.

Other Components in Low-Pressure Systems

For other components in the low-pressure systems, such as pumps, tanks, heat exchangers, flanges, and instrument lines, the staff finds that establishing an appropriate safety factor involves several complicating factors related to the individual component design. These factors include requirements for shop hydrotests, the method to determine the pressure class rating of the component, the specific material used for bolting and the bolt tension applied, or whether the component is qualified by test or analysis.

The remaining components in the low-pressure systems will be designed to a design pressure of 0.4 times the normal operating RCS pressure (i.e., 2,827 kPa (410 psig)). The staff finds that the margins to burst for these remaining components are at least equivalent to that of the piping at its minimum wall thickness since these components typically have wall thicknesses greater than that of the pipe minimum wall thickness.

ISLOCA Conclusion

The staff finds for the ABWR low-pressure piping systems that interface with the RCS pressure boundary, that using a design pressure equal to 0.4 times the normal operating RCS pressure of 7,067 kPa (1,025 psig) (i.e., 2,827 kPa (410 psig)) and using a minimum wall thickness of the low-pressure piping no less than schedule 40 provide an adequate basis for assuring that these systems can withstand full reactor pressure and thus meet the Commission-approved staff recommendations in SECY-90-016 and its applicable regulation for designing against intersystem LOCAs. The piping design is in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Subarticle NC/ND-3600. Furthermore, the staff will continue to require periodic surveillance and leak rate testing of the pressure isolation valves per technical specification (TS) requirements as a part of the inservice testing program.

As stated in SECY-90-016, for those low-pressure systems for which designing to withstand full reactor pressure is not practical, the design provides (1) the capability for leak testing of the pressure isolation valves, (2) valve position indication that is available in the control room 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.

Using these design guidelines, the staff concludes that--

(1) the likelihood of the low-pressure piping rupturing under full RCS pressure is low,

(2) the likelihood of intolerable leakage is low under ISLOCA conditions although some leakage may occur at flanges and valve bonnets, and

(3) some piping components might undergo gross yielding and permanent deformation under ISLOCA conditions.

SSAR Section 3.9.3.1 includes these design guidelines. Therefore, the staff concludes that there is reasonable assurance that the low-pressure piping systems interfacing with the RCPB are structurally capable of withstanding the consequences of an intersystem loss-of-coolant accident. The staff further concludes that the ABWR design meets the Commission-approved staff recommendation for ISLOCAs, as an applicable regulation as described in Section 1.6 of this report.

3.9.3.1.2 Conclusion

On the basis of the evaluation in Section 3.9.3.1 of this report, the staff concludes that GE meets 10 CFR 50.55a and GDC 1, 2, and 4 with respect to the design and service load combinations and associated stress and deformation limits specified for ASME Code, Class 1, 2, and 3 components by ensuring that systems and components are designed to quality standards commensurate with their importance to safety and that these systems can accommodate the effects of such postulated events as LOCAs and the dynamic effects resulting from earthquakes. The specified design and service combinations of loadings as applied to ASME Code Class 1, 2, and 3 pressure retaining components in systems designed to meet seismic Category I standards provide assurance that, in the event of an earthquake affecting the site or other service loadings owing to postulated events or system operating transients, the resulting combined stresses imposed on system components will not exceed allowable stress and strain limits for the materials of construction. Limiting the stresses under such loading combinations provides an acceptable basis for the design of system components to withstand the most adverse combination of loading events without loss of structural integrity.

3.9.3.2 Design and Installation of Pressure-Relief Devices

The staff reviewed SSAR Section 3.9.3.3 with regard to the design, installation, and testing criteria applicable to the mounting of pressure-relief devices used for the overpressure protection of ASME Code, Class 1, 2, and 3 components. This review, conducted in accordance with SRP Section 3.9.3, included evaluation of the applicable loading combinations and stress criteria. The review extended to consideration of the means provided to accommodate the rapidly applied reaction force when a safety/relief valve (SRV) opens and the transient fluid-induced loads applied to the piping downstream of an SRV in a closed discharge piping system. The information in SSAR Section 3.9.3.3 meets the applicable guidelines of SRP Section 3.9.3 and is acceptable.

In accordance with TMI Action Item II.D.1 of NUREG-0737, pressurized water reactor (PWR) and BWR licensees and applicants are required to conduct testing to qualify the RCS SRVs and associated piping and supports under expected operating conditions for design-basis transients and accidents. GE's response to Item II.D.1 is briefly discussed in SSAR, Appendix 1A, Section 1A.2.9. This section states that the SRV models have been tested under ABWR steam discharge conditions. It further states that if the ABWR design should contain any SRVs or discharge piping that is not similar to those that have been tested, the valves will be tested in accordance with Item II.D.1. This is acceptable.

In performing the hydraulic transient piping analyses associated with the SRV discharge, GE assumed a minimum rise time of 20 msec. Rise times faster than this value could result in higher loads than analytically predicted. The assumed rise time is based on past SRV designs and existing test data. Contingent upon the action described above to retest the SRVs if the COL applicant should purchase any SRV or install its SRV piping in a configuration that is not similar to those that have been tested, this approach is acceptable. The COL applicant should confirm that any SRVs or discharge piping installed in the ABWR standard plant that is not similar to those that have been tested will be tested in accordance with TMI Action Item II.D.1. This was DFSER COL Action Items 3.9.3.2-1 and 14.1.3.3.5.11-1. The resolution of these action items is discussed in Section 3.12.5.11 of this report.

On the basis of this evaluation, which states that the criteria in SSAR Section 3.9.3.3 as related to the design, installation, and testing of ASME Code, Class 1, 2, and 3 SRV mounting meet the applicable guidelines of SRP Section 3.9.3, the staff concludes that GE meets 10 CFR 50.55a and GDC 1, 2, and 4 by ensuring that SRVs and their installations are designed to standards that are commensurate with their safety functions and that they will accommodate the effects of discharge caused by normal operation as well as postulated events such as LOCAs and the dynamic effects resulting from the SSE. GE also meets GDC 14 and 15 with regard to ensuring that the RCPB design limits for normal operation, including anticipated operational occurrences, will not be exceeded. The criteria used by GE in the design and installation of ASME Code, Class 1, 2, and 3 SRVs provide adequate assurance that, under discharging conditions, the resulting stresses will not exceed allowable stress and strain limits for the materials of construction. Limiting the stresses under the loading combinations associated with the actuation of these pressure-relief devices provides a conservative basis for the design and installation of the devices for ensuring that the devices will withstand these loads without loss of structural integrity or impairment of the overpressure-protection function.

3.9.3.3 Component Supports

The staff reviewed SSAR Sections 3.9.3.4 and 3.9.3.5 with regard to the methodology used in the design of ASME Code Class 1, 2, and 3 component supports. The review included an assessment of the design and structural integrity of the supports. It addressed three types of supports: plate and shell, linear, and component standard types. All ASME Code Class 1, 2, and 3 component supports for the ABWR plant are constructed in accordance with ASME Code, Section III, Subsection NF. In addition, the SSAR states that the design is augmented by the application of Code Case N-476, Supplement 89.1, which governs the design of single-angle members. If eccentric loads or other torsional loads are not accommodated by designing the load to act through the shear center, analyses are performed in accordance with such torsional analysis methods such as "Torsional Analysis of Steel Members," AISC Publication T114-2/83. The staff position is that Subsection NF is an acceptable code for the design of piping supports. However, the rules must be augmented by guidelines acceptable to the staff governing the design of single-angle members of supports and the methodology used to accommodate torsional loads. At this time, although Code Case N-476 has not been endorsed by the staff in RG 1.84, the staff finds that it provides adequate design rules for the single-angle members. For torsional analysis of steel members, the staff finds that GE-proposed documents provide sufficient technical guidelines to perform a torsional analyses of steel members and are acceptable.

SSAR Section 3.9.3.4 defines the jurisdictional boundaries between pipe supports and such interface attachment points as structural steel in accordance with the ASME Code, Section III, Subsection NF, 1989 Edition. The staff's review of the jurisdictional boundaries described in the 1989 Edition finds that they are sufficiently defined to ensure a clear division between the pipe support and the structural steel and are acceptable.

SSAR Section 3.9.3.4.1 states that the loading combinations for the design of piping supports correspond to those used for the design of the supported pipe. As discussed in Section 3.9.3.1 of this report, the staff concludes that the loading combinations used for the supported pipe are consistent with SRP 3.9.3 and are acceptable. The stress limits for pipe supports are in accordance with the ASME Code, Section III, Subsection NF and Appendix F. The supports are generally designed or qualified by the load rating method as described in paragraph NF-3260 or by the stress limits specified in paragraph NF-3231. These methods and limits as specified in the 1989 Edition of the ASME Code, Section III, are acceptable.

SSAR Section 3.9.3.4 provides design criteria for the design of pipe supports using supplementary steel. The building structure component supports are designed in accordance with AISC, "Specification for the Design, Fabrication and Erection of Structural Steel for Buildings." The use of this specification is standard industry practice and has been proven to provide adequate design guidelines for the design of structural steel for use as pipe supports and is acceptable.

SSAR Section 3.9.3.4 of earlier amendments stated that the concrete anchor bolts, which would be used for pipe support base plates, will be designed to the applicable factors of safety defined in NRC Office of Inspection and Enforcement (IE) Bulletin 79-02, Revision 1. Loading combinations for component supports are discussed in Section 3.9.3.1 of this report. In general, the factors of safety for anchor bolts are acceptable. However, in the DFSER, the staff noted that GE had not discussed the use of specific types of anchor bolts to be used for pipe support base plates in the ABWR plant. For example, under-cut type anchor bolts behave in a ductile manner but the staff's position is that the safety factors in IE Bulletin 79-02, Revision 2, are still applicable unless GE provides justification for alternative safety factors. Therefore, the use of safety factors for anchor bolts other than those provided in IE Bulletin 79-02 must be justified and submitted to the staff for review and approval before their use. This was DFSER COL Action Items 3.9.3.3-1 and 14.1.3.3.6.4-1. In addition, the staff reported that of the type of concrete anchor bolt used for piping supports, the action item in IE Bulletin 79-02 relative to pipe support base-plate flexibility must be implemented. This was DFSER COL Action Item 14.1.3.3.6.4-2. The resolution of these COL action items is discussed in Section 3.12.6.4 of this report.

In the DFSER, the staff noted that GE had not provided the staff with any details about the specific analysis methods or procedures to be used for the ABWR pipe support design. For the staff to complete its review, GE was requested to include the following additional details of the pipe support design in the SSAR:

(1) information that addresses the types of snubbers and their characteristics (as delineated in SRP 3.9.3, Section II.3b) to be used in the ABWR standard plant

(2) information that addresses the use of seismic restraints other than snubbers (i.e., special engineered pipe supports such as energy absorbers and limit stops, and their modeling assumptions)

(3) information that addresses the pipe support stiffness values and support deflection limits used in the piping analyses

(4) information that addresses how the seismic excitation of the pipe supports (especially large frame-type structures) are to be considered in the design of the pipe support anchorage

(5) information that addresses the hot and cold gaps to be used between the pipe and the box-frame-type of support and the coefficient of friction to be used for considering friction forces between the pipes and the steel frames

(6) criteria that will ensure that the maximum deflections of the piping at support locations for static and dynamic loadings are within an allowable limit to preclude failure of the pipe supports and hangers

In the DFSER, all of these items constituted Open Items 3.9.3.3-1 and 14.1.3.3.6-1. The staff's evaluations of these open items are discussed in Sections 3.12.6.5, 3.12.6.6, 3.12.6.7, 3.12.6.8, 3.12.6.10, 3.12.6.11, and 3.12.6.13 of this report.

The staff noted in the DFSER that GE had not provided any information on the design criteria for the structural design of instrumentation line supports. The was staff requested that this information be included in the SSAR. This was DFSER Open Item 3.9.3.3-2 and part of DFSER Open Item 14.1.3.3.6-1. SSAR Subsections 3.7.3.8.1.9 and 3.9.3.4.1 contain GE's response to these open items. The staff's evaluation of this response is discussed in Section 3.12.6.12 of this report.

The industry has taken the position that ANS/AISC N-690 is useful in the design of instrumentation sensing line supports. Its use would have the effect of reducing the QA recordkeeping requirements and code stamping required by Subsection NF of ASME Section III. The staff's position on this issue is that for construction of ASME component supports, ANS/AISC N-690 alone is not an acceptable standard; ASME Code, Section III, Subsection NF, should be used. However, the staff is currently participating in the ASME effort to incorporate N-690 into Subsection NF. Subsequent to a staff-endorsed version of NF incorporating N-690, Subsection NF will also specify the rules acceptable to the staff for construction of ASME Class supports. In the DFSER, the staff stated that when this staff-approved version is available, the COL applicant seeking to use it may submit a request to the staff for approval on a plant-specific basis. This was DFSER COL Action Item 14.1.3.3.6.12-1. The staff's evaluation of this issue is discussed in Section 3.12.6.12 of this report.

On the basis of the evaluation in Section 3.9.3.3, supplemented by the evaluations in applicable portions of Section 3.12.6, of this report, the staff concludes that GE meets the requirements of 10 CFR 50.55a and GDC 1, 2, and 4 with regard to the design and service load combinations and associated stress and deformation limits specified for ASME Code, Class 1, 2, and 3 component supports by insuring that component supports are designed to quality standards commensurate with their importance to safety, and that these supports can accommodate the effects of normal operation as well as postulated events such as LOCAs and the dynamic effects resulting from the safe shutdown earthquake. The combination of loadings (including system operating transients) considered for each component support within a system, including the designation of the appropriate service stress limit for each loading combination, has met the positions and criteria of RGs 1.124 and 1.130 and are in accordance with NUREG-0484. The specified design and service loading combinations used for the design of ASME Code, Class 1, 2, and 3 component supports in systems classified as seismic Category I provide assurance that in the event of an earthquake or other service loadings due to postulated events or system operating transients, the resulting combined stresses imposed on system components will not exceed allowable stress and strain limits for the materials of construction. Limiting the stresses under such loading combinations provides a conservative design basis to assure that support components can withstand the most adverse combination of loading events without loss of structural integrity.

The staff's evaluation of Class CS components is given in Section 3.9.5 of this report.

3.9.4 Control Rod Drive Systems

The staff's review under SRP Section 3.9.4 included the control rod drive system (CRDS) up to its interface with the control rods. Those components of the CRDS that are part of the primary pressure boundary are classified as SC 1, QG A, and are designed according to ASME Code, Section III, Class 1 requirements and to the quality assurance requirements of 10 CFR Part 50, Appendix B. The CRDS will be capable of reliably controlling reactivity changes either under conditions of anticipated normal plant operational occurrences or under postulated accident conditions. The CRDS in the ABWR design consists of fine motion control rod drive mechanisms and the control rod drive hydraulic system. The staff reviewed the information in SSAR Section 3.9.4 related to the criteria used to ensure the structural integrity of this system during normal operation and under accident conditions. These criteria conform to SRP Section 3.9.4 and are acceptable. Loading combinations for the CRDS are discussed in Section 3.9.3.1 of this report. The functional design and testing of these systems is discussed in Section 4.6 of this report.

The staff's review acceptance criteria are based on meeting (1) GDC 1 and 10 CFR Part 50, Subsection 50.55a requiring that the CRDS be designed to quality standard commensurate with the importance of the safety functions to be performed; (2) GDC 2 requiring that the CRDS be designed to withstand the effects of an earthquake without loss of capability to perform its safety functions; and (3) GDC 14 requiring that the RCPB portion of the CRDS be designed, constructed, and tested for the extremely low probability of leakage or gross rupture.

The staff concludes that the design of the control rod drive system is acceptable for the ABWR and meets GDC 1, 2, and 14, and 10 CFR 50.55a. As stated in the first paragraph in this section, by designing the CRDS up to its interface with the control rods to acceptable loading combinations of normal operation and accident conditions using ASME Class 1 and 10 CFR Part 50, Appendix B, requirements, GE has assured the structural integrity of the CRDS. Therefore, GE meets GDC 1 and 10 CFR 50.55a with regard to designing components important to safety to quality standards commensurate with the importance of the safety functions to be performed. In addition, GE meets GDC 2 and 14 with regard to designing the control rod drive system to withstand the effects of earthquakes and anticipated normal operation occurrences with adequate margins to ensure its structural integrity and functional capability and with an extremely low probability of leakage or gross rupture of the RCPB. The staff's evaluations of the specified design transients, design and service loadings, and combinations of loads, are discussed in Sections 3.9.1 and 3.9.3.1 of this report. By limiting the stresses and deformations of the CRDS under such loading combinations, the design conforms to the appropriate guidelines in SRP Sections 3.9.3 and 3.9.4.

3.9.5 Reactor Pressure Vessel Internals

The staff reviewed, in accordance with SRP Section 3.9.5, the load combinations, allowable stress and deformation limits, and other criteria used in the design of the reactor internals. The staff's review acceptance criteria are based on meeting (1) GDC 1 and 10 CFR Part 50, Subsection 50.55a requiring that the reactor internals shall be designed to quality standards commensurate with the importance of the safety functions to be performed; (2) GDC 2 requiring that the reactor internals shall be designed to withstand the effects of earthquakes without loss of capability to perform its safety functions; (3) GDC 4 requiring that reactor internals shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operations, maintenance, testing, and postulated LOCA; and (4) GDC 10 requiring that reactor internals shall be designed with adequate margins to assure that specified acceptable fuel design limits are not exceeded during anticipated normal operational occurrences. SSAR Section 3.9.5.3.5 states that the core support structures for the ABWR are designed and constructed in accordance with ASME Code, Section III, Subsection NG. In accordance with Subsection NG-1100, this means that the manufacture and installation of the ABWR core support structures are in accordance with the NG rules required for materials, design, examination, and preparation of reports. This conforms to SRP Section 3.9.5 and is acceptable.

SSAR Section 3.9.5.3.6 gives the design bases for safety class reactor internals other than the core support structures. The design criteria, loading conditions and analyses that provide the basis for the design of these components meet the guidelines of ASME Code, Section III, Subsection NG-3000. These components are constructed so as not to adversely affect the integrity of the core support structures as required by ASME Code, Section III, Subsection NG-1122. These criteria conform to SRP Section 3.9.5 and are acceptable.

In accordance with SSAR Table 3.2-1, the core support structures and all other safety-related reactor internals are designed as Safety Class 2 components and to the quality assurance requirements of 10 CFR Part 50, Appendix B. In addition, as discussed in Section 3.9.1, "Special Topics for Mechanical Components," Section 3.9.2.4, "Dynamic System Analysis of Reactor Internals Under Faulted Conditions," and Section 3.9.3.1, "Loading Combinations, Design Transients, and Stress Limits," of this report, the SSAR contains acceptable criteria for the design of all safety-related reactor internals under normal, upset, emergency, and faulted loading conditions. Implementation of these criteria to the design of the reactor internal structures and components provides reasonable assurance that, in the event of an earthquake or of a system transient during normal plant operation, the resulting deflections and associated stresses imposed on these structures and components will not exceed allowable stresses and deformations under such loading combinations. This provides an acceptable design basis for ensuring that these structures and components will withstand the most adverse loading events that were postulated to occur during their service lifetime without loss of structural integrity or impairment of function.

The staff concludes that the design of reactor internals for the ABWR is acceptable and meets GDC 1, 2, 4, and 10 and 10 CFR 50.55a.

On the basis of these evaluations related to designing all safety-related reactor internals (1) as Safety Class 2; (2) to the quality assurance requirements of 10 CFR Part 50, Appendix B; and (3) to acceptable ASME Code, Section III, rules, GE meets GDC 1 and 10 CFR 50.55a with regard to designing the reactor internals to quality standards commensurate with the importance of the safety functions to be performed.

On the basis of these evaluations related to designing all safety-related reactor internals to acceptable loading combinations and stress limits when the internals are subjected to the loads associated with normal, upset, emergency, and faulted conditions, GE meets GDC 2, 4, and 10 with respect to designing components important to safety to withstand the effects of earthquakes and the effects of normal operation, maintenance, testing, and postulated LOCAs with sufficient margin to ensure that their capability to perform their safety functions is maintained and the specified fuel design limits are not exceeded.

3.9.6 Testing of Pumps and Valves

SRP Section 3.9.6 provides guidance for review of inservice testing (IST) of certain safety-related pumps and valves typically designated as ASME Code Class 1, 2, or 3. The staff's review acceptance criteria are based on meeting (1) GDC 37 as related to periodic functional testing of the ECCS to assure the leak tight integrity and performance of its active components; (2) GDC 40 as related to periodic functional testing of the containment heat removal system to assure the leak tight integrity and performance of its active components; (3) GDC 43 as related to periodic functional testing of the containment atmospheric cleanup systems to assure the leak tight integrity and the performance of the active components, such as pumps and valves; (4) GDC 46 as related to periodic functional testing of the cooling water system to assure the leak tight integrity and performance of the active components; (5) GDC 54 as related to piping systems penetrating containment being designed with the capability to test periodically the operability of the isolation and determine valve leakage acceptability; and (6) 10 CFR Part 50, Subsection 50.55a(f) as related to including pumps and valves whose function is required for safety in the inservice testing program to verify operational readiness by periodic testing.

In Section 3.9.3 of this report, the staff discusses the design of safety-related pumps and valves for the ABWR. The load combinations and stress limits used in the design of pumps and valves ensure that the integrity of the component pressure boundary will be maintained. In addition, a licensee will periodically test the performance and measure performance parameters of safety-related pumps and valves in accordance with ASME Code Section XI, as required by 10 CFR 50.55a(f). Periodic measurements of various parameters will be compared to baseline measurements to detect long-term degradation of the pump or valve performance. The tests, measurements, and comparisons will ensure the operational readiness of these pumps and valves. However, as discussed in SECY-90-016, the staff determined that ASME Code Section XI requirements do not assure the necessary level of component operability that is desired for evolutionary LWR designs. Accordingly, in SECY-90-016, as supplemented by the staff's April 27, 1990, response to comments by the ACRS, the staff recommended criteria to the Commission to be used to supplement Section XI of the ASME Code. In its SRM of June 26, 1990, on SECY-90-016, the Commission approved the staff's recommendations. The staff's proposed applicable regulations, as described in Section 1.6 of this report, for inservice testing are as follows.

> All pumps and valves of the standard design subject to the test requirements set forth in 10 CFR 50.55a(f) are subject to the following additional limitations and modifications.
>
> (1) Piping design must incorporate provisions for full flow testing at maximum design flow of pumps and check valves;
>
> (2) Check valve testing must incorporate the use of advanced non-intrusive techniques to address degradation and performance characteristics;
>
> (3) Provisions must be established to determine the frequency necessary for disassembly and inspection of pumps and valves to detect unacceptable degradation that cannot be detected through the use of advanced non-intrusive techniques; and
>
> (4) Provisions must be incorporated to test motor-operated valves under design basis differential pressure.

The staff's evaluation of GE's responses to ASME Code Section XI issues, including the above applicable regulations, is discussed in the following sections.

SSAR Section 3.9.6 states that IST of safety-related pumps and valves will be performed in accordance with the requirements of ASME/ANSI OMa-1988 Addenda to ASME/ANSI OM-1987, Parts 1, 6, and 10. SSAR Table 1.8-21 also indicates that the applicable code of record for the ABWR is the ASME Code 1989 Edition. ASME/ANSI Part 6 of Operations and Maintenance (OM), "Inservice Testing of Pumps," and Part 10, "Inservice Testing of Valves," are referenced in Section XI, ASME Code 1989 Edition. The 1988 Addenda and the 1989

Edition of Section XI have been incorporated by reference into 10 CFR 50.55a and are acceptable for the evolutionary LWR IST provided the analysis of leakage rates and corrective action requirements of OM Part 10, paragraph 4.2.2.3 are applied to containment isolation valve testing. The staff finds the containment isolation valve leak testing to be acceptable as discussed in Section 3.9.6.2.4 of this report.

SSAR Table 3.9-8 lists the IST parameters and frequencies for safety-related pumps and valves. This table was provided as part of the design certification and contains the information regarding the ability to test pumps and valves in accordance with the ASME Code. The staff's assessment of the IST plan is contained in Section 3.9.6.3 of this report. However, the development of a complete plant-specific IST program will be the responsibility of the COL applicant. The comprehensive pump and valve IST program will provide additional information beyond that contained in the IST plan. The IST program will include the tests performed on each pump and valve and the Code requirement met by each test; test parameters and frequency of the tests; the normal, safety, and fail-safe position of each valve; component type for each pump and valve; and P&ID coordinates for each pump and valve. In addition, the COL applicant will submit any requests for relief, which the NRC staff will review on the basis of the ASME Code edition referenced in 10 CFR 50.55a(f), the ABWR design, and the inservice testing methods available at the time of the COL application. This was DFSER COL Action Item 3.9.6.3-1. SSAR Section 3.9.7.3 includes this action item and states that the COL applicant will provide a detailed pump and valve IST plan. This is acceptable.

GE's description of the primary elements of the inservice testing plan for design certification in SSAR Section 3.9.6 are discussed below.

3.9.6.1 Testing of Safety-Related Pumps

In response to the staff's concern regarding the adequacy of design and qualification for safety-related pumps, GE stated in SSAR Section 3.9.6.1 that for each safety-related pump, the design basis and required operating conditions (including tests) under which the pump will be required to function will be established. GE further stated that the COL applicant will establish the design and qualification requirements and will provide acceptance criteria for these requirements. For each size, type, and model the COL applicant will perform testing encompassing design conditions that demonstrate acceptable flow rate and corresponding head, bearing vibration levels, and pump internals wear rates for the operating time specified for each system mode of pump operation. From these tests the COL applicant will also develop baseline hydraulic and vibration data for evaluating the acceptability of the pump after installation. The staff finds that the planned actions provide a reasonable assurance for the adequacy of the design and qualification for safety-related pumps and are, therefore, acceptable.

In response to the staff's policy concern about the adequacy of minimum-flow systems for safety-related pumps, GE states in SSAR Section 3.9.6.1 that safety-related pumps and piping configurations can accommodate inservice testing at a flow rate at least as large as the maximum design flow for the pump. This commitment is responsive to the guidelines on this issue contained in SECY-90-016 and meets Item 1 of the staff's recommendations, as an applicable regulation, on the pump and valve inservice testing issues as described in Sections 1.6 and 3.9.6 of this report. This is acceptable. In addition, GE also stated in the SSAR that it would evaluate the sizing of each minimum recirculation flow path to ensure that its use under all analyzed conditions will not result in degradation of the pump and to periodically measure the minimum recirculation flow rate to verify that it is in accordance with the design specification. In response to another staff request, GE states in the same SSAR section that it would provide the safety-related pumps with instrumentation to verify that the net positive suction head (NPSH) is greater than or equal to the NPSH required during all modes of pump operation. The staff finds this acceptable because they meet SRP 3.9.6 testability requirements for safety-related pumps.

In Section 3.9.6.1 of the DSER (SECY-91-235), the staff identified an open item relating to disassembly and inspection of safety-related pumps (Outstanding Issue 14). In its letter dated December 19, 1991, GE stated that the COL applicant will develop a program to establish the frequency and the extent of disassembly and inspection for all safety-related pumps, including the basis for the frequency and the extent of each disassembly. GE further stated that the program may be revised throughout the plant life to minimize disassembly, based on past disassembly experience. Subsequent to these clarifications, the staff concluded this program to be the responsibility of the COL applicant. Therefore, Outstanding Issue 14 was closed and was reidentified as DFSER COL Action Item 3.9.6.1-1. SSAR Section 3.9.6.1 includes this action item. This action item is responsive to the guidelines on this issue contained in SECY-90-016 and meets Item 3 of the staff's recommendations, as an applicable regulation, on the pump and valve inservice testing issues as described in Sections 1.6 and 3.9.6 of this report. This is acceptable.

3.9.6.2 Testing of Safety-Related Valves

3.9.6.2.1 Check Valves

In response to the staff's concern about the adequacy of design and qualification for safety-related check valves, SSAR Section 3.9.6.2.1 states that for each check valve with active safety-related function, the design basis and required operating conditions (including testing) under which the check valve will be required to perform will be established. The SSAR further states that the COL applicant will establish the design and qualification requirements and will provide acceptance criteria for these requirements. By testing each size, type, and model, the COL applicant will ensure the design adequacy of the check valve under design conditions, including severe transient loadings expected during the life of the valve such as waterhammer or pipe break. This testing of each size, type, and model will include test data from the manufacturer, field test data for dedication by the COL applicant, empirical data supported by test, or test (such as prototype) of similar valves that support qualification of the required valve where similarity must be justified by technical data. The COL applicant will ensure that the maximum loading on the check valve under design basis and the required operating conditions is within the structural capability limits for the individual parts of the check valve. In addition, the COL applicant will ensure proper check valve applications in the piping system design. Specific considerations will include selection of valve size and type based on the system flow conditions, installed location of valve with respect to source of turbulence, and correct orientation of valve in the system (e.g., vertical versus horizontal) as recommended or required by the valve manufacturer. The qualification acceptance criteria will include baseline data developed during qualification testing and will be used for verifying the acceptability of the check valve after installation. Furthermore, during the pre-operational testing, the COL applicant will test each safety-related check valve in the open and/or closed direction, as required by the safety function, under all normal operating system conditions. The testing requirements and acceptance criteria, including leak-tightness, disk stability, and correct valve sizing, are provided in the SSAR. The staff finds these planned actions provide a reasonable assurance for the adequacy of the design, qualification, and pre-operational testing for safety-related check valves and are, therefore, acceptable.

In response to the staff's concern about the full-flow testing of check valves, SSAR Section 3.9.6.2.1 states that ABWR safety-related piping systems incorporate provisions for testing to demonstrate the operability of the check valves under design-basis conditions. In response to another staff request, the SSAR states that advanced non-intrusive techniques will be used in the implementation of inservice testing program to periodically assess degradation and the performance characteristics of check valves. These are responsive to the guidelines on these issues contained in SECY-90-016 and meets Items 1 and 2 of the staff's recommendations, as an applicable regulation, on the pump and valve inservice testing issues as described in Sections 1.6 and 3.9.6 of this report and are, therefore, acceptable.

In Section 3.9.6.2.1 of the DSER (SECY-91-235), the staff identified an open item relating to disassembly and inspection of safety-related check valves (Outstanding Issue 15). In its letter dated December 19, 1991, GE stated that the COL applicant referencing the ABWR design will develop a program to establish the frequency and the extent of disassembly and inspection on the basis of suspected degradation of safety-related check valves, including the basis for the frequency and the extent of each disassembly. GE also stated that the program may be revised throughout the plant life to minimize disassembly based on past disassembly experience. Subsequent to these classifications, the staff determined this to be the responsibility of the COL applicant. Therefore, Outstanding Issue 15 was closed and was reidentified as DFSER COL Action Item 3.9.6.2.1-1. GE has included this action item in Section 3.9.6.2.1 of the SSAR. This action item is responsive to the guidelines on this issue contained in SECY-90-016 and meets Item 3 of the staff recommendations, as an applicable regulation, on the pump and valve inservice testing issue as described in Sections 1.6 and 3.9.6 of this report and is, therefore, acceptable.

3.9.6.2.2 Motor-Operated Valves

In response to the staff's concern regarding the adequacy of design and qualification for safety-related motor-operated valves (MOVs) SSAR Section 3.9.6.2.2 states that for each MOV with active safety-related function, the design basis and required operating conditions (including testing) under which the MOV will be required to perform are established for the development and implementation of the design, qualification, and pre-operational testing. For the design and qualification of MOVs, the SSAR provides commitments as follows. The COL applicant will establish the design and qualification requirements and will provide acceptance criteria for these requirements. The COL applicant will test each size, type, and model to determine the torque and thrust requirements to operate the MOV and will ensure the adequacy of the torque and thrust that the motor operator can deliver under design conditions. The COL applicant will also test each size, type, and model under a range of differential pressure and flow conditions up to the design conditions. These design conditions include fluid flow, differential pressure (including pipe break), system pressure, fluid temperature, ambient temperature, minimum voltage, and minimum and maximum stroke time requirements. This testing of each size, type, and model will include test data from the manufacturer, field test data for dedication by the COL applicant, empirical data supported by test, or test of similar valves (such as prototype) that support qualification of the required valve where similarity must be justified by technical data. From this testing, the COL applicant will demonstrate that the results of testing under in situ or installed conditions can be used to ensure the capability of the MOV to operate under design conditions. The COL applicant will also ensure that the structural capability limits of the individual parts of the MOV will not be exceeded under design conditions. Furthermore, the COL applicant will ensure that the valve specified for each application is not susceptible to pressure locking and thermal binding. For the pre-operational testing of MOVs, the SSAR provides commitments as follows. The COL applicant will test each MOV in the open and closed directions under static and maximum achievable conditions up to design basis conditions, using diagnostic equipment that measures torque and thrust and motor parameters. The COL applicant will test the MOV under various differential pressure and flow up to maximum achievable conditions and perform a sufficient number of tests to determine the torque and thrust requirements at design conditions. The specific testing parameters and acceptance criteria for demonstrating that the adequacy of the MOV functional performance has been met are provided in the SSAR. The staff finds these commitments regarding the design, qualification, and pre-operational testing for safety-related MOVs provide a reasonable assurance for demonstrating the adequacy of the MOV capability for the design basis conditions and are, therefore, acceptable.

The commitments just discussed also relate to the ITAAC information applicable to MOVs discussed in Section 3.9.6.4 of this report. To follow the 10 CFR Part 52 design certification process, any change to these commitments would involve an unreviewed safety question and, therefore, requires NRC review and approval prior to implementation. Furthermore any requested change to these commitments must either be specifically described in the COL application or submitted for license amendment after COL issuance.

In the DFSER, the staff noted that the COL applicant should address the design, qualification, and pre-operational testing for the MOVs as discussed in SSAR Section 3.9.6.2.2, prior to plant startup. This was COL Action Item 3.9.2.2-1. SSAR Section 3.9.7.3 includes the above information. The staff finds this to be acceptable.

In the DSER (SECY-91-235), the staff identified an open item relating to justifying the use of prototype qualification testing of MOVs (Outstanding Issue 16). The staff's specific concern is the need for proper justification of the applicability of prototype test data. In response to the staff's concern, GE committed to revise the SSAR to reference GL 89-10, Supplement 1, staff responses to Q22 and Q24 through Q28, which contain staff guidelines to justify prototype testing. Therefore, Outstanding Issue 16 was resolved and was reidentified as DFSER Confirmatory Item 3.9.6.2.2-1. As described in SSAR Section 3.9.6.2.2, testing of MOVs of each size, type, and model will include test of similar valves (such as prototype) that support qualification of the required valve where similarity must be justified by technical data. The staff found that this commitment for prototype qualification testing is acceptable, and there is no need to reference GL 89-10, Supplement 1, for conducting MOV prototype qualification testing. As a result, DFSER Confirmatory Item 3.9.6.2.2 is withdrawn and closed.

In response to the staff's concern about the periodic testing for MOVs, GE stated that the SSAR will be revised to address the COL applicant's program to periodically test MOVs. In the DFSER, the staff noted that the COL applicant will determine the optional frequency for valve stroking during inservice testing. This was part of DFSER COL Action Item 3.9.6.2.2-2. SSAR Section 3.9.6.2.2 states that periodic testing per GL 89-10 Paragraphs D and J will be conducted under adequate differential pressure and flow conditions that allow a justifiable demonstration of continuing MOV capability for design-basis conditions. SSAR Section 3.9.7.3 also states that the COL applicant will determine the optimal frequency of this periodic verification of the continuing MOV capability for design basis conditions and will include this requirement in the development of the detailed IST program. This is responsive to the guidelines on the MOV testing contained in SECY-90-016 and meets Item 4 of the staff's recommendations, as an applicable regulation, on the pump and valve inservice testing issues as described in Sections 1.6 and 3.9.6 of this report and is acceptable. GE further stated that the ASME Code provides criteria limits for the test parameters identified in SSAR Table 3.9-8 for the ASME Code inservice testing.

In GL 89-10, the staff recommended that MOVs in a safety-related system should either be designed to prevent mispositioning or be subjected to qualification testing to demonstrate capability to recover from mispositioning. This was DFSER Confirmatory Item 3.9.6.2.2-2. The BWR Owners' Group subsequently submitted a backfit appeal on that GL 89-10 recommendation. The staff, with the assistance of Brookhaven National Laboratory, reviewed and reevaluated the issue, using probabilistic risk assessment techniques, and determined that the recommendations for MOVs mispositioning is not necessary for BWRs. The staff subsequently issued GL 89-10, Supplement 4, and stated that the staff no longer considered the recommendations for inadvertent operation of MOVs to be within the scope of GL 89-10 for BWRs. Therefore, the DFSER Confirmatory Item 3.9.6.2.2-2 is withdrawn and resolved.

In the DSER (SECY-91-235), the staff identified an open item regarding disassembly and inspection of safety-related MOVs (Outstanding Issue 17). In its letter dated December 19, 1991, GE stated that the COL applicant referencing the ABWR design will develop a program to establish the frequency and the extent of disassembly and inspection based on suspected degradation of safety-related MOVs, including the basis for the frequency and the extent of each disassembly. GE further stated that the program may be revised throughout the plant life to minimize disassembly based on past disassembly experience. Subsequent to the above clarifications, the staff determined this program to be the responsibility of the COL applicant. Therefore, Outstanding Issue 17 was closed and was reidentified also as part of DFSER COL Action Item 3.9.6.2.2-2. GE has included this information in SSAR Section 3.9.6.2.2. This is responsive to the guidelines on this issue contained in SECY-90-016 and meets Item 3 of the staff's recommendations, as an applicable regulation, on the pump and valve inservice testing issues as described in Sections 1.6 and 3.9.6 of this report and is, therefore, acceptable.

SSAR Section 3.9.6.2.2 states that the inservice testing of MOVs will rely on diagnostic techniques that are consistent with the state of the art which will permit an assessment of the performance of the valve under actual loading conditions. This is responsive to the staff's guidelines on this issue and is, therefore, acceptable.

3.9.6.2.3 Power-Operated Valves

In response to the staff's concern about the adequacy of design and qualification for safety-related power-operated valves (POVs) other than MOVs, SSAR Section 3.9.6.2.3 states that for each POV with active safety-related function, the design basis and required operating conditions (including testing) under which the POV will be required to perform will be established. The SSAR further states that COL applicant will establish the design and qualification requirements and will provide acceptance criteria for these requirements. By testing each size, type, and model, the COL applicant will determine the force requirements to operate the POV and will ensure the adequacy of the force that the operator can deliver under design conditions. The COL applicant will also test each size, type, and model under a range of differential pressure and flow conditions up to the design conditions. This testing of each size, type, and model will include test data from the manufacturer, field test data for dedication by the COL applicant, empirical data supported by test, or test of similar valves (such as prototype) that support qualification of the required valve where similar must be justified by technical data. From this testing, the COL applicant will demonstrate that the results of testing under in-situ conditions can be used to ensure the capability of the POV to operate under design conditions. The COL applicant will also ensure that the structural capability limits of the individual parts of the POV will not be exceeded under design conditions. The COL applicant will ensure that packing adjustment limits are specified for the valve for each application such that it is not susceptible to stem binding. SSAR Section 3.9.6.2.3 also states that during pre-operational testing, the COL applicant will test each POV in the open and closed directions under static and maximum achievable conditions, using diagnostic equipment that measures or provides information to determine total friction, stroke time, seat load, spring rate, and travel under normal pneumatic or hydraulic pressure and minimum pneumatic or hydraulic pressure. The COL applicant will test the POV under various differential pressure and flow up to maximum achievable conditions, including design basis conditions. The COL applicant will perform a sufficient number of tests to determine the force requirements at design conditions. The specific testing parameters and acceptance criteria for demonstrating that the adequacy of the POV functional performance has been met are provided in the SSAR. The staff finds that these commitments provide a reasonable assurance for the adequacy of the design, qualification, and pre-operational testing for safety-related POVs and, are therefore, acceptable.

SSAR Section 3.9.6.2.3 also states that ABWR safety-related piping systems incorporate provisions for testing to demonstrate the operability of the POVs under design-basis conditions. Inservice testing will incorporate the use of advanced non-intrusive techniques to periodically assess degradation and the performance characteristics of POVs. These are responsive to the staff's guidelines on these issues and are acceptable. The SSAR further states that a program will be developed by the COL applicant to establish the frequency and the extent of disassembly and inspection based on suspected degradation of safety-related POVs, including the basis for the frequency and the extent of each disassembly. The SSAR also states that the program may be revised throughout the plant life to minimize disassembly based on past disassembly experience and identified the corresponding COL applicant actions in SSAR Section 3.9.7.3. This is responsive to the guideline on this issue contained in SECY-90-016 and meets Item 3 of the staff's recommendations, as an applicable regulation, on the pump and valve inservice testing issues as described in Sections 1.6 and 3.9.6 of this report. This is acceptable.

3.9.6.2.4 Isolation Valve Leak Tests

In response to the staff's concern about the leak-tight integrity of isolation valves, SSAR Section 3.9.6.2.4 states that the leak-tight integrity of each valve relied on to provide a leak-tight function will be verified. These valves include (1) pressure isolation valves that provide isolation of pressure differential from one part of a system to another or between systems, (2) temperature isolation valves whose leakage may cause unacceptable thermal stress fatigue or stratification in the piping and thermal loading on supports or whose leakage may cause steam binding of pumps, and (3) containment isolation valves that perform a containment isolation function, including valves that are not a part of the 10 CFR Part 50, Appendix J, Type C, testing program but whose leakage may cause loss of water inventory of a suppression pool. The staff's evaluations of SSAR Section 3.9.6.2.4 are as follows.

In the DSER (SECY-91-235), the staff noted a concern about the scope of containment isolation valves (Outstanding Issue 18). In response, GE stated that the scope of containment isolation valves will be in accordance with GDC 54 and agreed to amend the SSAR accordingly. This was DFSER Confirmatory Item 3.9.6.2.4-1. SSAR Section 3.9.6.2.4 includes this information. This is acceptable, and DSER Outstanding Issue 18 and DFSER Confirmatory Item 3.9.6.2.4-1 are resolved. The staff's evaluation of this issue is also discussed in Section 6.2.4 of this report.

In the DSER (SECY-91-235), the staff requested that GE comply with the analysis of leakage rates and corrective action requirements of ASME/ANSI OM Part 10, paragraph 4.2.2.3, for containment isolation valves (Outstanding Issue 19). In response, GE stated that leakage rate testing of containment isolation valves will be in accordance with ASME Code Section XI. However, this was not completely acceptable. OM Part 10, which is referenced in ASME Code Section XI, 1989 Edition, only requires containment isolation valves to be tested in accordance with Appendix J to 10 CFR Part 50 and does not require that corrective action be based on exceeding individual valve leakage limits. The staff's position is that the analysis of leakage rates and corrective action requirements of paragraph 4.2.2.3 in OM Part 10 are also applicable to containment isolation valves. Subsequently, the staff requested GE to revise its response in accordance with the staff's position and reidentified Outstanding Issue 19 as DFSER Open Item 3.9.6.2.4-1. SSAR Section 3.9.6.2.4 provides the requirements for containment isolation valves leakage rate testing in accordance with the staff's position, which is acceptable, and DSER Outstanding Issue 19 and DFSER Open Item 3.9.6.2.4-1 are resolved.

Several safety systems connected to the RCPB have design pressures below the rated RCS pressure. Also, some systems that are rated at full reactor pressure on the discharge side of pumps have pump suction below RCS pressure. To protect these systems from RCS pressure, two or more isolation valves will be placed in series to form the interface between the high-pressure RCS and the low-pressure system. SSAR Table 3.9-9 provides a list of RCS pressure isolation valves. In the DSER (SECY-91-235), the staff reported that its review of Table 3.9-9 was in progress (Outstanding Issue 20). Subsequently, the staff reviewed the RCS pressure isolation valves list provided in the SSAR and determined that this list contains all of the RCS pressure isolation valves and was acceptable, and that DSER Outstanding Issue 20 was resolved.

In the DFSER, the staff also identified a concern about the periodic leak testing of all pressure isolation valves. The staff's position is that the leak tight integrity of these valves must be ensured by periodic leak testing to prevent exceeding the design pressure of the low-pressure systems. This was DFSER TS Item 3.9.6.2.4-1. In response, GE stated that the periodic leak rate testing of the RCS pressure isolation valves in Table 3.9-9 will be performed in accordance with the surveillance requirements of the ABWR TS. GE also stated that the final proposed ABWR TS, to be considered and approved under the design certification program, will reflect the relevant surveillance requirements of the new BWR Standard Technical Specifications. This was DFSER TS Item 3.9.6.2.4-1. Subsequently, GE included this information in the ABWR TS Section 3.4.4, "RCS Pressure Isolation Valve Leakage," dated November 12, 1992. The periodic leak rate testing requirements for ABWR RCS pressure isolation valves are the same as the relevant surveillance requirements of the new BWR Standard TS and are acceptable. As a part of the resolution of the intersystem loss-of-coolant accident (ISLOCA) issue, low pressure piping systems that interface with the RCPB are required to be designed to withstand full RCS pressure to the extent practicable. The staff's evaluation of the ISLOCA design for ABWR piping systems has been found to be acceptable as discussed in Section 3.9.3.1.1 of this report.

3.9.6.3 Review of SSAR Table 3.9-8, Inservice Testing Plan

As discussed previously in Section 3.9.6 of this report, GE made certain commitments are made regarding the testability of safety-related pumps and valves in the ABWR design. In SSAR Section 3.9.6, GE stated that Code testing flexibility in the ASME/ANSI OM Part 6 and Part 10 produced no need for relief requests. In SSAR Table 3.9-8, GE listed the inservice testing parameters and frequencies for safety-related pumps and valves. In the DSER (SECY-91-235), the staff identified the review of SSAR Table 3.9-8 as Outstanding Issue 13. However, in the DFSER, Outstanding Item 13 was replaced by DFSER Open Item 3.9.6.3-1, which is discussed below.

The staff's review of SSAR Table 3.9-8 to ensure that GE's commitments regarding the ability to test pumps and valves can be met resulted in a list of questions transmitted to GE in a letter dated May 4, 1992. The staff stated that those questions should not be used to determine a comprehensive list of problem areas in SSAR Table 3.9-8 and that GE should systematically review and revise its ABWR IST plan, emphasizing the design configuration to provide assurance that its commitment regarding the ability to test pumps and valves can be met. Subsequently, a meeting with GE was held on June 8, 1992, in San Jose, California. As a result of that meeting, GE made several commitments that included addressing those questions identified in the staff's letter of May 4, 1992; revising SSAR Table 3.9-8 as well as the associated P&IDs; and performing a systematic review of its IST plan. GE also indicated that some exceptions to the Code requirements may be needed after it completes a systematic review of the IST plan. For the staff review of any Code exceptions, GE committed to identify the Code requirement, to provide a basis to justify the need for the code exceptions, and to describe its proposed alternative testing method. The staff also informed GE that it would be unacceptable to leave for the COL phase certain aspects of the final design related to testability of pumps and valves. Specifically, details of the piping configuration related to additional lines, valves, and instrumentation must be provided before FDA. The development and submittal of an acceptable IST plan was DFSER Open Item 3.9.6.3-1.

In response to the staff's request, GE subsequently revised SSAR Table 3.9-8. The staff, with the assistance of Science Applications International Corporation (SAIC), reviewed and evaluated the revised ABWR IST Plan as presented in SSAR Table 3.9-8. On the basis of that evaluation, the staff determined that the ABWR pump and valve IST plan provided a reasonable assurance that GE's commitment as described in SSAR Section 3.9.6 regarding the ability to test pumps and valves can be met and is acceptable, and that DFSER Open Item 3.9.6.3-1 is resolved. Details of that evaluation are included as Appendix I to this report. Appendix I consists of the Technical Evaluation Report prepared by SAIC that has been modified by the staff to reflect subsequent changes made to Table 3.9-8 after SSAR Amendment 20. Those changes were made as a result of later design changes as well as resolutions to the staff's concerns identified in the DSER (SECY-91-235).

In the DFSER, the staff also identified an open item that GE should establish criteria to be used by the COL applicant for developing pump and valve design specifications to ensure that these components are capable of performing their design-basis functions. This was DFSER Open Item 3.9.6.3-2. GE provided these criteria in Sections 3.9.6.1, 3.9.6.2.1, 3.9.6.2.2, and 3.9.6.2.3 of the SSAR. The staff has found the criteria for developing pump and valve design specifications to be acceptable as discussed in Sections 3.9.6.1, 3.9.6.2.1, 3.9.6.2.2, and 3.9.6.2.3 of this report, and DFSER Open Item 3.9.6.3-2 is resolved.

3.9.6.4 Certified Design Material

In the DFSER, the staff identified an Open Item 3.9.6.2.3-1 that requested GE to submit an acceptable generic ITAAC for demonstrating MOV capability. In response to the staff's request, GE submitted systems-specific ITAACs that include criteria applicable to MOVs. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Section 14.3 of this report, and DFSER Open Item 3.9.6.2.3-1 is resolved.

In Section 3.9.6.4 of the DFSER, the staff indicated the need for an acceptable ITAAC for POVs other than MOVs. The staff's specific concern was the inadequacy of the POV capability for the design-basis conditions. That was DFSER Open Item 3.9.6.4-1. In response to the staff's concern, GE provided information about the design, qualification, pre-operational and inservice testing requirements for safety-related POVs in SSAR Section 3.9.6.2.3. As discussed in Section 3.9.6.2.3 of this report, the staff finds that GE's commitments provide a reasonable assurance for demonstrating the adequacy of the POV capability for the design-basis conditions. Therefore, the staff subsequently determined that this particular ITAAC is not necessary. The staff also determined that although it was concluded that other POVs would not need Tier 1 treatment, the SSAR does not contain sufficient information on the design and qualification and on the pre-operational testing of other POVs. In response to the staff's concern, SSAR Section 3.9.6.2.3 includes this information. The staff's evaluation of that issue has been found to be acceptable as discussed in Section 3.9.6.2.3 of this report. Furthermore, on the basis of past operational experience and staff's inspections, the staff requested GE to provide an ITAAC for check valves. In response to the staff's request, GE submitted system-specific ITAACs that include criteria applicable to check valves. The adequacy and acceptability of the ABWR check valves ITAAC is evaluated in Section 14.3 of this report, and DFSER Open Item 3.9.6.4-1 is resolved.

3.9.6.5 Conclusion

Based on the evaluations described above, the staff concludes that the pump and valve IST program described in the SSAR is acceptable and meets the requirements of GDC 37, 40, 43, 46, and 54 and 10 CFR 50.55a(f). This conclusion is based on the commitments made regarding the testability of safety-related pumps and valves in the ABWR design, which will ensure the leaktight integrity and the operational readiness to perform necessary safety functions throughout the life of the plant. The pump and valve testing will include baseline preservice testing and visual inspection for leaks and other signs of distress. The staff further concludes that the ABWR standard plant meets the Commission-approved staff positions for inservice testing of pumps and valves contained in SECY-90-016 and its applicable regulation for inservice testing of pumps and valves as discussed in Sections 1.6 and 3.9.6 of this report. This is acceptable. This conclusion is based on that the commitments made in SSAR Sections 3.9.6.1, 3.9.6.2.1, 3.9.6.2.2, and 3.9.6.2.3 for safety-related pumps and valves will ensure (1) the full-flow testing of pumps and valves; (2) the use of advanced non-intrusive techniques for check valve testing; (3) the development of a disassembly and inspection program for pumps and valves; and (4) the adequacy of the MOV capability for the design-basis conditions.

3.10 Seismic and Dynamic Qualification of Mechanical and Electrical Equipment

SSAR Sections 3.9.2.2 and 3.10 provide information on the seismic and dynamic qualification of safety-related mechanical and electrical equipment. Section 3.9.3.2 also contains information related to pump and valve operability assurance. This information includes

- rationale used to determine if tests, analyses, or combinations of both will be performed
- criteria used to define the seismic and other relevant dynamic load input motions
- the proposed demonstration of the adequacy of the qualification program

The staff's review acceptance criteria are based on meeting (1) GDC 1 and 30 as related to qualifying equipment to appropriate quality standards commensurate with the importance of the safety functions to be performed; (2) GDC 2 and Appendix A to 10 CFR Part 100 as related to qualifying equipment to withstand the effects of natural phenomena such as earthquakes; (3) GDC 4 as related to qualifying equipment being capable of withstanding the dynamic effects associated with external missiles and internally generated missiles, pipe whip, and jet impingement forces; (4) GDC 14 as related to qualifying equipment associated with the reactor coolant boundary so as to have an extremely low probability of abnormal leakage, or rapidly propagating failure and of gross rupture; and (5) Appendix B to 10 CFR Part 50 as related to qualifying equipment using the quality assurance criteria provided.

GE will use the seismic qualification methodology in Section 4.4 of GE report NEDE-24326-1 (proprietary) for both mechanical and electrical equipment. This program conforms to IEEE 323 as modified and endorsed by RG 1.89, "Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plant," Revision 1. The program also meets the criteria in IEEE 344 as modified by RG 1.100, "Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants," Revision 2. In Tables 1.8-20 and 1.8-21 of the SSAR, GE agrees to use RG 1.100, Revision 2, June 1988, and IEEE 344, 1987. Section 9 of IEEE 344, 1987, recognizes the use of "experience data" as a method for seismic qualification of equipment. As used in IEEE 344, experience data includes both seismic experience and previous qualifications. In accordance with RG 1.100, Revision 2, the method of qualification will be reviewed by the staff on a case-by-case basis. In Amendment 27 to the SSAR, GE revised Section 3.10.1.1, "Selection of Qualification Method" to permit the use of experience data for seismic qualification of seismic Category I instrumentation and electrical equipment. In a letter dated May 14, 1993, GE provided a markup of the SSAR, which further revised Section 3.10.1.1 and added Section 3.10.5.3 to state that if dynamic qualification of seismic Category I instrumentation or electrical equipment is accomplished by experience, the COL applicant will provide the following to the NRC for review and approval:

- identification of the specific equipment
- the details of the methodology and the corresponding experience data for each piece of equipment

This information is in SSAR Sections 3.10.1.1 and 3.10.5.3. The staff concludes that the above commitment is consistent with the applicable portion of RG 1.100, Revision 2, and is acceptable.

The staff reviewed NEDE-24326-1 (proprietary) and approved the qualification methodology therein in an SER sent to GE on October 23, 1983. In response to the staff's request for information in Q 271.2, GE stated that the methodology in NEDE-24326-1 (proprietary) also conforms to the above commitments as shown in Tables 1.8-20 and 1.8-21 of the SSAR. This response is acceptable.

The methodology in NEDE-24326-1 (proprietary), supplemented by the information in SSAR Sections 3.9.2.2, 3.9.3.2, and 3.9.6, provides test and/or analysis criteria used to demonstrate the operability of active pumps and valves (i.e., those ASME Class 1, 2, or 3 components that must perform a mechanical motion to shut down the plant or mitigate the consequences of a postulated event). The staff concludes that the criteria described are consistent with the guidelines in SRP 3.10 and applicable portions of SECY-90-016, and are acceptable. To provide a more detailed basis for the staff's conclusion, in the following list are applicable guidelines from SRP 3.10 together with references to commitments in the SSAR which address each guideline.

(1) Tests and analyses are required to confirm the operability of all mechanical and electrical equipment during and after an earthquake of magnitude up to and including the SSE, and for all static and dynamic loads from normal, transient and accident conditions. Prior to SSE qualification, demonstrate that the equipment can withstand excitation less than the SSE without loss of structural integrity. Analyses alone, without testing, are acceptable as a basis for qualification only if the necessary functional operability of the equipment is assured by its structural integrity alone. When complete testing is impractical, a combination of tests and analyses is acceptable.

    Equipment that has been previously qualified by means of tests and analyses equivalent to those described here are acceptable provided that proper documentation of such tests and analyses is submitted.

    Commitments to most of the above criteria can be found in the SSAR Subsections 3.9.2.2, 3.9.3.2, and 3.10.1.1(B) and in NEDE-24326-1 (proprietary), Sections 4.4.2.5.1, 4.4.3.3 and 4.4.4.

    In addition, consistent with staff positions on recently licensed plants, Sections 3.9.3.2.1.1 and 3.9.3.2.3.1.4 in the SSAR, Amendment 29, and the markups of SSAR Sections 3.9.1.4.5, 3.9.1.4.11, and 3.9.3.2.5.1.2, and Table 3.9-2, Footnote (7) in a letter dated May 11, 1993, provide commitments that operability of active ASME Class 1, 2, and 3 valves and Class 2 and 3 pumps is further assured by limiting the stresses to the material elastic limit when the component is subjected to (1) the combination of normal operating loads, (2) SSE and other RB vibration loads, and (3) dynamic system loads (LOCA). Specifically, the average membrane stress resulting from this faulted condition (Service Level D) loads is limited to 75 percent of the material yield stress, and the maximum membrane plus bending stress is limited to 110 percent of the yield stress. Implementation of this acceptance criteria will provide assurance that valve bodies or pump cases will not distort to the extent that operability of the component is impaired. SSAR Sections 3.9.1.4 and 3.9.3.2, and Table 3.9-2 include this information. This is acceptable.

(2) Equipment should be tested in the operational condition. Operability should be verified during and/or after the testing, as applicable to the equipment being tested. Loadings simulating those of plant normal operation, such as thermal and flow-induced loading, if any, should be concurrently superimposed upon the seismic and other pertinent dynamic loading to the extent practicable. Particular attention should be paid, in operability qualification of mechanical equipment subjected to flow-induced loading, to incorporate degraded flow conditions such as those that might be encountered by the presence of debris, impurities, and contaminants in the fluid system. An example of this may be the operability of the containment sump pump recirculating water full of debris.

    Commitments to most of the above criteria can be found in the SSAR Subsections 3.9.2.2 and 3.9.3.2 and in NEDE-24326-1 (proprietary), Sections 4.4.2.5, 4.4.2.5.1 and 4.4.2.5.2.

(3) The characteristics of the required seismic and dynamic input motions should be specified by response spectrum or time history methods. These characteristics, derived from the structures or systems seismic and dynamic analyses, should be representative of the input motions at the equipment mounting locations.

Commitments to the above criteria can be found in the SSAR Subsection 3.7.3.1 and in NEDE-24326-1 (proprietary), Section 4.4.4.1.4.6.2.

(4) For seismic and dynamic loads, the actual test input motion should be characterized in the same manner as the required input motion, and the conservatism in amplitude and frequency content should be demonstrated (i.e., the test response spectrum should closely resemble and envelope the required response spectrum over the critical frequency range).

Commitments to the above criteria can be found in the SSAR Subsection 3.9.2.2.1 and NEDE-24326-1 (proprietary), Section 4.4.2.5.3(b).

(5) Since seismic and the dynamic load excitation generally have a broad frequency content, multi-frequency vibration input motion should be used. However, single frequency input motion, such as sine beats, is acceptable provided the characteristics of the required input motion indicate that the motion is dominated by one frequency (e.g., by structural filtering effects), or the anticipated response of the equipment is adequately represented by one mode, or in the case of structural integrity assurance, the input has sufficient intensity and duration to produce sufficiently high levels of stress for such assurance. Components that have been previously tested to IEEE 344-1971 should be reevaluated to justify the appropriateness of the input motion used, and requalified if necessary.

Commitments to these criteria can be found in the SSAR Subsection 3.9.3.2.3.1.4 and in NEDE-24326-1 (proprietary), Sections 4.4.2.5.3 and 4.4.2.5.6.

(6) For the seismic and dynamic portion of the loads the test input motion should be applied to one vertical axis and one principal horizontal axis (or two orthogonal horizontal axes) simultaneously unless it can be demonstrated that the equipment response in the vertical direction is not sensitive to the vibratory motion in the horizontal direction, and vice versa. The time phasing of the inputs in the vertical and horizontal directions must be such that a purely rectilinear resultant input is avoided. An acceptable alternative is to test with vertical and horizontal inputs in-phase, and then repeat the test with inputs 180 degrees out-of-phase. In addition, the test must be repeated with the equipment rotated 90 degrees horizontally.

Components that have been previously tested to IEEE 344-1971 should be requalified using biaxial test input motions unless justification for using a single axis test input motion is provided.

Commitments to the above criteria can be found in the SSAR Subsection 3.9.3.2.3.1.4 and NEDE-24326-1 (proprietary), Section 4.4.2.5.4.

(7) Dynamic coupling between the equipment and related systems, if any, such as connected piping and other mechanical components, should be considered. The fixture design should simulate the actual service mounting and should not cause any extraneous dynamic coupling to the test item. A commitment to this criteria can be found in the SSAR Subsections 3.9.2.2.1 and Item 7 of 3.10.1.1.

(8) For pumps and valves, the loads imposed by the attached piping should be properly taken into account. To assure operability under combined loadings, the stresses resulting from the applied test loads should envelope the specified service stress limit for which the component's operability is intended. As discussed in this Section 3.10, the SSAR Subsections 3.9.3.2.1.1, 3.9.3.2.3.1.4, and 3.9.3.2.5.1.2 contain criteria which addresses this issue.

(9) Selection of damping values for equipment to be qualified should be made in accordance with RG 1.61, "Damping Valves for Seismic Design of Nuclear Power Plants," Revision 0, and IEEE 344-1987. Higher damping values may be used if justified by documented test data with proper identification of the source and mechanism. SSAR Sections 3.7, 3.9.2.2, 3.9.3.2, and 3.10.2 contain criteria that addresses this issue.

(10) Section 3.10.2.1 of the SSAR states that the methodology for qualifying relays shall be such that testing is performed in both the open and closed positions.

NEDE-24326-1 (proprietary) provides qualification methodology only and contains no plant-specific information. In the DFSER, the staff noted that each COL applicant referencing this document should ensure that specific environmental parameters along with seismic and dynamic input response spectra are properly defined and enveloped in the methodology for its specific plant and implemented in its equipment qualification program. This was DFSER COL Action Item 3.10-1. SSAR Sections 3.9.3.2.3.2 and 3.9.3.2.5.2 of the SSAR state that documentation will be prepared to clearly show that the criteria outlined in SSAR Section 3.9.3.2 to demonstrate active pump and valve operability has been satisfied. This will be included as a part of the certified stress report for the pump or valve assembly. In addition, SSAR Section 3.10.5 states that the COL applicants shall maintain equipment qualification records in a permanent file that shall be readily available for audit. These are acceptable actions. The staff will audit these files to review the results of tests and analyses that were performed to (1) ensure that the criteria in the SSAR were properly implemented, (2) ensure that adequate qualification was demonstrated for all equipment and their supports, and (3) verify that all applicable loads were properly defined and accounted for in the testing and analyses performed.

3.10.1 Conclusions

On the basis of these evaluations, the staff concludes that GE has defined appropriate seismic and dynamic qualification of mechanical and electrical equipment and pump and valve operability programs. These programs meet applicable portions of GDC 1, 2, 4, 14, and 30, Appendix B to 10 CFR Part 50, and Appendix A to 10 CFR Part 100 and are acceptable. This conclusion is based on the following:

SSAR Table 3.2-1 identifies all ABWR safety-related mechanical and electrical equipment as (1) safety Class 1, 2, or 3, (2) seismic Category I, and (3) designed to the quality assurance requirements of 10 CFR Part 50, Appendix B. As discussed in Sections 3.2.1 and 3.2.2 of this report, the staff concludes that Table 3.2-1 is acceptable. On the basis of these evaluations, the staff concludes that GE meets GDC 1, 30, and 10 CFR Part 50, Appendix B, as they relate to qualifying safety-related mechanical and electrical equipment to appropriate quality standards commensurate with the importance of the safety function to be performed.

The qualification program, which will be implemented for mechanical, instrumentation, and electric equipment meets the requirements and recommendations of IEEE 344-1987 and the regulatory positions of RGs 1.61, 1.89, 1.92, 1.100, and SRP 3.9.3, provides adequate assurance that such equipment will function properly under all imposed design and service loads, including the loadings imposed by the safe shutdown earthquake, postulated accidents, and loss-of-coolant accidents. On the basis of this program, complemented by the staff's evaluations of (1) seismic classifications in Section 3.2.1 of this report, (2) protection from external missiles and internally generated missiles in Section 3.5 of this report, (3) analyses to withstand dynamic effects of postulated pipe breaks in Section 3.6.2 of this report, and (4) loading combinations and stress limits in Section 3.9.3.1 of this report, GE meets GDC 2, Appendix A to 10 CFR Part 100, GDC 4 and 14, as they relate to qualifying equipment to (1) withstand the effects of natural phenomena such as earthquakes, (2) be capable of withstanding the dynamic effects associated with external missiles, internally generated missiles, and pipe whip and jet impingement forces, and (3) demonstrate that equipment associated with the reactor coolant pressure boundary has a low probability of abnormal leakage, rapidly propagating failure, or gross failure.

To follow the 10 CFR Part 52 design certification process, any change to the commitments involving seismic and dynamic qualification of mechanical and electrical equipment discussed in Section 3.10 of this report would involve an unreviewed safety question and, therefore, requires NRC review and approval prior to implementation. Furthermore, any requested change to these commitments must either be specifically described in the COL application or submitted for license amendment after COL issuance.

3.10.2 Methods and Procedures of Analysis or Testing of Supports of Electrical Equipment and Instrumentation

SSAR Section 3.10.3, Amendment 33, described the procedures and criteria for the seismic qualification and design of the nuclear steam supply system (NSSS) electrical equipment supports; seismic Category I supports for battery racks, instrument racks, control consoles, cabinets, and panels; seismic Category I local instrument supports; and seismic Category I instrument tubing supports. GE provided the methods and criteria used for the design of the seismic Category I electrical raceway (cable trays and conduit) supports in SSAR Section 3.8.4.4.2, Amendment 33. The following covers only the staff's evaluation of the procedures and criteria for the design of the seismic Category I electrical raceway supports.

SSAR Section 3.10.3.2.2, Amendment 23, described the procedures and criteria for the design of the seismic Category I electrical raceway supports. GE used the response spectrum method to analyze the composite system of the electrical raceways and supports and calculate the seismic loads and the RB vibration (RBV) loads resulting from a safety relief valve (SRV) discharge or LOCA inside the containment. The input to the dynamic analysis is the seismic and RBV FRS generated for the supporting floor. In case the supports are attached to a wall or to two different locations, the input is the upper bound FRS envelope obtained by superimposing the FRS of both floors or locations. In addition, in many cases GE combined several FRS by superposition to generate an upper bound

FRS envelope as the input to facilitate the design. The staff found the analysis methods and approaches for the design of the cable trays, conduit, and their supports acceptable.

However, SSAR Section 3.8.4.4.2, Amendment 33, revised the analysis methods by stating that all seismic Category I cable trays and conduit supports are designed by one of the methods discussed in SSAR Subsection 3.7.3 or by the design-by-rule methods as approved by the NRC. On the basis of the staff's review and evaluation discussed in Section 3.9.2.2 of this report, the methods provided in SSAR Section 3.7.3 are acceptable for the analysis and design of the cable trays, conduit, and their supports. As for the use of the design by rule methods, SSAR Section 3.7.3.8.2, Amendment 33, stated that for distributive systems such as cable trays, conduit, and HVAC ducts, an alternative to qualification by analysis described in SSAR Subsection 3.7.3.8.1 is the design-by-rule method approved by the NRC at the time of COL application. This is also acceptable to the staff. The basis to accept the use of the design-by-rule method is provided as follows.

According to SSAR Section 3.8.4.2.4, Amendment 33, the design of seismic Category I electrical raceway supports uses codes, standards, and specifications applicable to the building structures to which they are attached. These codes include ANSI/AISC Standard N-690 (1984 Edition), AISI SG-673, "Specification for the Design of Cold-Formed Steel Structural Members," and National Electrical Manufacturers Association (NEMA), "Fittings and Supports for Conduit and Cable Assemblies." The supports are designed and located to withstand the dynamic loads generated from the analyses in three directions by means of vertical, transverse, and longitudinal support and bracing systems. As discussed in SSAR Section 3.8.4.3.3, Amendment 32, the design considers the dead loads, live loads, and seismic loads plus other RBV dynamic loads. SSAR Section 3.8.4.4.3.1, Amendment 33, also discussed two methods used in the analysis and design of cable tray supports:

(1) Rigid support with flexible trays - In this method, trays were modeled as flexible elastic systems and analyzed by the response spectrum method. The resulting reactions were used for the design of the supports.

(2) Flexible support with flexible trays - In this method, the composite system of trays and supports were modeled and analyzed by computer as a multidegree of freedom elastic system. The support motion was prescribed by the appropriate floor response spectrum. The resulting responses were used to obtain design loads for the supports.

Since the conduit systems are more flexible and have comparatively less dead load, a rigid support approach (Method (1) above) applied for the cable tray design was used.

According to the staff review of SSAR Section 3.8.4, Amendment 33, and the design audits conducted by the staff, the supports, including those for the non-seismic Category I cable trays and conduits, were designed to meet seismic Category I requirements. These design criteria and procedures meet the guidelines of SRP Section 3.8.4 and are acceptable. In the early amendments of the SSAR, GE did not provide the design procedures and criteria for the seismic Category I cable trays and conduit. This was DFSER Open Item 3.10.3-1. SSAR Section 3.8.4, Amendment 33, provided the design procedures and criteria for the seismic Category I cable trays and conduit, which are acceptable. Therefore, DFSER Open Item 3.10.3-1 is resolved.

As for the concern about the potential interaction between the non-seismic Category I cable trays and conduit and the seismic Category I cable trays and conduit, SSAR Section 3.7.5.4 describes the COL applicant's action for the as-built plant assessment. The staff's review and evaluation are discussed in Section 3.7.2 of this report. On the basis of the previous discussion, the staff concludes that the procedures and criteria for the design of seismic Category I raceway supports are acceptable.

3.11 Environmental Qualification of Mechanical and Electrical Equipment

The staff reviewed the ABWR design environmental qualification requirements for mechanical and electrical equipment in accordance with SRP Section 3.11, Revision 2. Equipment that is used to perform a necessary safety function must be demonstrated to be capable of maintaining functional operability under all service conditions postulated to occur during its installed life for the time it is required to operate. This requirement, which is embodied in GDC 1 and 4 of Appendix A and Criteria III, XI, and XVII of Appendix B to 10 CFR Part 50, is applicable to equipment located inside as well as outside the containment. More detailed requirements and guidance related to the methods and procedures for demonstrating this capability for electrical equipment are in 10 CFR 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants;" NUREG-0588, Revision 1, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment," which supplements IEEE 323 and various NRC regulatory guides and industry standards, and RG 1.89, "Environmental Qualification of Certain Electric

Equipment Important to Safety for Nuclear Power Plants," Revision 1.

The NRC staff issued NUREG-0588 in December 1979 to promote a more orderly and systematic implementation of equipment qualification programs by industry and to provide guidance to the NRC staff for its use in ongoing licensing reviews. The positions in the NUREG series report provide guidance on (1) how to establish environmental service conditions, (2) how to select methods that are considered appropriate for qualifying equipment in different areas of the plant, and (3) other areas such as margin, aging, and documentation. A final rule on environmental qualification of electrical equipment important to safety for nuclear power plants became effective on February 22, 1983. This rule, 10 CFR 50.49, specifies the requirements to be met for demonstrating the environmental qualification of electrical equipment important to safety located in a harsh environment. RG 1.89, Revision 1 (June 1984), identifies the guidelines that have to be met for complying with this rule. In conformance with 10 CFR 50.49, electrical equipment for BWRs referencing the ABWR standard design may be qualified according to the criteria specified in Category I of NUREG-0588, Revision 1, and RG 1.89, Revision 1.

The qualification requirements for mechanical equipment are principally contained in Appendices A and B to 10 CFR Part 50. The qualification methods defined in NUREG-0588 can also be applied to mechanical equipment.

To document the degree to which the environmental qualification program for the ABWR complies with the NRC environmental qualification requirements and criteria, GE provided SSAR Section 3.11, "Environmental Qualification of Safety-Related Mechanical and Electrical Equipment," and SSAR Appendix 3I (proprietary), in a response dated January 13, 1989, to the staff's request for additional information dated September 12, 1988.

3.11.1 Completeness of Qualification of Electrical Equipment Important to Safety

The following three categories of electrical equipment important to safety must be qualified in accordance with the provisions of 10 CFR 50.49(b)(1), (b)(2), and (b)(3).

- safety-related electrical equipment (relied on to remain functional during and following design-basis events)
- non-safety-related electrical equipment whose failure under the postulated environmental conditions could prevent satisfactory accomplishment of the safety functions by the safety-related equipment
- certain post-accident monitoring equipment (Category I and II accident- monitoring instrumentation as specified in RG 1.97, Rev. 3)

In the SSAR, GE stated that the design of the information systems important to safety will be in conformance with the guidelines of Revision 3 of RG 1.97. However, the footnote for Subsection 50.49(b)(3) references Revision 2 for selection of the types of post-accident monitoring equipment. In issuing Revision 3, the NRC staff stated that the conformance with Revision 3 would not alter the implementation of Subsection 50.49. Therefore, conformance with Revision 2 is not required because conformance with Revision 3 meets the underlying purpose of the rule. As a result, an exemption from Subsection 50.49(b)(3) is justified by the special circumstances set forth in Subsection 50.12(a)(2)(ii).

In the early SSAR amendments, GE stated that for the ABWR, all three categories of electrical equipment mentioned above and located in a harsh environment will be environmentally qualified. GE also identified an interface requirement that requires COL applicants to list all electrical equipment within the scope of 10 CFR 50.49 in their plant-specific environmental qualification documents (EQDs). GE's approach for selecting and identifying electrical equipment required to be environmentally qualified for the ABWR was considered acceptable. However, following a further review, the staff determined that this item should be reclassified as a COL action item and that the staff will review specific details provided by the COL applicant to demonstrate compliance with 10 CFR 50.49(b)(1), (b)(2), and (b)(3). In the DFSER, the staff noted that the details will include a list of systems and their components that are included in the plant environmental qualification program and the design features for preventing the potential adverse consequence identified in IE Information Notice 79-22, "Qualification of Control Systems." This was DFSER COL Action Item 3.11.1-1. GE has included this action in SSAR Section 3.11.1. This is acceptable.

3.11.2 Qualification Methods

3.11.2.1 Electrical Equipment in a Harsh Environment

The environmental qualification program presented in GE Topical Report NEDE-24326-1 (proprietary) outlines the methodology to qualify NSSS system safety-related electrical equipment subject to a harsh environment. GE adopted this program for the ABWR (SSAR Section 3.11.2).

The staff reviewed the topical report and found that the qualification methodology conforms to 10 CFR 50.49 and its associated standards, except for the position on the time margin. NUREG-0588 states that the time margin for certain categories of equipment (these categories are identified in this NUREG report) should be a minimum of 1 hour. The topical report has not addressed this requirement. While GE addressed the time margin in SSAR Section 3.11.1, Amendment 14, the staff noted in the DSER (SECY-91-153) that the report made no reference to the 1-hour time margin requirement discussed in NUREG-0588. Therefore, the staff identified the time-margin issue as Outstanding Issue 12 in the DSER (SECY-91-153) and required it to be resolved in accordance with NUREG-0588, Revision 1 or as amplified in RG 1.89, Revision 1.

In response to this request, GE revised its position on time-margin in Section 3.11.1 of the SSAR, Amendment 17. However, it was not clear that the intent is to comply with the guidance of NUREG-0588, Revision 1, Category 1, paragraph 3, as amplified in RG 1.89, Revision 1, Regulatory Position C.4. Therefore, in the DFSER, the staff noted that GE should confirm that SSAR Section 3.11.1 will be updated to reflect compliance with this guidance. This was DFSER Confirmatory Item 3.11.2.1-1. Subsequently, GE addressed this issue in Section 3.11.1 of Amendment 32 to the SSAR by stating that: "Some mechanical and electrical equipment may be required by the design to perform an intended safety function within minutes of the occurrence of the event but less than 10 hours into the event. Such equipment shall be shown to remain functional in the accident environment for a period of at least 1 hour in excess of the time assumed in the accident analysis unless a time margin of less than 1 hour can be justified. Such justification will include for each piece of equipment: (1) consideration of a spectrum of breaks; (2) the potential need for the equipment later in the event or during recovery operations; (3) determination that failure of the equipment after performance of its safety function will not be detrimental to plant safety or mislead the operator; and (4) determination that the margin applied to the minimum, operability time, when combined with other test margins, will account for the uncertainties associated with the use of analytical techniques in the derivation of environmental parameters, the number of units tested, production tolerances, and test equipment inaccuracies." This is consistent with the staff position on time margin as stated in NUREG-0588, Revision 1, and is acceptable. Therefore, DFSER Confirmatory Item 3.11.2.1-1 is resolved. As discussed in the next section, the time-margin issue for mechanical equipment is similarly resolved.

3.11.2.2 Safety-Related Mechanical Equipment in a Harsh Environment

Although no detailed requirements exist for mechanical equipment, GDC 1 and 4 and Criteria III and XVII of Appendix B to 10 CFR Part 50 contain the following requirements related to equipment qualification:

- Components shall be designed to be compatible with the postulated environmental conditions, including those associated with LOCAs.
- Measures shall be established for the selection and review for the suitability of application of materials, parts, and equipment that are essential to safety-related functions.
- Design control measures shall be established for verifying the adequacy of design.
- Equipment qualification records shall be maintained and shall include the results of tests and materials analyses.

In the early SSAR amendments, GE stated that the qualification program for safety-related mechanical equipment for the ABWR design will include all safety-related mechanical equipment identified in SSAR Section 3.2. GE further stated that the mechanical equipment qualification program to be applied to the ABWR will use applicable portions of the NRC-approved Topical Report NEDE-24326-1 (proprietary) and RG 1.89, Revision 1, and will be consistent with the program for qualification of mechanical equipment in a harsh environment described in the NRC-approved GESSAR (GE SSAR) II design. The ABWR program scope looks not only at the metallic components of the equipment but also at the nonmetallic components. Metallic components that form a pressure boundary are considered to be qualified by the nature of their pressure retention capability as demonstrated by the application of an ASME Boiler and Pressure Vessel stamp. Nonmetallic components, such as greases, gaskets, and lubricants, will be shown to be capable of performing their intended functions under accident environments. The design of safety-related mechanical equipment associated with the ABWR will be performed under the same internal procedural controls as that used for the design of mechanical components associated with the GESSAR II design. These controls ensure that components are designed to be compatible with their postulated operating environments, that measures are established for the selection and review of the suitability of application of the material, parts, and equipment that are essential to safety-related functions, and that there are design control measures for verifying the adequacy of the design. As stated in NEDE-24326-1 (proprietary), a complete set of qualification records are developed for each safety-related component.

In the early SSAR amendments, GE also identified an interface requirement that required COL applicants to provide in their plant-specific EQDs (1) a list of all safety-related mechanical equipment located in harsh-environment plant zones and (2) the methodology used to qualify the equipment located in harsh as well as mild-environment plant zones. However, following a further review, the staff determined that this item should be reclassified as DFSER COL Action Item 3.11.2.2-1. GE has included this action in Sections 3.11.1 and 3.11.6 of the SSAR. This is acceptable.

The staff concludes that the information GE provided on the selection and identification of mechanical equipment required to be environmentally qualified and the qualification methods for the equipment for the ABWR standard design is acceptable. The staff concludes that on the basis of SSAR Section 3.11.1, the time-margin issue as related to safety-related mechanical equipment is resolved similarly to that for the electric equipment discussed in the previous section.

3.11.3 Completeness of Information in Tables of SSAR Appendix 3I

SSAR Section 3.11 defines all the environmental conditions (normal, abnormal, test, accident, and post-accident) to which the applicable equipment may be exposed during plant operation. SSAR Appendix 3I contains the tables specifying the design limits or time-based profile of thermal environmental parameters (pressure, temperature, and relative humidity) and/or design limits for radiation environmental parameters (gamma dose rate and total gamma integrated dose) for each plant area or zone in the area under normal and/or abnormal and accident environmental conditions. The tables also include the neutron flux during normal operating conditions for different zones of the primary containment. The areas for which the environmental data are tabulated are the primary containment, secondary containment portion of the RB, remaining portions of the RB, turbine building, control building, radwaste building, service building, and outdoor area. Except for the radwaste building and the outdoor area, all other areas are further subdivided into zones on the basis of thermal and radiation environmental conditions determined for the zones. The SSAR considers a postulated RC (steam or water) pressure boundary pipe rupture as the limiting accident for calculating the design limits or the time-based profile for thermal environmental parameters during accident conditions for all zones. The SSAR considers the design-basis LOCA as the limiting accident for calculating the design limits for radiation environmental parameters during accident conditions for all applicable zones. The total normal and accident doses in a zone is based on integrating the ambient dose rate in the zone over a 60-year period and the accident dose rate in the zone over a 6-month period, respectively.

GE states that the environmental conditions identified in the tables in Appendix 3I are upper-bound envelopes for these conditions in various areas or zones to which the applicable equipment has to be designed and qualified. The environmental parameters specified in these tables are for the upper-bound envelopes. GE states that the parameters do not include margins that may be required to satisfy equipment qualification requirements. GE further notes that these tables include identification of significant enveloping abnormal conditions and each enveloping accident event that affects the zone environment. GE provides these tables for use by COL applicants in developing their plant-specific environmental qualification programs for equipment important to safety. The staff reviewed the tables in Appendix 3I of the early SSAR amendments and found a number of deficiencies. They were collectively identified in the DSER (SECY-91-153) as Outstanding Issue 13. The reclassification of this issue to a number of DFSER open issues, confirmatory issues, and COL action items and corresponding resolutions are discussed below.

In the DSER (SECY-91-153), the staff noted that the tables in SSAR Appendix 3I did not include the chemical environmental conditions (chemical composition and the resulting pH) to which the applicable equipment may be exposed during accident conditions. Subsequently, GE responded that reactor water quality characteristics for the design-basis loss-of-coolant accident (DBLOCA) are contained in SSAR Section 3I.3.2.3. Additionally, in a facsimile dated June 1, 1991, GE stated that SSAR Section 3I.3.2.3 would be updated to include information on water quality characteristics for normal operations. This was DFSER Confirmatory Item 3.11.3-1. GE has included this information in SSAR Section 3I.3.2.3. The staff reviewed this information and found it acceptable. This resolved DFSER Confirmatory Item 3.11.3-1.

The staff notes that, in the SSAR, the table and figure numbers have been reassigned as shown in Table 3.3 of this report, from those in the previous SSAR amendments. The staff reviewed these reassignments and determined them acceptable. In the discussions below, the reassigned numbers are considered unless otherwise noted.

Table 3.3 Reassigned number for tables and figures

Previous No.	No.in SSAR Amnt. 32	Title of Table
3I.3-A	3I-1	Plant Environment Data and Location Cross Reference Table of Figure Numbers
3I.3-1	3I-2	Thermodynamic Environment Conditions Inside primary Containment Vessel, Plant Normal Conditions
3I.3-2	3I-3	Thermodynamic Environment Conditions Inside RB (Secondary Containment), Plant Normal Operation Conditions
3I.3-3	3I-4	Thermodynamic Environment Conditions Inside RB (Outside Secondary Containment), Plant Normal Operating Conditions
3I.3-4	3I-5	Thermodynamic Environment Conditions Inside Control Building, Plant Normal Operating Conditions
3I.3-5	3I-6	Thermodynamic Environment Conditions Inside Turbine Building, Plant Normal Operation Conditions
3I.3-9	3I-7	Radiation Environment Conditions Inside Primary Containment Vessel, Plant Normal Operating Conditions
3I.3-10	3I-8	Radiation Environment Conditions Inside RB (Secondary Containment), Plant Normal Operation Conditions
3I.3-11	3I-9	Radiation Environment Conditions Inside RB (Outside Secondary Containment), Plant Normal Operating Conditions
3I.3-12	3I-10	Radiation Environment Conditions Inside Control Building, Plant Normal Operation Conditions
3I.3-13	3I.11	Radiation Environment Conditions Inside Turbine Building, Plant Normal Operating Conditions
3I.3-14	3I-12	Thermodynamic Environment Conditions Inside Primary Containment Vessel, Plant Accident Conditions
3I.3-15	3I-13	Thermodynamic Environment Conditions Inside RB (Secondary Containment), Plant Accident Conditions
3I.3-16	3I-14	Thermodynamic Environment Conditions Inside RB (Outside Secondary Containment), Plant Accident Conditions
3I.3-18	3I-15	Thermodynamic Environment Conditions Inside Control Building, Plant Accident Conditions
3I.3-19	3I-16	Radiation Environment Conditions Inside Primary Containment Vessel, Design-basis accident
3I.3-20	3I-17	Radiation Environment Conditions Inside RB Design-basis accident (Secondary Containment)
3I.3-21	3I-18	Radiation Environment Conditions Inside RB Design-basis accident Conditions (Outside Secondary Containment)
3I.3-22	3I-19	Radiation Environment Conditions Inside Control Building Design-basis accident Conditions
Figure 3I.2-1	Figure 3I-1	Zones in Primary Containment Vessel

In the DSER (SECY-91-153), the staff noted that the tables in Appendix 3I did not include the beta radiation dose rate and the integrated beta dose for applicable zones. In response, GE stated that accurate radiation environments should include consideration of the source term and the design and location and materials of construction of the equipment in the various environmental zones. GE further stated that while the source term is known, the COL applicant will determine the design, specific location, and materials of construction of various pieces of equipment. As a result, GE developed an ITAAC for this issue ("Table 3.73.11C: Equipment Qualification for Radiation"). The staff reviewed the ITAAC and, in the DFSER, determined that the acceptance criteria should be modified to state: "The maximum expected lifetime exposure for each piece of equipment within the scope of 10 CFR 50.49 shall not exceed the demonstrated qualified value as determined in accordance with the requirements of 10 CFR 50.49 paragraph (f)." This was DFSER Open Item 3.11.3-1. In the SSAR, GE provided revised tables for Appendix 3I that included beta radiation dose rates and integrated beta doses for applicable zones. The revised tables replace Tables 3I.3-9 through 3I.3-13 and Tables 3I.3-19 through 3I.3-22. The staff reviewed the revised tables and found them acceptable. This resolved DFSER Open Item 3.11.3-1.

In the DSER (SECY-91-153), the staff noted that the Appendix 3I tables did not identify whether the subject zone is environmentally mild or harsh and also did not list the typical equipment located in each zone. In response, GE proposed a change to SSAR Section 3.11.2 to define a mild environment as: "Mild environment is that which, during or after a design-basis event will at no time be significantly more severe than that existing during normal and abnormal events." The staff understands that IST is included as a normal or abnormal condition (IST is not an environmental qualification program requirement). This proposed definition is consistent with the requirements of 10 CFR 50.49. In the DFSER, the staff noted that it would verify the incorporation of the proposed definition for a mild environment. This was DFSER Confirmatory Item 3.11.3-2. GE has included this information in SSAR Section 3.11.2. This is acceptable and resolved DFSER Confirmatory Item 3.11.3-2. However, for current generation operating reactors, the staff's definition of what constitutes a mild radiation environment for electronic components, such as semi-conductors or any electronic component containing organic materials, is different from what it is for other equipment. The staffs's position is that a mild radiation environment for electronic equipment is a total integrated dose of < 10 Gy (10E3 R). For other equipment it is < 100 Gy (10E4 R). With the expected significant increase in the quantity and variety of electronic components in newer generation plants, the staff has increasing concerns about the efforts being made to ensure that these components are environmentally qualified and the capability of the component to be environmentally qualified. As a result, in the DFSER, the staff commented that GE should confirm that its position on the environmental qualification of electronic components is consistent with the staff's. This was DFSER Open Item 3.11.3-2. In response, GE addressed this item in the SSAR by stating that electronic equipment subject to radiation exposure in excess of 1000 R and other equipment in excess of 10,000 R is qualified in accordance with 10 CFR 50.49. This is acceptable and resolved DFSER Open Item 3.11.3-2. In addition, to respond to DSER Outstanding Issue 13, GE proposed changes to Tables 3I.3-1 through 3I.3-22 to include references to P&ID and IED drawings that will identify typical equipment for each zone. In the DFSER, the staff found these proposed changes acceptable but noted that the changes should be incorporated into the SSAR. This was DFSER Confirmatory Item 3.11.3-3. The proposed changes are incorporated into the SSAR. Therefore, DFSER Confirmatory Item 3.11.3-3 is resolved.

In the DSER (SECY-91-153), the staff noted that the environmental conditions during abnormal plant operational conditions were placed under abnormal/accident conditions in the Appendix 3I tables. Additionally, the staff was not certain whether GE considered the adverse environmental conditions resulting from abnormal events such as SRV discharges and loss of non-safety-related HVAC and their durations in developing the environmental data for these tables for applicable zones. In response, SSAR Amendment 14 revised the Appendix 3I tables to properly include these abnormal occurrences in the normal plant operating conditions that are used for determining the qualified life of the equipment required to be qualified. This is acceptable.

In the DSER (SECY-91-153), the staff noted that the Appendix 3I tables did not explicitly identify the limiting accident (e.g., high-energy line break such as MSL, reactor core isolation cooling, RHR, or CUW line break and DBLOCA inside the containment) for each zone (e.g., steam tunnel, RHR pump room) that results in the most severe environment, particularly thermal, in the zone. SSAR Section 3.11.1 states: "The environmental conditions shown in the Appendix 3I tables are upper-bound envelopes used to establish the environmental design and qualification bases of safety-related equipment. The upper-bound envelopes indicate that the zone data reflects the worse case expected environment produced by a compendium of accident conditions." The staff interprets that the SSAR considers a spectrum of break sizes and the mass and energy releases from the considered break sizes and thereby developed the environmental qualification profiles based on the most limiting combination of the considered break sizes. This is acceptable.

In the DSER (SECY-91-153), the staff noted that the Appendix 3I tables did not contain information on the environmental conditions resulting from spray or submergence or on the consequent wetting of equipment in applicable zones arising from piping failures nor the duration of the spray or submergence. In response, GE indicated in SSAR Section 3I.3.2.3 that containment spray may continue up to 100 days. In addition, GE modified SSAR Section 3I.3.2.3 by stating that "equipment will be qualified for submergence or will not be submerged except where submergence is mitigated by safety function performed by barrier separated redundant equipment." In the DFSER, the staff noted that during construction, the COL applicant should ensure that issues identified in Information Notice 89-63 related to flooding above the flood level and equipment wetting are addressed. This was DFSER COL Action Item 3.11.3-1. GE has included this information in Section 3.11.1 of the SSAR. This is acceptable.

In the DSER (SECY-91-153), the staff noted that the Appendix 3I tables did not contain (1) radiation environmental data under normal plant operating conditions for the radwaste building, outdoor area, and control building zones; (2) thermal environmental data under accident conditions for the radwaste building, service building zones, and outdoor area; and (3) radiation environmental data under accident conditions for the turbine building zones, radwaste building, service building zones, and outdoor area. In the DSER (SECY-91-153), the staff also indicated that if some of these areas identified are not expected to house any equipment required to be qualified and therefore not requiring environmental data, this should be stated. Subsequently, GE identified the various environmental zones within the scope of 10 CFR 50.49. However, the radiation environment that would result from normal operations and a design-basis accident was not determined. GE further commented that an accurate determination of radiation levels in the various harsh environmental zones requires consideration of specific equipment design details such as geometry and materials of construction and equipment location within each zone; therefore, radiation dose and dose rates should be determined by the COL applicant. This determination should also include the radiation contribution from recirculation fluid lines near the applicable areas. In the DFSER, the staff evaluated GE's position discussed above and stated that it would evaluate the relevant radiation zones on a plant-specific basis. This was DFSER COL Action Item 3.11.3-2. However, upon further evaluation, GE decided to include the relevant radiation zone data, including doses and dose rates, in the tables in Appendix 3I. This information is included in the SSAR, which is acceptable, and DFSER COL Action Item 3.11.3-2 is deleted.

In the DSER (SECY-91-153), the staff noted that the Appendix 3I tables did not contain sufficient information on thermal environmental conditions (e.g., duration of different conditions) in various zones under normal plant operating conditions to develop a meaningful time-based thermal environmental profile for the zones. GE provided a proposed table (Table 3I.3-A) and a proposed amendment to all the tables in Appendix 3I that contain thermodynamic environmental conditions for both normal operating conditions and design-basis accidents. In the DFSER, the staff noted that the proposed information would be sufficient to develop time-based profiles for the various identified zones. This was DFSER Confirmatory Item 3.11.3-4. GE has included this information in Appendix 3I of the SSAR. This is acceptable and resolves DFSER Confirmatory Item 3.11.3-4.

In the DSER (SECY-91-153), the staff noted inconsistencies in the units used to specify the pressures (e.g., kg/cm g, mm Aq). Consistent units are provided in the SSAR. This is acceptable.

In the DSER (SECY-91-153), the staff noted that the meaning of the statement "the pressure will be kept negative or positive" is not clear (see Note 2 to Tables 3I.3-3 through 3I.3-7). This statement was subsequently interpreted by the staff, in the DFSER, to mean that in various areas of the plant site, such as the control building, the RB, and the primary containment building, where the atmospheric pressure may be required to be negative that it will be maintained below 0.0 kPag (0.0 psig) and if the atmospheric pressure is required to be positive, that it will be maintained above 0.0 kPag (0.0 psig).

The staff noted in the DSER (SECY-91-153) that the integrated gamma accident dose in the primary containment for the ABWR was given as 6 x 10^5 Gy (6 x 10^7 rads), which is less than the typical value of about 2 x 10^6 Gy (2 x 10^8 rads) quoted in the safety analysis reports of several operating reactors (e.g., Perry: 2.7 x 10^6 Gy (2.7 x 10^8 rads); River Bend: 1.7 x 10^6 Gy (1.7 x 10^8 rads); Clinton: 2.0 x 10^6 Gy (2 x 10^8 rads); Nine Mile Point: 1.4 x 10^6 Gy (1.4 x 10^8 rads). It was not clear why the ABWR integrated gamma accident dose is lower than the corresponding doses quoted for several operating reactors. As a response to the above, GE's position, which was provided in Section 5.3.2.1.5 of SSAR Amendment 15, did not adequately address this issue. In the DFSER, the staff noted that, to resolve this issue, GE must fully explain why the ABWR integrated gamma accident dose is lower than the corresponding doses quoted for several operating reactors. This was DFSER Open Item 3.11.3-3. Subsequently, GE determined that the integrated gamma dose for an accident inside primary containment for ABWR is 2 x 10^6 Gy (2 x 10^8 rads). Table 3I.16 of Appendix 3I of the SSAR provides this information. The staff concludes in its engineering judgement that this is a reasonable value and is consistent with values determined at other BWRs. This is acceptable and resolved DFSER Open Item 3.11.3-3.

On the basis of its review of the tables in SSAR Appendix 3I the staff concludes that the tables in Appendix 3I are acceptable.

3.11.4 Adequacy of Interface Requirements

As a result of staff review of interface requirements, these requirements have been reclassified as ITAAC items that will be reviewed as part of the ITAAC program. In these instances COL applicants must (1) present a summary of environmental conditions and qualified conditions for each applicable item of equipment located in a harsh-environment zone in the system component evaluation work sheets as described in Table I-1 of GE Topical Report NEDE-24326-1-P and compile these sheets in their plant-specific environmental qualification documents and (2) record and maintain in an auditable file the results of all qualification tests for applicable equipment.

Additionally, although not identified as an interface requirement, the DFSER states that COL applicants should develop a surveillance and maintenance program for each applicable equipment item located in a mild-environment zone to ensure its operability during its design life. The vendors of equipment located in a mild environment are required to submit a certificate of compliance certifying that the subject equipment was qualified according to the requirements identified to ensure its capability to perform its safety-related function in its applicable environment. This was DFSER COL Action Item 3.11.3-3. SSAR Sections 3.11.2 and 3.11.6.3 include this action. This is acceptable.

3.11.5 Conclusions

On the basis of the evaluation discussed, the staff finds that the program for environmental qualification of electrical equipment for the ABWR standard design is in compliance with all the requirements of 10 CFR 50.49 and is, therefore, acceptable.

3.12 Piping Design

The DFSER "Piping Design" section, which was under Section 14.1.3.3, is revised and incorporated into a new section, Section 3.12, of this report. In the following, except for identification numbers for the DFSER open, confirmatory, COL action, and TS items, all other designations for subsections, paragraphs, and the like are reassigned and keyed to Section 3.12 of this report, as appropriate.

3.12.1 Introduction

This section provides the staff's safety evaluation of GE's design acceptance criteria (DAC) approach for the ABWR piping design. The staff used the SRP guidelines to evaluate the piping design information in the ABWR 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 only to the ASME Boiler and Pressure Vessel Code Class 1, 2, and 3 piping and supports, but also 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 of the adequacy of the ABWR piping design analysis methods, design procedures, acceptance criteria, and related ITAAC that are to be used for the completion and verification of the ABWR piping design is provided in the following sections of this report. The staff's evaluation includes

- 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
- LBB approach applicable to the ABWR
- generic piping design ITAAC

The staff must arrive at a final safety determination that, if the COL applicant successfully completes the piping design and analyses and the ITAAC as required by 10 CFR Part 52, using the design methods and acceptance criteria discussed herein, there will be adequate assurance that the piping systems will perform their safety-related functions under all postulated combinations of normal operating conditions, system operating transients, postulated pipe breaks, and seismic events.

3.12.2 Codes and Standards

GDC 1 requires that structures, systems, and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function. 10 CFR 50.55a requires that systems and components of boiling and pressurized water-cooled nuclear power reactors must meet the requirements of the ASME Code. It specifies the latest edition and addenda endorsed by the NRC and any limitations. RGs 1.84 and 1.85 list ASME Code Cases that the NRC staff finds acceptable.

In SSAR Tables 1.8-21 and 3.2-3, GE identified the ASME Code, Section III, and the specific edition and addenda that will be used for the design of ASME Code, Class 1, 2, and 3 pressure retaining components. In SSAR Table 5.2-1, the Code Cases that may be used are also identified.

3.12.2.1 ASME Boiler and Pressure Vessel Code

For the ABWR design certification, GE has established that the ASME Boiler and Pressure Vessel Code, Section III, will be used for the design of ASME Code Class 1, 2, and 3 pressure retaining components and their supports. The specific edition and addenda are provided in SSAR Tables 1.8-21 and 3.2-3. The ASME Code is considered Tier 1 information; however, the specific edition and addenda are considered Tier 2 information. The specific edition and addenda are considered Tier 2 information because of the continually evolving technical nature associated with the design and construction practices (including inspection and examination techniques) of the Code. Fixing a specific edition and addenda during the design certification stage may result in inconsistencies between design and construction practices during the detailed design and construction stages. The ASME Code involves a consensus process to reflect the evolving design and construction practices of the industry. Although the reference to a specific edition of the Code for the design of ASME Code class components and their supports is suitable to reach a safety finding during the design certification stage, the construction practices and examination methods of an updated Code that would be effective at the COL application stage must be consistent with the design practices established at the design certification stage.

The staff finds that the specification of the ASME Code as Tier 1 information and the specific edition and addenda as Tier 2 information is appropriate because it would provide the means for the COL applicant to revise or supplement the referenced Code edition with portions of the later Code editions and addenda needed to ensure consistency between the design for the ABWR pressure retaining components and their supports and construction practices. In this manner, the updated reference Code to be used at the time of the COL application is ensured to be consistent with the latest design, construction, and examination practices at that time. However, where the staff finds that there may be a need to specify certain design parameters from a specific Code edition or addenda during its design certification review, particularly when that information is of importance to establish a significant aspect of the design or is used by the staff to reach its final safety determination, such considerations, if necessary, are reflected in the various sections of this safety evaluation.

Therefore, all ASME Code Class 1, 2, and 3 pressure retaining components and their supports must be designed in accordance with the requirements of ASME Code, Section III, using the specific edition and addenda provided in the ABWR SSAR. However, the COL applicant should also ensure that the design is consistent with the construction practices (including inspection and examination methods) of the ASME Code edition and addenda as endorsed in 10 CFR 50.55a in effect at the time of COL application. The portions of the later Code editions and addenda must be identified to the NRC staff for review and approval with the COL application. This was DFSER COL Action Item 14.1.3.3.2.1-1. GE has included this action in SSAR Section 3.9.7.4, which is acceptable.

3.12.2.2 ASME Code Cases

The only acceptable ASME Code cases that may be used for the design of ASME Code Class 1, 2, and 3 piping systems in the ABWR standard plant are those either conditionally or unconditionally approved in RGs 1.84 and 1.85 in effect at the time of design certification as listed below. However, the COL applicant may submit with its COL application for staff review and approval future code cases that are endorsed in RGs 1.84 and 1.85 at the time of COL application provided they do not alter the staff's safety findings on the ABWR certified design.

- In RG 1.84, the staff has conditionally endorsed ASME Code Case N-411, "Alternative Damping Values for Response Spectra Analysis of Classes 1, 2, and 3 Piping, Section III, Division 1." This Code Case is acceptable for the ABWR. The acceptability of the Code Case and its application is further discussed in Section 3.12.5.4 of this report.

Other ASME Code Cases requested by GE that are applicable to the ABWR piping and support design are listed below.

- ASME Code Case N-71-15, "Additional Materials for Subsection NF, Classes 1, 2, 3, and MC Component Supports Fabricated by Welding, Section III, Division 1." This Code Case has been endorsed by the staff in RG 1.85.

- ASME Code Case N-122, "Stress Indices for Structure Attachments, Class 1, Section III, Division 1." This Code Case has been endorsed by the staff in RG 1.84.

- ASME Code Case N-247, "Certified Design Report Summary for Component Standard Supports, Section III, Division 1, Classes 1, 2, 3 and MC." This Code Case has been endorsed by the staff in RG 1.84.

- ASME Code Case N-249-9, "Additional Material for Subsection NF, Classes 1, 2, 3 and MC Component Supports Fabricated Without Welding, Section III, Division 1." This Code Case has been endorsed by the staff in RG 1.85.

- ASME Code Case N-309-1, "Identification of Materials for Component Supports, Section III, Division 1." This Code Case has been endorsed by the staff in RG 1.84.

- ASME Code Case N-313, "Alternate Rules for Half-Coupling Branch Connections, Section III, Division 1." This Code Case has been endorsed by the staff in RG 1.84.

- ASME Code Case N-316, "Alternate Rules for Fillet Weld Dimensions for Socket Welded Fittings, Section III, Division 1, Class 1, 2, 3." This Code Case has been endorsed by the staff in RG 1.84.

- ASME Code Case N-318-3, "Procedure for Evaluation of the Design of Rectangular Cross Section Attachments on Class 2 or 3 Piping, Section III, Division 1." This Code Case has been conditionally endorsed by the staff in RG 1.84 and is discussed further in Section 3.12.5.16 of this report.

- ASME Code Case N-319, "Alternate Procedure for Evaluation of Stress in Butt Weld Elbows in Class 1 Piping, Section III, Division 1." This Code Case has been endorsed by the staff in RG 1.84.

- ASME Code Case N-391, "Procedure for Evaluation of the Design of Hollow Circular Cross Section Welded Attachments on Class 1 Piping, Section III, Division 1." This Code Case has been endorsed by the staff in RG 1.84.

- ASME Code Case N-392, "Procedure for Evaluation of the Design of Hollow Circular Cross Section Welded Attachments on Classes 2 and 3 Piping, Section III, Division 1." This Code Case has been endorsed by the staff in RG 1.84.

- ASME Code Case N-393, "Repair Welding Structural Steel Rolled Shaped and Plates for Component Supports, Section III, Division 1." This Code Case has been endorsed by the staff in RG 1.84.

- ASME Code Case N-414, "Tack Welds for Class 1, 2, 3 and MC Components and Piping Supports." This Code Case has been endorsed by the staff in RG 1.84.

- ASME Code Case N-430, "Requirements for Welding Workmanship and Visual Acceptance Criteria for Class 1, 2, 3 and MC Linear-Type and Standard Supports." This Code Case has been endorsed by the staff in RG 1.84.

All of the above Code Cases are listed in Table 5.2-1 of the SSAR. In addition, in Sections 3.9.3.4 and 3.9.3.5 of the SSAR, Code Case N-476, "Class 1, 2, & 3, and MC Linear Component Supports - Design Criteria for Single Angle Members, Section III, Division 1, Subsection NF," is referenced as augmenting ASME Subsection NF rules for the design of component supports. As stated in Section 3.9.3.3 of this report, the staff finds this Code Case acceptable. Therefore, the staff concludes that, since all of these Code Cases either meet the guidelines of RGs 1.84 or 1.85, or have been reviewed and endorsed by the staff, they are acceptable for use on the ABWR design.

3.12.2.3 Design Specifications

ASME Code, Section III, requires that a design specification be prepared for Class 1, 2, and 3 components such as pumps, valves, and piping systems. The design specification is intended to become a principal document governing the design and construction of these components and should specify loading combinations, design data, and other design data inputs. The Code also requires a design report for ASME Code, Class 1, 2, and 3 piping and components. In the SSAR, GE committed to construct all safety-related components, such as vessels, pumps, valves and piping systems, to applicable requirements of the ASME Section III. During its review of the SSAR, the staff reviewed selected documents related to design specifications and design reports. Those documents were not specifically for the ABWR, but were provided by GE and reviewed by the staff as a demonstration of how design specifications and design reports will be prepared for ABWR plants. The staff determined that the demonstration documents, with modifications, would meet Code requirements. However, because the documents were not specifically for the ABWR, they would have to be modified before the staff can conclude that the design specification and design report requirements in ASME Code, Section III, Subsection NCA, have been met. In order for the staff to reach this conclusion, the COL applicant should submit representative design documents (e.g., design specifications) for NRC staff review as part of the COL application (DFSER COL Action Items 3.9.3.1-2 and 14.1.3.3.2.3-1). SSAR Section 3.9.7 states that COL applicants referencing the ABWR design will make available to the staff design specifications and design reports required by the ASME Code for vessels, pumps, valves, and piping systems for the purpose of audit. This is acceptable.

3.12.2.4 Conclusions

On the basis of these discussions and the evaluation of SSAR Section 3.9.3.1 and Tables 1.8-21, 3.2-3, and 5.2-1, the staff concludes that the piping systems important to safety are designed to quality standards commensurate with their importance to safety. The staff's conclusion is based on the following:

(a) GE satisfies the requirements of GDC 1 and 10 CFR 50.55a by specifying appropriate codes and standards for the design and construction of safety-related piping and pipe supports, and

(b) GE identified ASME Code Cases that may be applied to ASME Code, Class 1, 2, and 3 piping and pipe supports, which are acceptable to the staff.

3.12.3 Analysis Methods

The staff reviewed the information in SSAR Section 3.9.1 related to the design transients and methods of analysis used for all seismic Category I piping and pipe supports designated as ASME Code Class 1, 2, and 3 under ASME Code, Section III, as well as those not covered by the Code. It reviewed the assumptions and procedures used for the inclusion of transients in the design and fatigue evaluation of ASME Code Class 1 and core support components. It also reviewed the computer programs used in the design and analysis of seismic Category I components and their supports, as well as experimental and inelastic analytical techniques.

3.12.3.1 Experimental Stress Analysis

SSAR Section 3.9.1.3 identifies several components for which experimental stress analysis is performed in conjunction with analytical evaluation. Such components in the piping area include the piping seismic snubbers and pipe whip restraints. The staff's evaluation of the experimental stress analysis methods is discussed in Section 3.9.1 of this report. The staff's evaluation of the analysis methods used to qualify these components is discussed further in Sections 3.12.6 and 3.12.7 of this report.

3.12.3.2 Modal Response Spectrum Method

GE performed system and subsystem analyses on an elastic basis. Modal response spectrum and time history methods form the basis for the analyses of all major seismic Category I piping systems and components. SSAR Section 3.7.3.8 describes the piping dynamic analysis procedure using the modal response spectrum method. First a mathematical model is constructed to reflect the dynamic characteristics of the piping system. The mode shapes and natural frequencies of the piping model are computed. Using a given direction of earthquake motion, the modal participation factors for each mode are calculated. Using the appropriate response spectrum curve, the spectral accelerations for each mode are determined. For a piping system supported at points with different dynamic excitations, an enveloped response spectrum of all attachment points is used. From the mode shapes, participation factors and spectral accelerations of each mode, the modal responses are calculated. They include the modal forces, shears, moments, stresses and deflections. For a given direction, the modal responses are combined in accordance with the methods described in SSAR Section 3.7.3.7.

The modal response calculations are performed for each of the three earthquake directions (two horizontal and the vertical). The total seismic response from the simultaneous application of the three-directional components of earthquake loading are obtained by combining the maximum codirectional responses of each of the three components by the square-root-of-the-sum-of-squares (SRSS) method as described in SSAR Section 3.7.3.6.

For piping systems that are anchored and restrained to floors and walls of buildings that have differential movements during a seismic event, additional forces and moments are induced in the system. Additional static analyses are performed to determine these loads as described in SSAR Section 3.7.3.8.1.8. The maximum differential displacements are applied to the piping anchors and restraints. Three analyses are performed: two in the horizontal directions and one in the vertical direction. The resulting stresses are placed in the secondary stress category because they are displacement induced and self-limiting. These secondary loads are combined with the primary (inertia) loads by the SRSS method.

The staff reviewed the SSAR description of the modal response spectrum method and found that it is consistent with the applicable guidelines in SRP Section 3.9.2 and is acceptable.

3.12.3.3 Independent Support Motion Method

As an alternative to the enveloped response spectrum method, the independent support motion (ISM) analysis method may be used. The theory and development of the governing equations of motion for this method are presented in SSAR Subsection 3.7.2.1.4. Additional requirements associated with the application of this method are described in the SSAR Subsection 3.7.3.8.1.10, "Multiply Supported Equipment and Components with Distinct Inputs." This section states that when this method of analysis is used, the following conditions must be met: (1) ASME Code Case N-411-1 damping is not used; (2) a support group is defined by supports which have the same time history input. This usually means all supports located on the same floor, or portions of a floor, of a structure; and (3) The responses from motions of supports in two or more different groups are combined by the SRSS procedure.

The staff finds that contingent upon these conditions, this alternative to the enveloped response spectrum method is acceptable. More details of the evaluation are discussed in Section 3.9.2.2. of this report.

3.12.3.4 Time-History Method

A time history analysis may be performed using either the modal superposition method or the direct integration method. The modal superposition method is described in SSAR Subsection 3.7.2.1.2. This approach involves the calculation and utilization of the natural frequencies, mode shapes, and appropriate damping factors of the particular system toward the solution of the equations of dynamic equilibrium. The orthogonality of the mode shapes is used to effect a coordinate transformation of the displacements, velocities, and accelerations such that the response in each mode is independent of the response of the system in any other mode. Through this transformation, the problem becomes one of solving n independent differential equations rather than simultaneous differential equations. As long as the system is linear, the principle of superposition holds and the total response of the system oscillating simultaneously in n modes may be determined by direct addition of the responses of the individual modes.

The direct integration method is described in SSAR Section 3.7.3.1. This method involves the direct step-by-step numerical integration of the equations of motion and does not require the solution of an eigenvalue problem. The response in all modes is calculated simultaneously. The numerical integration time step, Δt, must be sufficiently small to accurately define the dynamic excitation and to render stability and convergence of the solution up to the highest frequency of significance. The integration time step is considered acceptable when smaller time steps introduce no more than a 10-percent error in the total dynamic response. For most of the commonly used integration methods, the maximum time step is limited to one-tenth of the smallest period of interest, which is generally the reciprocal of the cutoff frequency. In direct integration analysis, the damping is input in the form of α and β damping constants, which give the percentage of critical damping, λ as a function of the circular frequency, ω.

The total seismic response is predicted by combining the responses from the three orthogonal components (two horizontal and one vertical) of the earthquake. When separate time-history analyses are performed for each directional component, the combined response may be obtained by taking the SRSS of the maximum codirectional responses caused by each component. As an alternative, the combined response may be obtained by algebraically adding the codirectional responses from each analysis at each time step or the total response may be obtained directly by applying the three component motions simultaneously in one analysis. When either alternative method is used, the three component motions must be mutually statistically independent.

When the time-history method of analysis is used, the time-history data is broadened plus and minus 15 percent of Δt in order to account for modeling uncertainties. For loads such as safety-relief valve blowdown, tests have been performed that confirm the conservatism of the analytical results. Therefore, for these loads, the calculated force time histories are not broadened.

The staff reviewed the SSAR descriptions of the modal superposition and the direct integration time-history analysis methods and found them to be in compliance with the applicable guidelines of SRP Section 3.9.2 and acceptable.

3.12.3.5 Inelastic Analysis Method

GE has not provided any information on the use of inelastic analysis methods for the ABWR piping analyses. If inelastic methods are to be used in any ABWR piping analyses, then the staff requires that the details of the inelastic method and its acceptance criteria, as well as the scope and extent of its application, be submitted to the staff for approval prior to its use by a COL applicant.

3.12.3.6 Small-Bore Piping Method

In the DFSER, the staff noted that GE had not provided any specific information about the method to be used for the structural design of small-bore piping systems and instrumentation lines in the ABWR plant. The staff requested that this information be included in the SSAR. This was DFSER Open Items 3.9.2.2-5 and 14.1.3.3.3.6-1. SSAR Section 3.7.3.8.1.9, "Design of Small Branch and Small Bore Piping," discusses the use of small-bore piping handbooks in lieu of performing piping analysis for piping 50.8 mm (2 in.) and less nominal pipe size, and small branch lines 50.8 mm (2 in.) and less nominal pipe size. It states that (a) the small-bore piping handbook must be currently accepted by the regulatory agency for use on equivalent piping at other nuclear power plants at the time of application; (b) when the handbook meets the purpose of the Design Report, it must meet all of the ASME requirements for a piping design report for piping and its supports; and (c) formal documentation exists showing that piping designed and installed in accordance with the handbook is conservative in comparison to results from a detailed stress analysis for all loads and load combinations, that it is not less reliable owing to loss of flexibility or excessive supports, and that it satisfies required clearances around sensitive components. The piping handbook methodology will not be applied when specific information is needed on pipe stresses, cumulative usage factors, accelerations, or break locations.

The staff reviewed the methodology described in SSAR Section 3.7.3.8.1.9, which states that the static and dynamic analysis methods defined in Section 3.7.3 of the SSAR will be used to provide the formal documentation showing that piping designed and installed to the small bore piping handbook is conservative in comparison to a detailed stress analysis. The staff finds this acceptable, and DFSER Open Items 3.9.2.2-5 and 14.1.3.3.3.6-1 are resolved.

3.12.3.7 Non-Seismic/Seismic Interaction (II/I)

All non-seismic Category I piping (or other systems and components) should be isolated from seismic Category I piping. This isolation may be achieved by designing a seismic constraint or barrier or by locating the two sufficiently apart to preclude any interaction. If it is impractical to isolate the seismic Category I piping system, the adjacent non-seismic Category I system should be evaluated to the same criteria as the seismic Category I system.

For non-seismic Category I piping systems attached to seismic Category I piping systems, the dynamic effects of the non-seismic Category I system should be considered in the analysis of the seismic Category I piping. In addition, the non-seismic Category I piping from the attachment point to the first anchor should be evaluated to ensure that, under all loading conditions, it will not cause a failure of the seismic Category I piping system. Section 3.7.3.13 in the SSAR contains criteria that are consistent with these staff positions and applicable portions of SRP 3.9.2 and RG 1.29, "Seismic Design Classification," Revision 3, and is therefore, acceptable.

3.12.3.8 Main Steamline and Bypass Line in the Turbine Building

For the ABWR plant design, GE eliminates the main steam isolation valve leakage control system and relies on the use of an alternative leakage path that takes advantage of the large volume and surface area in the main steam piping, drain line, bypass line, and condenser to hold up and plate out the release of fission products following core damage. In this manner, the main steam piping, drain line, bypass line, and condenser will be used to mitigate the consequences of an accident and will be required to remain functional during and after a SSE.

For this reason, the staff's position is that the main steam piping beyond the second outermost isolation valve and up to the seismic interface restraint and the connecting branch lines up to the first normally closed valve should be classified as QG B (Safety Class 2) and seismic Category I. The MSL from the seismic interface restraint up to but not including the turbine stop valve (including branch lines to the first normally-closed valve) should be classified as QG B and inspected in accordance with the applicable portions of ASME Code, Section XI, but may be classified as non-seismic Category I if it has been analyzed, using a dynamic seismic analysis method to demonstrate its structural integrity under SSE loading conditions. However, all pertinent QA requirements of Appendix B to 10 CFR Part 50 are applicable to ensure that the quality of the piping material is commensurate with its importance to safety during normal operational, transient, and accident conditions. To ensure integrity of the remainder of the alternative leakage path, the main steam bypass line, although it is not required to be classified as safety-related or as seismic Category I, the line from the first valve up to the condenser inlet, the main steam drain line from the first valve to the condenser, and the main steam piping between the turbine stop valve and the turbine inlet should be analyzed, using a dynamic seismic analysis, to demonstrate its structural integrity under SSE loading conditions. This was DSER Outstanding Issue 3 in SECY-91-153 and DFSER Confirmatory Item 14.1.3.3.3.8-1.

SSAR Section 3.2.5.3, "Main Steamline Leakage Path," defines the piping and components that make up the main steam leakage path and provides their classifications and requirements for dynamic seismic analysis. It also states that a plant-specific walkdown will be conducted to confirm that the as-built main steam piping, bypass lines to the condenser, and the main condenser are not compromised by non-seismically designed systems, structures, and components. From its review, the staff review concludes that the SSAR adequately reflects these staff positions as described. On the basis of the evaluation as discussed and the evaluation reported in more detail in Section 3.2.1 of this report, DSER Outstanding Issue 3 and DFSER Confirmatory Item 14.1.3.3.3.8-1 are resolved.

Lastly, the main steam piping, drain line, and bypass line in the turbine building should be protected from the collapse of any non-seismic Category I structure in the event of an SSE. As a confirmatory measure, a plant-specific walkdown should be performed before operation to assess the potential failures of non-seismically designed SSCs overhead, adjacent to, and attached to the alternative leakage path (i.e., the main steam piping, by-pass line, and the main condenser). In the DSER, the staff noted that this walkdown should be performed as a part of the ITAAC verification of non-seismic/seismic interaction. This was later identified in the DFSER as Open Item 3.2.1-3 and Confirmatory Item 14.1.3.3.3.8-2. Upon further consideration, the staff found that the design detail and as-built and as-procured information for non-seismically designed SSCs in the turbine building as they affect the alternative leakage function of the main steam, bypass, and drain lines, and the main condenser are not required for design certification and the staff will review the spacial relationship between these SSCs and the main steam piping, bypass, and drain lines, and main condenser to ensure compliance with GDC 2. Subsequently, GE revised the SSAR and added SSAR Section 3.2.5.3 that contains a commitment to perform plant-specific walkdowns consistent with this staff position. This commitment to perform walkdowns is acceptable. Therefore, DFSER Open Item 3.2.1-3 and DFSER Confirmatory Item 14.1.3.3.3.8-2 are withdrawn and resolved. The resolution of this issue is also discussed in Section 3.2.1 of this report.

3.12.3.9 Buried Piping

SSAR Section 3.7.3.12 originally outlined the criteria that will be used in the analysis of buried seismic Category I piping systems. These criteria conformed to the applicable guidelines in SRP Section 3.9.2. However, GE did not give any details on how the criteria are to be applied in the design of buried piping. In the DFSER, the staff requested that the SSAR be revised to address, as a minimum, (1) the maximum bearing loads, (2) the categorization of seismic stresses in the Code evaluation, and (3) the allowable stress limits for the piping. This was DFSER Open Items 3.9.2.2-7 and 14.1.3.3.3.9-1.

SSAR Section 3.7.3.12 states that all underground seismic Category I piping systems are installed in tunnels. The tunnels are analyzed as buried structures. The piping analysis is performed using one of the methods described in Section 3.7.3 of the SSAR. Because the SSAR states that the buried piping systems will not be in direct contact with the soil, the staff concludes that the information requested in the DFSER is no longer applicable. Therefore, DFSER Open Items 3.9.2.2-7 and 14.1.3.3.3.9-1 are withdrawn and closed.

3.12.3.10 ASME Code, Section III, Appendix N

The staff has not endorsed the use of ASME Code, Section III, Appendix N, which is a non-mandatory appendix that is still evolving and does not currently agree with some regulatory positions. Therefore, for the ABWR piping design, if the methodology in Appendix N is not consistent with regulatory positions discussed herein, the regulatory positions shall be used.

3.12.3.11 Conclusions

On the basis of the evaluations in Section 3.12.3, the staff concludes that the analysis methods to be used for all seismic Category I piping systems as well as non-seismic Category I piping systems that are important to safety are acceptable. The analysis methods utilize piping design practices that are commonly used in the industry and provide an adequate margin of safety to withstand the loadings as a result of normal operating, transient, and accident conditions.

3.12.4 Piping Modeling

GDC 2 requires that components important to safety should be designed to withstand effects of natural events including earthquakes. 10 CFR Part 50, Appendix B requires that design quality should be controlled for ensuring structural and functional integrity of seismic Category I components. For determining design adequacy, each piping system is idealized as a mathematical model and dynamic analysis using computer programs is performed. Modeling techniques should be in conformance with generally recognized engineering practice and computer programs should be verified per one or more methods suggested in SRP Section 3.9.1. A piping benchmark program described in NUREG/CR-6049 is also provided by the NRC for aiding the verification process.

SSAR Sections 3.7.3.3 and 3.9.2.2 describe piping modeling techniques and SSAR Section 3.9.1.2 discusses quality control of computer programs and computer results.

3.12.4.1 Computer Codes

This section addresses the computer codes to be used to analyze piping systems in the ABWR design. SSAR Appendix 3D includes all computer programs for static and dynamic analyses to determine the structural and functional integrity of seismic Category I and non-seismic Category I items. Design control measures to verify the adequacy of the design of safety-related components are required by Appendix B to 10 CFR Part 50. SSAR Section 3.9.1.2 states that the quality of the programs and the computer results are controlled either by GE or by outside computer program developers. In addition, the programs are verified by one or more of the methods recommended in SRP Section 3.9.1.

To review GE's computer verification program for ABWR piping modeling, the staff performed an independent confirmatory piping stress analysis of representative piping systems in the ABWR standard plant. The purpose of this analysis was to verify the adequacy of the computer program used by GE to generate the sample piping analyses that were audited by the staff on March 23 through 26, 1992, at GE's offices in San Jose, California. These were DFSER Open Items 3.9.1-2, 14.1.3.3.4.1-1 and 14.1.3.3.4.3-1. The results of the confirmatory analysis verify that this computer program is adequate with acceptable accuracy. The staff concludes that the computer program verification process for the ABWR is acceptable. Therefore, DFSER Open Items 3.9.1-2, 14.1.3.3.4.1-1, and 14.1.3.3.4.3-1 are resolved.

3.12.4.2 Dynamic Piping Model

For the dynamic analysis of seismic Category I piping, each system is idealized as a mathematical model consisting of lumped masses interconnected by elastic members. The stiffness matrix for the piping system is determined, using the elastic properties of the pipe. This includes the effects of torsional, bending, shear, and axial deformations as well as change in stiffness as a result of curved members.

The staff reviewed the method for selecting the number of masses or degrees of freedom in the piping mathematical model to determine its dynamic response. GE's internal documents that were audited by the staff on March 23 through 26, 1992, showed pipe and fluid masses are lumped at nodes that are selected to coincide with the locations of large masses (e.g., valves, pumps, and tanks) and with locations of significant geometric changes (e.g., pipe elbows, reducers, and tees). Additional mass points are selected to ensure that the spacing between any two adjacent piping nodes and masses is no greater than an idealized value. This value corresponds to the length of a simply supported beam with a uniformly distributed mass whose undamped natural frequency is equal to the cutoff frequency. Since this approach, in effect, would capture all modes up to the cutoff frequency, the staff finds that the ABWR method for locating mass points is acceptable. In the DFSER, the staff requested that the SSAR be revised to reflect this approach as described (DFSER Confirmatory Items 3.9.2.2-2 and 14.1.3.3.4.2-1). SSAR Subsection 3.7.3.3.1.2, "Selection of Mass Points," provides more detailed mass point selection criteria for dynamic piping models. The staff reviewed these criteria and determined that the description in the SSAR reflects this approach as discussed and is, therefore, acceptable; therefore, DFSER Confirmatory Items 3.9.2.2-2 and 14.1.3.3.4.2-1 are resolved.

The effect of pipe support stiffness on the piping response must be considered in the analytical model. Supports must be modeled in accordance with the SSAR. If supports are not modeled as stated in the SSAR, justification will be provided to validate the stiffness values used in the piping model. The justification should include verification that the generic values are representative of the types of pipe supports used in the piping system. This alternative approach to use generic stiffness values and its bases should be submitted to the staff for review and approval before use. This was DFSER COL Action Item 14.1.3.3.4.2-1. SSAR Section 3.7.5.3, "Piping Analysis, Modeling of Supports," states that the COL applicant will provide the information requested in this action item. This is acceptable.

Additionally, because the amplified response spectra are generally specified at discrete building node points, any additional flexibility between these points and the pipe support (e.g., supplementary steel) also should be addressed. In the DFSER, the staff requested that the

SSAR be revised to incorporate this information. This was DFSER Open Item 3.9.2.2-2 and DFSER Confirmatory Item 14.1.3.3.4.2-2. SSAR Section 3.7.3.3.1.8, "Response Spectra Amplification at Support Attachment Points," states that the drywell equipment and pipe support structures (DEPSS) should meet the criteria given in SSAR Subsection 3.7.3.3.4. It further states that, if this criteria cannot be met, the COL applicant will generate the amplified response spectra at piping attachment points considering the DEPSS as part of the structure, using the dynamic analysis method described in SSAR Section 3.7.2, or will analyze the piping systems considering the DEPSS as part of the pipe support. The staff reviewed this clarification and determined that it is acceptable; therefore, DFSER Open Item 3.9.2.2-2 and DFSER Confirmatory Item 14.1.3.3.4.2-2 are resolved.

If piping terminates at non-rigid equipment (e.g., tanks, pumps, or heat exchangers), the analytical piping model should consider the flexibility and mass effects of this equipment. In the DFSER, the staff requested that the SSAR be revised to address how the flexibility and masses of equipment attached to the piping are to be modeled. This was DFSER Open Item 3.9.2.2-3 and DFSER Confirmatory Item 14.1.3.3.4.2-3. SSAR Subsection 3.7.3.3.1.6, "Modeling of Piping Supports," states that stiffnesses of supporting structures are included in the piping analysis model. Anchors at equipment such as tanks, pumps and heat exchangers are modeled with calculated stiffness properties. It also states that mass effects will be included for equipment that have a fundamental frequency of less than 60 Hz. A simplified model of the equipment will be included in the piping system model. The staff concludes that this issue is adequately addressed; therefore, DFSER Open Item 3.9.2.2-3 and DFSER Confirmatory Item 14.1.3.3.4.2-3 are resolved.

3.12.4.3 Piping Benchmark Program

To verify the adequacy of the computer program used by the COL applicant to complete the ABWR piping system design and analyses, the NRC staff established mathematical models of representative piping systems in the ABWR standardized plant that were used in a piping analysis benchmark program. The mathematical models are based on the dynamic piping model and on the piping stress analysis criteria described in Section 3.12.4.2 and Section 3.12.5, respectively, of this report. The benchmark program verifies the adequacy of linear-elastic, dynamic piping analysis methods using the enveloped response spectrum method, multiple response spectrum method, and time-history method of analyses.

The benchmark program essentially consists of constructing mathematical models of the ABWR feedwater piping system inside containment and an SRV discharge line inside the suppression pool wetwell area, using the COL applicant's computer program. The piping configuration for the piping models are described in NUREG/CR-6049, "Piping Benchmark Problems for the GE ABWR," and include piping dimensions, pipe sizes, materials, valve weights, support and anchor stiffnesses, and support locations. The piping input parameters for the benchmark analyses also are specified in the piping benchmark program and include damping values, loading definitions, and load combinations.

When the COL applicant's dynamic piping analyses are completed, the results of the analyses must be compared with the results of the benchmark problems provided in the piping benchmark program. The piping analysis results to be compared and evaluated include the system modal frequencies, the maximum pipe moments, the maximum support loads and equipment reactions, and the maximum pipe deflections. The acceptance criteria or range of acceptable values are specified in the piping benchmark program and must be satisfied. The COL applicant must document and submit any deviations from these values as well as the justification for such deviations to the NRC staff for review and approval before initiating final certified piping analyses. The piping benchmark program was DFSER COL Action Item 14.1.3.3.4.3-1. In SSAR Section 3.9.1.2 states that the COL applicant will provide this information as discussed. This is acceptable. The benchmark program provides assurance that the computer program used to complete the ABWR piping design and analyses produces results that are consistent with results considered acceptable to the NRC staff.

3.12.4.4 Decoupling Criteria

When analyzing piping systems, the size of the mathematical model might exceed the capacity of the computer program if large and small-bore piping are included. Thus, the small-bore branch lines are generally decoupled from the large-bore main piping. Originally, the SSAR did not provide any criteria for the decoupling of the piping systems in the analysis model. However, in a letter to the NRC dated February 24, 1992, GE provided the decoupling criteria in a GE document entitled, "ABWR SSAR Main Steam, Feedwater and SRVDL Piping Systems Design Criteria and Analysis Methods (draft)," Revision 0, dated February 1992. This document stated that if the ratio between pipe diameters of the branch line to main line is less than one-third, the branch line can be excluded from the piping model of the main line.

In the DFSER, the staff requested that GE provide the basis for this one-third ratio and revise the SSAR to include additional information on how mass and other effects are accounted for in the analysis (DFSER Open Items 3.9.2.2-4 and 14.1.3.3.4.4-1). SSAR Sections 3.7.3.3.1.3 and 3.7.3.8.1.9 provide more detailed decoupling criteria. The basic criterion was changed from a ratio of run to branch pipe diameter of 3 to 1, or more, to a moment of inertia ratio of 25 to 1, or more. In addition, small branch lines shall be designed with no concentrated masses, such as valves, in the first one-half span length from the main run pipe; and with sufficient flexibility to prevent restraint of movement in the main run pipe. Quantitative requirements for assessing the adequacy of the flexibility were provided. The staff reviewed the revised decoupling requirements and found them acceptable; therefore, DFSER Open Items 3.9.2.2-4 and 14.1.3.3.4.4-1 are resolved.

3.12.4.5 Conclusions

On the basis of these discussions and evaluation of SSAR Sections 3.7.3.3 and 3.9.2.2, the staff concludes that design control measures are acceptable to ensure quality of computer programs and design methods. The staff's conclusion is based on the following:

(1) GE satisfies the requirements of GDC 2 by providing criteria for the seismic design and analysis of all seismic Category I piping and pipe supports using prescribed modeling techniques and design methods that are in conformance with generally recognized engineering practice.

(2) GE meets Appendix B to 10 CFR Part 50 by demonstrating the applicability and validity of the computer programs for performing piping seismic analysis.

(3) Computer programs to be used by the COL applicant to complete its analyses of the ABWR piping systems will be verified and validated using the NRC staff's piping benchmark program.

3.12.5 Pipe Stress Analysis Criteria

3.12.5.1 Seismic Input (Envelope vs. Site-Specific Spectra)

The ABWR standard plant is designed for an SSE ground motion defined by a RG 1.60 response spectrum anchored to a peak ground acceleration of 0.3*g*. Amplified building response spectra are generated for the ABWR standard plant to account for varying soil properties in the United States by enveloping 14 site conditions. GE has proposed that the COL applicant use these enveloping amplified building response spectra provided in the SSAR to complete the design and analyses of the ABWR piping systems.

The staff recognizes that the enveloping amplified building response spectra for the ABWR plant contain conservatisms that may be excessive for certain specific site conditions. If amplified building response spectra are generated using site-dependent properties, then the approach and method used must be submitted to the staff for review and approval as part of the COL application. The method used to generate the amplified building response spectra should be consistent with the method described in the SSAR Section 3.7.2 as approved by the staff. This was DFSER COL Action Item 14.1.3.3.5.1-1. Upon further consideration, the staff determined that a COL Action Item is not appropriate because this is only an option for the COL applicant. Therefore, DFSER COL Action Item 14.1.3.3.5.1-1 is deleted.

3.12.5.2 Design Transients

SSAR Table 3.9-1 lists the design transients for five plant operating conditions and the number of either plant operating events or cycles for each of the design transients that will be used in the design and fatigue analyses of the ASME Code Class 1 piping systems.

The operating conditions are

- ASME Service Level A - normal conditions
- ASME Service Level B - upset conditions — incidents of moderate frequency
- ASME Service Level C - emergency conditions — infrequent incidents
- ASME Service Level D - faulted conditions — low-probability postulated events
- testing conditions

The number of cycles in the original list of transients in SSAR Table 3.9-1 appeared to be based on a 40-year life. In the DFSER, the staff noted that for a design life of 60 years, the number of cycles for each transient should be increased by a factor of 1.5. GE was requested to revise the SSAR to reflect this factor. This was DFSER Open Items 3.9.1-1 and 14.1.3.3.5.2-1. The resolution of this item is discussed in Section 3.9.1 of this report.

The number of events or cycles resulting from each of the listed design transients that are applicable to other ASME

Code Class piping systems is to be documented by the COL applicant in its design specification and/or stress report for each component. This was DFSER COL Action Item 14.1.3.3.5.2-1. SSAR Section 3.9.7.2 states that a COL applicant will identify ASME Class 2 or 3 piping systems under severe cyclic loads and perform analysis to ensure integrity for 60 years. This is acceptable.

3.12.5.3 Loadings and Load Combinations

The staff reviewed the methodology used for load combinations and the selected values of allowable stress limits. GE provided the design criteria for all ASME Code, Class 1, 2, and 3 piping and piping supports, using the load combinations and stress limits given in SSAR Section 3.9.3.1. GE stated that the method used in the combination of dynamic responses of piping loadings shall be in accordance with NUREG-0484, Revision 1.

From its review, the staff concludes that appropriate combinations of normal, operating transients, and accident loadings are specified to provide a conservative design envelope for the design of piping systems. The load combinations are consistent with the guidelines provided in SRP Section 3.9.3 and are acceptable.

3.12.5.4 Damping Values

RG 1.61, "Damping Values for Seismic Design of Nuclear Power Plants," Revision 0, contains recommended values of damping to be used in the seismic analysis of SSCs. In addition, RG 1.84, Revision 25, conditionally endorses ASME Code Case N-411-1. The damping values used by GE are the same as those specified in either RG 1.61 or those specified in ASME Code Case N-411-1 as permitted by RG 1.84, which is acceptable.

The SSAR uses the damping values specified in ASME Code Case N-411 with the independent support motion (ISM) method of response spectrum analysis. The staff's position on the application of N-411 damping values to the ISM method of analysis is that it is acceptable when the ISM method is used in accordance with the information and recommendations in Sections 2.3 and 2.4 of NUREG-1061, Volume 4. In the DFSER, the staff requested GE to confirm that N-411 damping in an ISM analysis will be applied in accordance with the staff's position. This was DFSER Confirmatory Item 14.1.3.3.5.4-1. SSAR Subsection 3.7.3.8.1.10, "Multiple-Supported Equipment and Components with Distinct Inputs," states that when the ISM response spectrum method of analysis is used, ASME Code Case N-411-1 damping will not be used. On the basis of this new commitment, the staff concludes that GE will only use the recommended damping values given in RG 1.61 in an ISM response spectrum analysis. This is acceptable. Therefore, DFSER Confirmatory Item 14.1.3.3.5.4-1 is resolved.

The staff's position on the use of N-411 damping values with ASME Code Case N-420 is that the two code cases may only be used in separate analyses as a further condition of RG 1.84, because the damping values established in Code Case N-411 might not be entirely appropriate for the damping characteristics of the linear energy absorbing supports. Therefore, the two Code Cases are not to be used in the same analysis. In the DFSER, GE was requested to confirm compliance with this staff position. This was DFSER Confirmatory Item 14.1.3.3.5.4-2. SSAR Subsection 3.7.3.8.1.7, "Damping Ratio," states that the ASME Code Case N-411-1 damping cannot be used for analyzing linear energy absorbing supports designed in accordance with ASME Code Case N-420. This is acceptable; therefore, DFSER Confirmatory Item 14.1.3.3.5.4-2 is resolved.

3.12.5.5 Combination of Modal Responses

For the response spectrum method of analysis, the modal responses are combined by the SRSS method. Closely spaced modes are combined using the criteria of RG 1.92, "Combining Modal Responses and Spatial Components in Seismic Response Analysis," Revision 1. The SSAR considers all modes with frequencies below 33 Hz in computing equipment and component response for seismic loadings, as stated in SSAR Sections 3.7.3.7.1 and 3.7.3.7.2. The staff concludes that this method is consistent with the applicable guidelines of SRP Section 3.9.2 and is acceptable.

3.12.5.6 High-Frequency Modes

For seismic analysis, consideration of high-frequency modes to preclude missing mass effects must be included. The staff's guidelines for this are provided in SRP Section 3.7.2, Appendix A.

For the analyses of vibratory loads (other than seismic) with significant high-frequency input (i.e., 33 to 100 Hz), the staff's positions are as follows:

(1) GE should address the methodology for the combination of high-frequency modal results. The high-frequency modes must be combined in accordance with the guidelines provided in RG 1.92. Use of other combination methods, such as the algebraic modal combination method for combining high-frequency modes, will require further justification and staff approval before use.

(2) GE should address non-linear analyses used to account for gaps between the pipe and its supports when subjected to vibratory loads with significant high-frequency. The description of and justification for such analyses must be submitted to the staff for review and approval before use.

In the DFSER, the staff requested that the SSAR be revised to reflect these two staff positions by stating that, if an alternative method is used, the details of its basis must be submitted to the staff for review and approval before use. This was DFSER Confirmatory Item 14.1.3.3.5.6-1. SSAR Subsection 3.7.3.7.3, "Methodologies Used to Account for High-Frequency Modes," provides a detailed procedure for calculating the responses associated with high-frequency modes above the cutoff frequency and combining them with the low-frequency modal responses. The staff reviewed this procedure and found that it follows the guidelines provided in SRP Section 3.7.2, Appendix A, and is acceptable.

GE also revised the SSAR to address the two issues related to analyses of vibratory loads with significant high-frequency input. SSAR Subsection 3.7.3.7, "Combination of Modal Responses," addresses the first issue. It clarifies the position that modal responses for modes below the cutoff frequency will be combined in accordance with methods that follow the guidelines of RG 1.92. The combination methods are applicable for seismic loads with 33 Hz cutoff frequency as well as for loads with higher frequency input, such as suppression pool dynamic loads that may have cutoff frequencies as high as 100 Hz. The responses associated with modes above the cutoff frequency are calculated and combined in accordance with the guidelines of SRP Section 3.7.2, Appendix A, as previously discussed. The staff reviewed this information and found that this revision adequately addressed the first issue. The second issue is addressed in the SSAR Section 3.7.3.3.4, "Analysis of Frame Type Supports," which states that nonlinear analysis methods to account for gaps between pipe and supports subjected to high frequency vibration loads, such as suppression pool loads will not be used. The staff reviewed this information and found it acceptable. The SSAR does not provide an analytical method to account for non-linear effects of excessively large gaps. Should such large gaps exist, it would mean a change to the commitments involving piping analysis methodology discussed in the SSAR, and it would result in an unreviewed safety question. Therefore, details of non-linear analysis should be submitted to the staff for approval prior to its implementation. On the basis of this evaluation, DFSER Confirmatory Item 14.1.3.3.5.6-1 is resolved.

3.12.5.7 Fatigue Evaluation for ASME Code Class 1 Piping

ASME Code, Section III requires that the cumulative damage from fatigue be evaluated for all ASME Code Class 1 piping. The cumulative fatigue usage factor should take into consideration all cyclic effects caused by the plant operating transients for a 60-year design life. However, recent test data indicates that the effects of the reactor environment could reduce the fatigue resistance of certain materials. A comparison of the test data with the Code requirements indicates that the margins in the ASME Code fatigue design curves might be less than originally intended. The DFSER reported that the staff was developing an interim position, which would be available at a later date, to account for the environmental effects in the fatigue design of the affected materials. At this time, the staff is assessing the potential generic implication of this issue on all operating plants. Depending on the severity of the issue, certain actions might be required to generically address this concern. This was inappropriately identified as Open Items 3.9.3.1-1 and 14.1.3.3.5.7-1 in the DFSER. DFSER Open Items 3.9.3.1-1 and 14.1.3.3.5.7-1 are subsumed by DFSER Open Item 14.1.3.3.5.7-2, which is discussed below.

Originally, for the ABWR, GE discussed with the staff its tentative procedure used for a foreign BWR plant design in progress. The information was provided to the staff during an audit held at the GE offices in San Jose, California, on March 23 through 26, 1992. The specified material for the ASME Code Class 1 piping in the ABWR is carbon steel. Using the GE position, additional fatigue evaluations would not be required when certain conditions are met, such as when the fluid temperature is below 245 °C (473 °F), the oxygen content is below 0.3 ppm, or the tensile stress hold time does not exceed 10 seconds. The exemption rules also extend to piping elbows and tees and valve bodies when these components are conservatively designed and analyzed, using the stress index method. Thus, only the circumferential girth butt welds in piping are considered to be critical by GE and are evaluated for environmental effects. The approach used by GE to account for the environmental effects on the girth butt welds is to modify the local peak stress through (1) the notch factor, (2) the mean stress factor, (3) the environmental correction factor, and (4) the butt-weld strength reduction factor. In the DFSER, the staff requested GE to include in its SSAR the proposed approach for accounting for the environmental effects in its fatigue analyses. This was DFSER Open Items 3.9.3.1-1 and 14.1.3.3.5.7-2.

SSAR Subsection 3.9.3.1.1.7, "Environmental Effects on Fatigue Evaluation of Carbon Steel Piping," 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. The SSAR describes the conditions for which these evaluations would be performed and the methodology for performing them. The staff found the approach consistent with the approach presented by GE at the March 1992 audit previously summarized. GE's procedure provided supplemental guidelines that enhance the design margin beyond the requirements of the ASME Code, Section III for fatigue evaluation. This is acceptable; and DFSER Open Items 3.9.3.1-1 and 14.1.3.3.5.7-2 are resolved.

3.12.5.8 Fatigue Evaluation of ASME Code Class 2 and 3 Piping

SSAR Section 3.9.3.1 states that the design life for the ABWR is 60 years. In response to an earlier staff request, SSAR Sections 3.9.3.1 and 3.9.7.2 stated that COL applicants will identify all ASME Code Class 2, 3, and QG D components that will be subjected to loadings that could result in thermal or dynamic fatigue and provide the analyses that are similar to those required by ASME Code, Section III, Subsection NB (ASME Class 1). This was DFSER COL Action Items 3.9.3.1-1 and 14.1.3.3.5.8-1. In the SSAR, GE modified these COL actions by further revising Sections 3.9.7.2 and 3.9.3.1 to state that COL applicants will identify ASME Code Class 2 or 3 or QG D components that are subjected to cyclic loadings, including operating vibration loads and thermal transient effects, of a magnitude and/or duration so severe that the 60-year design life cannot be assured by required Code calculations and, if similar designs have not already been evaluated, either provide an appropriate analysis to demonstrate the required design life or provide designs to mitigate the magnitude or duration of the cyclic loads. SSAR Section 3.9.3.1 provides criteria on the magnitude of severe thermal transients that should be evaluated for possible effect on plant life. These transients are temperature rate changes faster than 830 °C/hour (1494 °F/hour), when the total fluid temperature change is greater than 55 °C (100 °F). It also states specifically that COL applicants will perform ASME Code Class 1 fatigue analysis of the SRV discharge piping in the wetwell and the SRV quenchers. This is acceptable.

On the basis of current data, the staff believes that the margins built into the ASME fatigue design curves may not be sufficient to account for variations in the original fatigue test data because of various environmental effects. Therefore, consistent with the staff position discussed in Section 3.12.5.7 of this report, the staff's position for ASME Code Class 2 and 3 piping for which a fatigue analysis is performed is that the environmental effects should be considered in the fatigue analysis. This was DFSER Open Item 14.1.3.3.5.8-1. SSAR Section 3.9.3.1 states that environmental effects will be considered in the fatigue analyses of the SRV discharge piping and SRV quenchers in accordance with the requirements for ASME Code, Section III, Class 1, carbon steel piping discussed in Section 3.9.3.1.1.7 of this report. This is acceptable, and DFSER Open Item 14.1.3.3.5.8-1 is resolved.

3.12.5.9 Thermal Oscillations in Piping Connected to the Reactor Coolant System

In accordance with NRC Bulletin 88-08, the staff requests that licensees and applicants review systems connected to the RCS to determine whether any sections of this piping that cannot be isolated can be subjected to temperature oscillations that could be induced by leaking valves.

In the design of the ABWR emergency core cooling systems, both the residual heat removal (RHR) system and high-pressure core flooder (HPCF) have piping that will be directly connected to the reactor pressure vessel (RPV). In the unisolable sections of RHR piping, leaking toward the RPV cannot occur because the pressure will always be higher on the reactor side during normal plant operation when the upstream pumps are not operating. In the HPCF system design, the only unisolable piping connected to the RPV will be the section of pipe between the reactor nozzle and the upstream isolation check valve. Cold water in this system will be upstream of the injection valve (gate valve) that is outside the primary containment. The region upstream of the injection valve will operate at a pressure lower than reactor pressure except when the HPCF safety function is required. Therefore, cold water will not flow to the unisolable pipe section and stratification will not be a problem in the HPCF system.

In the DFSER, the staff concluded that GE had adequately addressed the potential problems described in Bulletin 88-08. Subsequently, however, the staff noted that GE did not address the potential problem described in Supplement 3 to Bulletin 88-08, although Supplements 1 and 2 were addressed adequately. It involved the development of potential cyclic stratified flow and associated thermal striping that may occur because of possible leakage past the valve disk and out the valve stem packing gland. This flow stratification and striping may occur when the pressure on the upstream side of the valve is less than the RPV system pressure during normal operation. Supplement 3 reported an incident in which this phenomenon induced thermal fatigue, which led to a through-wall pipe crack in a foreign reactor. The staff reviewed the SSAR and requested that GE provide additional information to address this concern. GE then identified the ABWR piping sections that could be susceptible to unacceptable thermal stresses, owing to this phenomenon. They include sections of the RHR system and the HPCF system. For the affected piping sections, GE will require that either (1) the gate valve in each of the unisolable piping sections be located at a distance equal or greater than 25 pipe diameters from the RPV nozzle or (2) stress analysis be performed to show that stresses and fatigue from potential stratification and thermal striping are acceptable per the ASME Code. The SSAR incorporated these new requirements by including an additional note on the P&ID's of the RHR system (SSAR Figure 5.4-10) and the HPCF system (SSAR Figure 6.3-7). This is acceptable; therefore, the staff concludes that GE has adequately addressed the potential problem described in Supplement 3 to Bulletin 88-08.

3.12.5.10 Thermal Stratification

Thermal stratification is a phenomenon that can occur in long runs of horizontal piping when two streams of fluid at different temperatures flow in separate layers without appreciable mixing. Under these stratified flow conditions, the top of the pipe may be at a much higher temperature than the bottom. This thermal gradient produces pipe deflections, support loads, pipe-bending stresses, and local stresses that may not have been accounted for in the original piping design. The effects of thermal stratification have been observed in both BWR and PWR feedwater piping as discussed in NRC Information Notice (IN) 84-87 and NRC IN 91-38.

During an audit conducted at GE's offices in San Jose, California, on March 23 through 26, 1992, the staff asked GE to explain and demonstrate how the thermal stratification phenomenon was considered in the ABWR piping design. In response, GE stated that thermal stratification will be considered as a normal design load in the ASME Code Class 1 stress and fatigue evaluation of the feedwater piping. The ABWR sample problem criteria document states that the feedwater line will be analyzed for two thermal stratification load cases: (1) thermal stratification in the piping at the RPV nozzle and (2) thermal stratification in the feedwater header piping. The loads will be included in the piping fatigue analysis and in the evaluations of the head fitting and RPV nozzles. The temperature differences and locations for the stratification loads were defined in the feedwater piping pressure/temperature cycle diagrams. This was DFSER COL Action Item 14.1.3.3.5.10-1. SSAR Section 3.9.3.1 states that the COL applicant shall design for thermal stratification in accordance with the requirements of that section. This is acceptable.

The staff reviewed and discussed the thermal stratification analysis methodology with the cognizant GE engineers. The staff found the analysis method acceptable with the exception of an apparent discrepancy in load application. GE defined the stratified temperature profile in the pipe cross section as a constant hot temperature in the top half and cold temperature in the bottom half, with a step change in temperature at the centerline. However, in the pipe stress analysis, a linear top-to-bottom temperature profile was applied. The linear temperature profile provides lower bending moments and stresses than the step change profile. GE was asked to justify (1) the adequacy of the piping analysis load input and (2) the omission of the high-cycle fatigue effects resulting from thermal striping and why it should not be considered in the analysis. The staff also asked GE to provide additional justification for their methodology, including test information to support their thermal stratification load definition. This was DFSER Open Items 3.9.3.1-2 and 14.1.3.3.5.10-1.

SSAR Section 3.9.3.1 states that thermal stratification is one of the specific operating conditions that is included in the loads and load combinations contained in the piping design specifications and design reports. Stratification can occur in the feedwater piping during plant startup and during hot standby conditions following scram. If evidence of stratification is detected in any other piping system during design or startup, it will be evaluated to determine whether it is significant in terms of stress and deflection. As a general guideline, if temperature differences between the top and bottom of the pipe are less than 27 °C (48.6 °F), it is assumed insignificant and not included in the design specification and design reports. The staff reviewed this information and found it acceptable.

In addition, SSAR Section 3.9.2.1.3, "Thermal Stratification in Feedwater Piping," describes a special test that will be performed as part of the startup program to monitor the conditions and effects of thermal stratification in sections of the feedwater piping where stratification is anticipated. The test program will measure temperatures around the pipe circumference, strains at points of high stress, and pipe displacements, which are due to bowing caused by stratification. GE committed to perform this test to address the staff concern over the adequacy of the linear temperature profile assumed in the thermal stratification analysis. From its review, the staff concludes that the test should provide appropriate data to evaluate the adequacy of the assumed temperature profile.

In a letter dated April 19, 1993, GE presented thermal striping evaluation for the feedwater header pipe. Calculations summarized in the letter report show that stresses at the pipe wall caused by striping are well below the metal endurance limit and are, therefore, negligible. The staff concluded that GE had provided a technical approach to adequately address the issue of thermal stratification in the ABWR feedwater piping systems. GE has also included this information in the SSAR. On the basis of this evaluation, DFSER Open Items 3.9.3.1-2 and 14.1.3.3.5.10-1 are resolved.

3.12.5.11 Safety-Relief Valve Design, Installation, and Testing

SSAR Section 3.9.3.3 contains the design, installation, and testing criteria applicable to the mounting of pressure relief devices used for the overpressure protection of ASME Code Class 1, 2, and 3 components. The staff reviewed this information in accordance with SRP Section 3.9.3, including an evaluation of the applicable loading combinations and stress criteria. The review extended to consideration of the means to accommodate the rapidly applied reaction force when a safety valve or relief valve opens and the transient fluid-induced loads are applied to the piping downstream of a safety valve or relief valve in a closed discharge piping system. The information in Section 3.9.3.3 of the SSAR meets the applicable guidelines of SRP Section 3.9.3 and is, therefore, acceptable.

In accordance with TMI Action Item II.D.1 of NUREG-0737, both PWR and BWR licensees and applicants are required to conduct testing to qualify the RCS relief and safety valves and associated piping and supports under expected operating conditions for design-basis transients and accidents. GE's response to Item II.D.1 is discussed in SSAR Section 1A.2.9. This section states that the safety-relief valve models that will be used for ABWR plants have been tested under ABWR steam discharge conditions. It further states that if the ABWR design should contain any safety-relief valves or discharge piping that is not similar to those that have been tested, the COL applicant will test the valves in accordance with TMI Item II.D.1. This is acceptable.

In performing the hydraulic transient piping analyses associated with the SRV discharge, the SSAR assumes a minimum rise time of 20 msec. Rise times faster than this value could result in higher loads than analytically predicted. The assumed rise time is based on past SRV designs and existing test data. The COL applicant must retest the SRVs if it should purchase any SRV or install its SRV piping in a configuration that is not similar to those that have been tested. This was DFSER COL Action Items 3.9.3.2-1 and 14.1.3.3.5.11-1. In Amendment 33 of the SSAR, GE added Section 1A.3.7, "Testing of SRV and Discharge Piping," to state that the COL applicant will confirm that any SRVs or discharge piping installed that is not similar to those that have been tested will be tested in accordance with SSAR Section 1A.2.9. The staff considers this acceptable. Therefore, COL Action Items 3.9.3.2-1 and 14.3.3.5.11-1 are resolved.

3.12.5.12 Functional Capability

Note 9 to the SSAR, Table 3.9-2, states that all ASME Code Class 1, 2, and 3 piping systems that are essential for safe shutdown under the postulated events listed in the table are designed to meet the recommendations in NUREG-1367, "Functional Capability of Piping Systems," dated November 1992. In no case shall the piping stress exceed the limits designated for Service Level D in the ASME Code, Section III. The Service Level D limits are 3.0 S_m (not to exceed 2.0 S_y) for ASME Code Class 1 piping and 3.0 S_h (not to exceed 2.0 S_y) for Class 2 and 3 piping. Dynamic testing conducted by EPRI, GE, and the NRC has established that these stress levels do not result in a loss of piping functional capability. Thus, the staff concludes the methodology and stress levels for ensuring the functional capability of piping systems acceptable.

3.12.5.13 Combination of Inertial and Seismic Motion Effects

Piping analyses must include the effects caused by the relative building movements at supports and anchors (seismic anchor motion) as well as the seismic inertial loads. This is necessary when piping is supported at multiple locations within a single structure or is attached to two separate structures.

The effects of relative displacements at support points must be evaluated by imposing the maximum support displacements in the most unfavorable combination. This can be performed, using a static analysis procedure. Relative displacements of equipment supports (e.g., pumps or tanks) must be included in the analysis along with the building support movements.

When required for certain evaluations, such as support design, the responses that are due to the inertia effect and relative displacement effect should be combined by the absolute sum method.

In lieu of this method, time histories of support excitations may be used, in which case both inertial and relative displacement effects are already included. Consideration of these effects and analyses was DFSER Open Item 14.1.3.3.5.13-1.

Section 3.7.3.8.1.8 of the SSAR describes the methodology for considering the effects caused by relative building movements. The displacements that are obtained from the dynamic building analysis are applied to the piping anchors and restraints corresponding to the maximum differential displacements that could occur. Three analyses are performed: one for each of the two horizontal differential displacements and one for the vertical. The resulting stresses in the piping are treated as secondary stresses because they are self-limiting. The primary loads owing to inertia are combined with the secondary loads owing to the relative displacements by the SRSS method.

The SRSS combination of these responses deviates from the acceptance criteria of SRP Section 3.9.2, which specifies the more conservative absolute sum method. To justify this deviation, SSAR Section 3.7.3.8.1.8 states that the inertia and relative support displacement responses are dynamic in nature and their peak values are not expected to occur at the same time. Anchor movement effects are computed from static analyses in which the support movement effects are applied to produce the most conservative loads on the piping. In view of this, GE believes that an SRSS combination is appropriate. GE provided a description of the theoretical basis for this position based on an independent support-motion time-history analysis. The staff found the approach technically justifiable and consistent with the recommendations of NUREG-1061, Volume 4, for independent support motion analysis. Therefore, the staff concludes that the SRSS combination of responses owing to inertia and relative displacements is acceptable for the ABWR. On the basis of the above, DFSER Open Item 14.1.3.3.5.13-1 is resolved.

3.12.5.14 Cutoff Frequency for Hydrodynamic Loadings

SSAR Section 3.7 states that the cutoff frequency for dynamic analysis is 60 Hz for suppression pool hydrodynamic loads. For piping systems with a fundamental frequency greater than 20 Hz, the cutoff frequency is 100 Hz for suppression pool hydrodynamic loads. These cutoff frequencies were previously used in the hydrodynamic analyses for currently operating BWR plants. Because the hydrodynamic load methodology used for the ABWR is the same as that used for the operating BWR plants, the cutoff frequency is acceptable for the ABWR.

3.12.5.15 OBE as a Design Load

In SECY-93-087, the staff recommended eliminating the OBE from the design for advanced light-water reactors. The Commission approved the staff recommendations in its SRM dated July 21, 1993. For the ABWR, GE originally proposed that the OBE be equal to one-third of the SSE. The DFSER noted that the staff was discussing details with GE for the necessary actions that will be required for eliminating the OBE from the design of SSCs in the ABWR and that the staff's evaluation of using a single-earthquake design based on only the SSE will be addressed in the FSER. This was DFSER Open Items 3.1-1 and 14.1.3.3.5.15-1.

In a letter to GE dated September 11, 1992, the staff transmitted a guidance document that identified the necessary changes to existing seismic design criteria that are acceptable for implementing the proposed rule change as it pertains to the design of safety-related SSCs in the GE ABWR. This document included specific supplemental criteria for fatigue, seismic anchor motion, and piping stress limits that should be applied when the OBE is eliminated. A detailed discussion of this document is in Section 3.1.1 of this report. For fatigue evaluation, two SSE events with 10 maximum stress cycles per event (or an equivalent number of fractional cycles) should be considered. The effects of SAM owing to the SSE should be considered in combination with the effects of other normal operational loadings that might occur concurrently. For Class 1 primary stress evaluation, seismic loads need not be evaluated for consideration of Level B Service Limits for Eq. (9). However, for satisfaction of primary plus secondary stress range limits in Eq. (10), the full SSE stress range or a reduced range corresponding to an equivalent number of fractional cycles must be included for Level B Service limits. These load sets should also be used for evaluating fatigue effects. In addition, the stress that is due to the larger of the full range of SSE anchor motion or the resultant range of thermal expansion plus half the SSE anchor motion range, must not exceed 6.0 S_m. For Class 2 and 3 piping, seismic loads are not required for consideration of occasional loads in satisfying the Level B Service Limits for Eq. (9). Seismic anchor motion stresses are not required for consideration of secondary stresses in Eq. (10). However, stresses that are due to the combination of range of moments caused by thermal expansion and SSE anchor motions must not exceed 3.0 S_h.

SSAR Section 3.9, defines a number of revised requirements associated with the elimination of the OBE. SSAR Section 3.7.3.2, "Determination of Number of Earthquake Cycles," states that the SSE is the only design earthquake considered for the ABWR. The fatigue evaluation of ASME components would take into consideration two SSE events with 10 peak stress cycles per event. Alternately, an equivalent number of fractional vibratory cycles may be used (but with an amplitude not less than one-third of the maximum SSE amplitude) when derived in accordance with Appendix D of IEEE Standard

344-1987. The staff found this commitment consistent with the NRC guidance document previously discussed above and the Commission-approved staff recommendations on the issue of OBE elimination as discussed in Section 3.1.1 of this report. This is acceptable.

SSAR Table 3.9-1, "Plant Events," was revised to replace the OBE cycles with 20 peak SSE cycles for evaluation of Service Level B limits. A footnote was added to require the effects of SAMs owing to SSE to be evaluated to ensure functionality during and following an SSE. The staff finds this commitment consistent with the NRC guidance document and the Commission-approved staff recommendations on the issue of OBE elimination as discussed in Section 3.1.1 of this report. This is acceptable.

SSAR Table 3.9-2, "Load Combinations and Acceptance Criteria," was revised to replace the normal operation and OBE load combination with a normal and SSE combination for fatigue evaluation of Class 1 components under Service Level B. Footnote 7 of this table defines the changes and additions to the stress limits of ASME Code, Section III, Subsections NB-3600 and NC/ND-3600. For Class 1 piping, the stress owing to the larger of the full range of SSE SAM moment or the resultant range of thermal plus one-half the SAM moment must not exceed 6.0 S_m. SSE inertia and SAM loads must be included in NB-3600 Eqs. (10) and (11). For Class 2 and 3 piping, the stress that is due to the larger of the full range of SSE SAM moment or the resultant range of thermal plus one-half the SAM moment must not exceed 3.0 S_h. SSE inertia and SAM loads must not be included in NC/ND-3600 Eqs. (9), (10), and (11). The staff finds these revised stress limits consistent with the requirements of the NRC guidance document. On the basis of this information and the more detailed discussion of this issue in Section 3.1.1 of this report, DFSER Open Items 3.1-1 and 14.1.3.3.5.15-1 are resolved.

3.12.5.16 Welded Attachments

For the analysis of local stresses at welded attachments to piping (e.g., lugs, trunnions, or stanchions), the SSAR presents several ASME Code Cases. Code Case N-318-3 is acceptable to the staff and is endorsed in RG 1.84. The staff noted that this Code Case is conditionally approved in RG 1.84 on the basis that the applicant specifies (1) the method of lug attachment, (2) the piping system involved, and (3) the location in the system where the case is to be applied. The staff concludes, however, that for the ABWR design certification, these conditions in RG 1.84 are not needed to reach a safety conclusion and, therefore, are not required.

Code Cases N-391 and N-392 are endorsed by the staff in RG 1.84 and are acceptable.

3.12.5.17 Modal Damping for Composite Structures

The staff reviewed the issue of modal damping for composite structures during its audit on March 23 through 26, 1992, at GE's offices in San Jose, California. At that time, the SSAR did not describe the application of modal damping for composite structures in the analysis of piping systems. However, a review of a GE internal document entitled, "Piping Systems Design Criteria and Analysis Methods," contained a table of damping values for various types of piping supports. The damping values for the piping supports (e.g., snubbers and struts) were higher than the damping values tabulated for the piping.

GE indicated that these values were presented because modal damping for composite structures could be used in a response spectrum analysis as an option. In the DFSER, the staff reported that if GE plans to use the modal damping for composite structures as an option for piping analysis, then a description and justification of the approach must be provided in the SSAR. This was DFSER Open Items 3.9.2.2-6 and 14.1.3.3.5.17-1.

SSAR Section 3.7.3.8.1.7, "Damping Ratio," states that strain-energy-weighted modal damping can also be used in the dynamic analysis of piping systems. Strain-energy weighing is used to obtain the modal damping coefficient owing to the contributions of damping from different elements of the piping system. The element damping values are specified in SSAR Table 3.7-1. The procedure for calculating strain-energy-weighted modal damping is given in SSAR Section 3.7.2.15. The staff reviewed the information in the SSAR and found it acceptable; therefore, DFSER Open Items 3.9.2.2-6 and 14.1.3.3.5.17-1 are resolved.

3.12.5.18 Minimum Temperature for Thermal Analyses

In the DFSER, the staff noted that GE had not provided any information that would establish a minimum temperature at which an explicit piping thermal expansion analysis would be required. Unless GE had provided this information in the SSAR, the staff would have required that thermal analyses be performed for all temperature conditions above ambient. These were DFSER Open Items 3.9.3.1-3 and 14.1.3.3.5.18-1.

SSAR Section 3.9.3.1, "Loading Combinations, Design Transients and Stress Limits," states that piping loads due to the thermal expansion of the piping and thermal anchor movements at supports are included in the piping load combinations. All operating modes are evaluated, and the maximum moment ranges are included in the fatigue evaluation. Piping systems with maximum operating temperatures of less than or equal to 65.6 °C (150 °F) are not required to be analyzed for thermal expansion loading because, when below this temperature, thermal-induced stresses will be low and inconsequential to piping designs. The staff reviewed this information and concluded that GE had defined a reasonable and acceptable minimum temperature at which an explicit thermal analysis would be performed. On the basis of this evaluation, DFSER Open Items 3.9.3.1-3 and 14.1.3.3.5.18-1 are resolved.

3.12.5.19 Intersystem LOCA

In SECY-90-016, dated January 12, 1990, the NRC staff discussed the resolution of the intersystem LOCA issue for advanced light-water reactor plants by requiring that low-pressure piping systems that interface with the RCPB be designed to withstand full RCS pressure to the extent practicable. In its June 26, 1990, SRM, the Commission approved these staff recommendations provided that all elements of the low-pressure systems are considered. In the DFSER, the staff noted that GE had not yet submitted the details of the piping design for the full RCS pressure. This was DFSER Open Item 14.1.3.3.5.19-1.

SSAR Subsection 3.9.3.1, "Loading Combinations, Design Transients, and Stress Limits," requires that low-pressure piping systems that interface with the RC boundary be designed with a pipe wall thickness calculated for a pressure equal to 0.4 times the RCS pressure but not less than that of a schedule 40 pipe. On the basis of a staff review and discussions with GE and the more detailed evaluation in Section 3.9.3.1.1 of this report, the staff finds this requirement acceptable. Therefore, DFSER Open Item 14.1.3.3.5.19-1 is resolved.

3.12.5.20 Conclusions

One the basis of its review, the staff concludes that--

(1) GE meets GDC 1 and 10 CFR Part 50, Appendix B with regard to piping systems being designed, fabricated, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed, and with appropriate quality control.

(2) GE meets GDC 2 and 10 CFR Part 100, Appendix A with regard to design transients and resulting load combinations for piping and pipe supports to withstand the effects of earthquakes combined with the effects of normal or accident conditions.

(3) GE meets GDC 4 with regard to piping systems important to safety being designed to accommodate the effects of and to be compatible with the environmental conditions of normal and accident conditions.

(4) GE meets GDC 14 with regard to the reactor coolant pressure boundary of the primary piping systems being designed, fabricated, constructed, and tested to have an extremely low probability of abnormal leakage, of rapid propagating failure, and of gross rupture.

(5) GE meets GDC 15 with regard to the reactor coolant piping systems being designed with specific design and service limits to assure sufficient margin that the design conditions are not exceeded.

3.12.6 Pipe Support Criteria

3.12.6.1 Applicable Codes

The staff reviewed the methodology used in the design of ASME Code Class 1, 2, and 3 component supports as described in SSAR Sections 3.9.3.4 and 3.9.3.5. The staff also assessed the design and structural integrity of three types of supports: plate and shell, linear, and component standard types. All ASME Code Class 1, 2, and 3 component supports for the ABWR standard plant will be constructed in accordance with ASME Code, Section III, Subsection NF. In addition, the SSAR states that the design is augmented by the application of Code Case N-476, Supplement 89.1, which governs the design of single-angle members. It further states that if eccentric loads or other torsional loads are not accommodated by designing the load to act through the shear center, analyses will be performed in accordance with such torsional analysis methods as "Torsional Analysis of Steel Members," AISC Publication T1142/83.

Although Code Case N-476 has not been endorsed by the staff in RG 1.84, the staff finds that it provides adequate design rules for the single-angle members. The staff finds that GE's proposed documents provide sufficient technical guidelines to perform a torsional analysis of steel members and are acceptable.

The staff finds Subsection NF acceptable for the design of piping supports. The staff has not endorsed the use of ANSI/AISC N-690, "Specification for the Design, Fabrication and Erection of Steel Safety-Related Structures for Nuclear Facilities," in lieu of Subsection NF (see Section 3.12.6.12 of this report for more information).

3.12.6.2 Jurisdictional Boundaries

SSAR Section 3.9.3.4, defines the jurisdictional boundaries between pipe supports and interface attachment points, such as structural steel, in accordance with the ASME Code, Section III, Subsection NF (1989 edition). The staff's review of the jurisdictional boundaries described in the 1989 edition finds that they are sufficiently defined to ensure a clear division between the pipe support and the structural steel and are acceptable.

3.12.6.3 Loads and Load Combinations

SSAR Section 3.9.3.4.1 states that the loading combinations for the design of piping supports correspond to those used for the design of the supported pipe. The staff's evaluation of the load combinations for the supported pipe is contained in Section 3.12.5.3 of this report. The stress limits for pipe supports are in accordance with the ASME Code, Section III, Subsection NF and Appendix F (1989 Edition). The supports are generally designed or qualified by the load rating method as described in NF-3260 or by the stress limits specified in NF-3231. This is acceptable.

3.12.6.4 Pipe Support Baseplate and Anchor Bolt Design

SSAR Section 3.9.3.4 of earlier amendments stated that the concrete anchor bolts, which would be used for pipe support base plates, will be designed to the applicable factors of safety defined in Office of Inspection and Enforcement (IE) Bulletin 79-02, Revision 1. Loading combinations for component supports are discussed in the previous section. In general, the factors of safety for anchor bolts are acceptable. However, in the DFSER, the staff noted that in SSAR Section 3.9.3.4, "Component Supports," GE had not discussed the use of specific types of anchor bolts to be used for pipe support base plates in the ABWR standard plant. For example, under-cut-type anchor bolts behave in a ductile manner, but the staff's position is that the safety factors in IE Bulletin 79-02, Revision 2, are still applicable unless justification for alternative safety factors is provided. Therefore, the COL applicant should justify the use of safety factors for anchor bolts other than those provided in IE Bulletin 79-02, Revision 2, and submit the justification to the staff for review and approval before their use. These were DFSER COL Action Items 3.9.3.3-1 and 14.1.3.3.6.4-1. In addition, the staff noted that irrespective of the type of concrete anchor bolt used for piping supports, the COL applicant should implement the action item in IE Bulletin 79-02 related to pipe support baseplate flexibility. This was DFSER COL Action Item 14.1.3.3.6.4-2.

SSAR Section 3.9.3.4 states that concrete anchor bolts (including under-cut-type anchor bolts) that are used for pipe support base plates will be designed to the applicable factors of safety that are defined in IE Bulletin 79-02, Revision 2. SSAR Section 3.9.3.4 further states that pipe support base plate flexibility will be accounted for in the calculation of concrete anchor bolt loads, in accordance with IE Bulletin 79-02, Revision 2. On the basis of these statements, the staff concludes that the ABWR pipe support baseplate and anchor bolt designs will be in accordance with applicable portions of IE Bulletin 79-02, Revision 2, and are acceptable. Loading combinations for component supports are discussed in the section above.

In addition, SSAR Section 3.9.3.4 states that the COL applicants shall provide justification for the use of safety factors for concrete anchor bolts other than those specified in IE Bulletin 79-02. This justification must be submitted to the NRC for review and approval prior to installing of the bolts. SSAR Section 3.9.3.4 also states that COL applicants shall account for pipe support base plate flexibility in accordance with IE Bulletin 79-02. This is acceptable.

3.12.6.5 Use of Energy Absorbers and Limit Stops

The DFSER noted that GE had not provided the specific analysis methods or procedures to be used for the ABWR pipe support design. The staff requested that GE address in the SSAR the use of seismic restraints other than snubbers and their modeling assumptions. This was part of DFSER Open Items 3.9.3.3-1 and 14.1.3.3.6-1.

SSAR Section 3.7.3.3.1.7, "Modeling of Special Engineered Pipe Supports," states that modifications to the normal linear-elastic piping analysis methodology used with conventional pipe supports are required to calculate the loads acting on the supports and on the piping components when special engineered supports are used. These supports include energy absorbers and limit stops, which are described in SSAR Section 3.9.3.4.1(G). The modifications are needed to account for greater damping of the energy absorbers and the non-linear behavior of the limit stops. If these special devices are used, the modeling and analytical methodology will be in accordance with methodology accepted by the regulatory agency at the time of design certification or at the time of COL application, per the discretion of the COL applicant. In addition, the information required by RG 1.84 related to the use of energy absorbers (ASME Code Case N-420) will be provided to the regulatory agency. This is acceptable. Although the staff has not yet endorsed the use of limit stops, the commitment in SSAR Section 3.7.3.3.1.7 to use design criteria in accordance with that accepted by the regulatory agency at the time of either design certification or COL application, is also acceptable to the staff. On the basis of this evaluation, the applicable parts of DFSER Open Items 3.9.3.3-1 and 14.1.3.3.6-1 are resolved.

3.12.6.6 Use of Snubbers

The DFSER noted that GE had not provided the specific analysis methods or procedures to be used for the ABWR pipe support design. The staff requested that GE address in the SSAR the types of snubbers to be used in the ABWR standard plant and their characteristics. This was part of DFSER Open Items 3.9.3.3-1 and 14.1.3.3.6-1.

SSAR Section 3.9.3.4.1, "Piping," provides a description of the snubbers, design criteria and testing, installation, and examination requirements. It states that both mechanical and hydraulic type snubbers will be used. It describes how their operational and performance characteristics will be verified by test. From its review of the description, the staff concluded that the information provided is consistent with applicable portions of SRP Section 3.9.3 and is acceptable; therefore, the applicable parts of DFSER Open Items 3.9.3.3-1 and 14.1.3.3.6-1 are resolved.

3.12.6.7 Pipe Support Stiffnesses

The DFSER noted that GE had not provided the specific analysis methods or procedures to be used for the ABWR pipe support design. The staff requested that GE address in the SSAR the pipe support stiffness values and support deflection limits used in the piping analyses. This was part of DFSER Open Items 3.9.3.3-1 and 14.1.3.3.6-1.

SSAR Sections 3.7.3.3.1.6, "Modeling of Piping Supports," and 3.7.3.3.4, "Analysis of Frame Type Supports," describe the criteria for using equivalent stiffnesses for snubbers and struts, including their supporting structures, and for frame type supports. Methods for determining the stiffnesses of each support type were provided. Stiffnesses for frame type supports and for supporting structures must be included unless they can be shown to be rigid. GE provided a deflection limit criteria that must be met for supports that are modeled as rigid. The staff reviewed this information and found it acceptable. Therefore, DFSER Open Items 3.9.3.3-1 and 14.1.3.3.6-1 are resolved with regard to the issue of pipe support stiffness.

3.12.6.8 Seismic Self-Weight Excitation

The DFSER reported that GE had not provided the specific analysis methods or procedures to be used for the ABWR pipe support design. The staff requested that GE address in the SSAR the seismic excitation of the pipe supports (especially large frame-type structures) in the design of the pipe support anchorage. This was part of DFSER Open Items 3.9.3.3-1 and 14.1.3.3.6-1.

SSAR Section 3.7.3.3.4 includes a description of the design loads to be considered in the analysis of frame-type supports. It explains that in addition to the loads transmitted from the piping to the support, the support internal loads caused by the weight, thermal, and inertia effects, which are due to the support structure itself, must be included in the support analysis. Therefore, the seismic self-weight excitation will be included, which is acceptable; therefore, DFSER Items 3.9.3.3-1 and 14.1.3.3.6-1 are resolved with regard to this issue.

3.12.6.9 Design of Supplementary Steel

SSAR Section 3.9.3.4 provides design criteria for the design of pipe supports, using supplementary steel. Supplementary steel for pipe supports are designed in accordance with ASME Code, Section III, Subsection NF. The use of Subsection NF is standard industry practice and has been proven to provide adequate design guidelines for the design of structural steel for use as pipe supports. This is acceptable.

3.12.6.10 Consideration of Friction Forces

The DFSER reported that GE had not provided the specific analysis methods or procedures to be used for the ABWR pipe support design. The staff requested that GE address in the SSAR the coefficient of friction to be used for considering friction forces between the pipe and the steel frames. This was part of DFSER Open Items 3.9.3.3-1 and 14.1.3.3.6-1.

SSAR Section 3.7.3.3.4, "Analysis of Frame Type Supports," includes a description of the design loads that must be considered in the analysis of frame-type supports. One of the design loads includes friction loads caused by a pipe sliding on the support. In calculating these loads, GE will use static coefficients of friction of 0.80 for steel on steel and 0.15 for lubricated plates. The staff finds this acceptable because the magnitude of these specified static coefficients is sufficiently conservative to define the friction force developed at different kinds of contact surfaces of a sliding pipe on its support; therefore, the applicable parts of DFSER Open Items 3.9.3.3-1 and 14.1.3.3.6-1 are resolved.

3.12.6.11 Pipe Support Gaps and Clearances

The DFSER noted that GE had not provided the specific analysis methods or procedures to be used for the ABWR pipe support design. The staff requested that GE address the hot and cold gaps to be used between the pipe and the box-frame-type of support. This was part of DFSER Open Items 3.9.3.3-1 and 14.1.3.3.6-1.

SSAR Section 3.7.3.3.4 includes information on allowable gaps. It states that the total gap or diametral clearance between the pipe and frame support shall be between 1.6 mm (1/16 in.) and 4.76 mm (3/16 in.) when the pipe is in either the hot or cold condition. The staff finds this acceptable because a small gap will ensure validity of using the linear analysis methodologies presented in the SSAR for piping design; therefore, the applicable parts of DFSER Open Items 3.9.3.3-1 and 14.1.3.3.6-1 are resolved.

3.12.6.12 Instrumentation Line Support Criteria

In the DFSER the staff noted that GE had not provided any information on the design criteria for the structural design of instrumentation line supports. The staff requested that this information be included in the SSAR. This was DFSER Open Item 3.9.3.3-2 and part of DFSER Open Item 14.1.3.3.6-1.

SSAR Sections 3.7.3.8.1.9 and 3.9.3.4.1 state that supports for ASME Code, Section III instrumentation lines are analyzed in accordance with SSAR Subsection 3.7.3, "Seismic Subsystem Analysis," and designed in accordance with SSAR Section 3.9.3.4, "Component Supports." Thus, these instrumentation lines will be analyzed, using the same methodology that will be applied to small-bore piping, and will be designed in accordance with ASME Code, Section III, Subsection NF. The staff finds this acceptable because these criteria and this guidance are sufficient to ensure code compliance and seismic design adequacy of the instrumentation line supports; therefore, DFSER Open Item 3.9.3.3-2 and the applicable part of DFSER Open Item 14.1.3.3.6-1 are resolved.

The industry has taken the position that ANS/AISC N-690 is useful in the design of instrumentation sensing line supports and has recommended that the industry be allowed to use it. Its use would have the effect of reducing the QA recordkeeping requirements and Code stamping required by Section NF of ASME Code, Section III. The staff believes that ANS/AISC N-690 alone is not an acceptable standard for construction of ASME component supports. ASME Code, Section III, Subsection NF, should be used. However, the staff is currently participating in the ASME effort to incorporate N-690 into Subsection NF. Subsequent to a staff-endorsed version of NF incorporating N-690, Subsection NF will specify the rules acceptable to the staff for construction of ASME Code Class supports. In the DFSER, the staff stated that when this staff-approved version is available, the COL applicant seeking its use may submit a request to the staff for approval on a plant-specific basis. This was DFSER COL Action Item 14.1.3.3.6.12-1. Upon further consideration, the staff determined that a COL action item would not be appropriate because, as discussed in Section 3.12.2.1 of this report, DFSER COL Action Item 14.1.3.3.2.1-1 provides a commitment for the COL applicant to ensure that the design will be consistent with ASME Code and addenda as endorsed in 10 CFR 50.55a in effect at the time of application. Therefore, DFSER COL Action Item 14.1.3.3.6.12-1 is deleted.

3.12.6.13 Pipe Deflection Limits

In the DFSER the staff noted that GE had not provided the specific analysis methods or procedures to be used for the ABWR pipe support design. The staff requested that GE include this information in the SSAR design criteria that will ensure that the maximum deflections of the piping at support locations for static and dynamic loadings are within an allowable limit to preclude failure of the pipe supports and hangers. This was part of DFSER Open Items 3.9.3.3-1 and 14.1.3.3.6-1.

SSAR Section 3.9.3.4.1, "Piping," includes design criteria to ensure that maximum deflections at pipe supports will be within allowable limits. The staff reviewed the deflection criteria and found them acceptable because they are compatible with design assumptions and standard engineering practice; therefore, the applicable portions of DFSER Open Items 3.9.3.3-1 and 14.1.3.3.6-1 are resolved.

3.12.6.14 Conclusions

On the basis of these discussions and the evaluation of SSAR Sections 3.9.3.3, 3.9.3.4, 3.9.3.5, 3.7.3.3, the staff concludes that supports of piping systems important to safety are designed to quality standards commensurate with their importance to safety. The staff's conclusion is based on the following:

(1) GE satisfies the requirements of GDC 1 and 10 CFR 50.55a by specifying methods and procedures for the design and construction of safety-related pipe supports in conformance with general engineering practice, and

(2) GE satisfies the requirements of GDC 2 and GDC 4 by designing and constructing safety-related pipe supports to withstand the effects of normal operation as well as postulated events such as LOCAs and dynamic effects resulting from the SSE.

3.12.7 High-Energy Line-Break Criteria

GDC 4 requires that SSCs important to safety be designed to be compatible with and to accommodate the effects of the environmental conditions resulting from normal operations, maintenance, testing, and postulated accidents, including LOCAs. It also requires that they be adequately protected against 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 plant.

In accordance with SRP Section 3.6.2, Revision 2, the staff reviewed GE's proposed criteria and methodology to postulate pipe breaks and leakage cracks and to analyze the effects of breaks in high-energy fluid systems on adjacent safety-related SSCs with regard to pipe whip and jet impingement loadings. In the DFSER, the staff noted that the COL applicant should use these criteria and methodology to postulate locations of pipe breaks and leakage cracks to ensure adequate protection against the dynamic effects of postulated ruptures of piping in the ABWR standard design. This was DFSER COL Action Item 14.1.3.3.7-1. SSAR Section 3.6.5.1, "Details of Pipe Break Analysis Results and Protection Methods," states that the COL applicant shall provide a summary of the dynamic analyses applicable to high-and moderate-energy piping systems in accordance with Section 3.6.2.5 of RG 1.70. This is acceptable.

In the DFSER, the staff also identified Open Items 3.6.2-2 and 14.1.3.3.7-1 regarding the edition of ANSI/ANS-58.2, "Design Basis for Protection of Light Water Nuclear Power Plants Against the Effects of Pipe Rupture," which is referenced in SSAR Section 3.6.2.2.1. Originally, there were inconsistencies between the criteria for evaluating the effects of fluid jets on essential SSCs specified in SSAR Section 3.6.2.3.1 and corresponding criteria specified in SRP 3.6.2 and ANSI/ANS-58.2, 1988 Edition. SSAR Table 1.8-21, "Industrial Codes and Standards Applicable to ABWR," specifies the current 1988 Edition of the ANSI/ANS-58.2 Standard. In addition, the criteria described in SSAR Section 3.6.2.3.1 are consistent with SRP Section 3.6.2 and ANSI/ANS-58.2, 1988 edition. This is acceptable; therefore, DFSER Open Items 3.6.2-2 and 14.1.3.3.7-1 are resolved.

By letter dated September 11, 1992, the staff provided GE with guidance for the use of a single-earthquake design for SSCs in the ABWR. This guidance proposes that the criteria for postulating pipe breaks in seismically designed, high-energy piping systems be based on normal and operational transients only, excluding earthquake loadings. SSAR Sections 3.6.1 and 3.6.2 and Tables 3.9.1 and 3.9.2 incorporate the staff guidance for the use of a single-earthquake for the postulation of pipe break locations. On the basis of this incorporation and the staff's evaluation of this issue in Section 3.1.1.3 of this report, the staff concludes that the ABWR criteria are consistent with staff positions discussed in Section 3.1.1 of this report, which is acceptable.

The staff also told GE that relaxation of the current BTP MEB 3-1, Revision 2, June 1987, Section B.1.c.(1)(b) criteria to the previous BTP MEB 3-1, Revision 1, July 1981 criteria would be acceptable. These criteria are for the postulation of pipe breaks in ASME Code, Section III, Class 1 high-energy fluid-system piping in areas other than containment penetration. This staff position stated is based on the following rationale:

> For ASME Code Class 1 piping, the NRC position for postulating pipe breaks is delineated in the Branch Technical Position MEB 3-1 of the SRP 3.6.2. Prior to issuance of Revision 2 of BTP MEB 3-1 in June 1987, breaks were postulated at intermediate locations between terminal ends of a pipe run if the maximum stress range as calculated by Code Eq. (10) > 2.4 S_m <u>and</u> either by Eq. (12) or Eq. (13) > 2.4 S_m. These stated were implemented in many plants operating today.
>
> In Revision 2 of BTP MEB 3-1, the same criteria are maintained for break exclusion in the containment penetration areas. However, for other areas, the criteria were revised requiring that breaks be postulated at any intermediate locations when only Eq. (10) exceeds 2.4 S_m. The use of Eq. (12) and Eq. (13) was eliminated.
>
> The staff reviewed the impact of BTP MEB 3-1, Revision 2, and found that the Revision 2 criteria are inconsistent in that they allow higher limits in the containment penetration areas than in other areas. It would appear that the break exclusion area should provide a margin greater than (or at least equal to) the margin for areas outside the break exclusion area. In addition, the Revision 2 criteria will result in a significant increase in the number of postulated pipe breaks, which may be counter-productive in terms of enhancing plant safety.

For these reasons, the staff concludes that the Revision 1 criteria related to allowing the use of Eq. (12) and Eq. (13) should be reinstated for the postulation of intermediate pipe breaks in ASME Code Class 1 piping systems. SSAR Section 3.6.2 includes the use of Revision 1 of BTP MEB 3-1 criteria in B.1.c.(1).(b). On the basis of this evaluation and a similar evaluation in Section 3.6.2 of this report, the staff concludes that these criteria are an acceptable deviation of applicable criteria in SRP 3.6.2, Revision 2.

During the audit performed at GE in San Jose, California, on March 23 through 27, 1992, the staff discussed with GE the as-yet-developed procedures for the ABWR for postulated pipe break analyses. This was a part of DFSER Open Item 3.6.2-1. SSAR, Appendix 3L, "Evaluation of Postulated Ruptures in High Energy Pipes," defines a procedure for evaluating dynamic effects of fluid dynamic forces resulting from postulated ruptures in high-energy piping systems. The four major steps in the evaluations include: (1) the identification of rupture locations and rupture geometry, (2) the design and selection of pipe whip restraints, (3) the procedure for dynamic time-history analysis with simplified models, and (4) the procedure for dynamic time-history analysis, using detailed piping models. The staff's review concludes that the procedure in Appendix 3L is consistent with applicable guidelines in SRP Sections 3.6.2 and 3.9.2 and are acceptable; therefore, this part of DFSER Open Item 3.6.2-1 is resolved. The remainder of DFSER Open Item 3.6.2-1 is discussed in Section 3.6.2 of this report.

3.12.7.1 High-Energy Piping Systems

Pipe whip need only be considered for those high-energy piping systems having fluid reservoirs with sufficient capacity to develop a jet stream. The criteria for determining high- and moderate-energy lines in SRP Section 3.6.1, Branch Technical Position (BTP) ASB 3-1, are adequately defined in Section 3.6.2.1 of the SSAR. All high-energy systems are listed in SSAR Tables 3.6-3 and 3.6-4.

3.12.7.2 Pipe Break Criteria Within the Containment Penetration Areas

Breaks are not postulated in the ABWR design for those portions of high-energy piping between the isolation valves outside and inside the containment that are designed to meet ASME Code, Section III, Article NE-1120, and the additional design guidelines in SRP Section 3.6.2, including BTP MEB 3-1, Revision 2, June 1987. These guidelines recommend that an augmented inservice inspection program be implemented for those portions of piping within the break-exclusion region. In the DFSER, the staff noted that the COL applicant should perform a 100-percent volumetric examination of circumferential and longitudinal pipe welds in the break-exclusion region during each inspection interval as defined in Article IWA-2400, ASME Code, Section XI. This was DFSER COL Action Item 14.1.3.3.7.2-1. SSAR Section 3.6.5.3, "Inservice Inspection of Piping in Containment Penetration Areas," states that the COL applicant shall perform an augmented inservice inspection program, as defined. This is acceptable.

3.12.7.3 Pipe Break Criteria Outside the Containment Penetration Areas

For ASME Code, Class 1, 2, and 3, and non-ASME seismic Category I high- and moderate-energy lines that are not in the containment penetration area, Section 3.6.2 of the SSAR provides the criteria for determining postulated break and crack locations and the methodology used to evaluate the dynamic effects of pipe whip, jet thrust, and jet impingement that result from such breaks.

SRP Section 3.6.2 guidelines state that if a structure separates a high-energy line from an essential component, the separating structure should be designed to withstand the consequences of the pipe break in the high-energy line that produces the greatest effect at the structure, irrespective of the fact that the pipe break criteria in SRP Section 3.6.2 might not require such a break location to be postulated.

The ABWR structures are designed to withstand the dynamic effects of pipe breaks where the pipe rupture criteria require break locations to be postulated. In addition, for areas where physical separation of redundant trains is not practical, a high-energy line separation analysis (HELSA) will be performed by the COL applicant to determine which high-energy lines meet the spatial separation requirements and which lines require further protection. For the HELSA evaluation, which is discussed in Section 3.6.1.3.2.2 of the SSAR, no particular break points are evaluated. Breaks are postulated at any point in the piping system and any structure identified as necessary by the HELSA evaluation are designed for worst-case loads. This was DFSER COL Action Item 14.1.3.3.7.3-1. SSAR Section 3.6.5.1, Item (8), states that the COL applicant will perform the HELSA as described. This is acceptable.

Using this HELSA evaluation, the staff finds that an adequate level of protection is provided to ensure that the safety-related function of components, systems, and equipment will not be adversely impacted by a postulated high-energy line break. Plant arrangement provides physical separation to the extent practical and the HELSA evaluation ensures that no more than one redundant train can be damaged. If damage could occur to more than one division of a redundant safety-related system within 9.14 m (30 ft) of any high-energy piping, other protection devices such as barriers, shields, enclosures, deflectors, or pipe whip restraints are used. When necessary, the protection requirements are met through the use of walls, floors, columns, abutments, and foundations. Thus, the staff finds that the HELSA criteria satisfy the intent of the SRP 3.6.2

guideline by ensuring that structures are adequately designed to withstand the consequences of a worst-case pipe break with no adverse impact on the safety-related function of systems, components, and equipment and are acceptable.

3.12.7.4 Conclusions

The staff concludes that the criteria for postulating pipe rupture and crack locations and the methodology for evaluating the subsequent dynamic effects resulting from these ruptures comply with SRP Section 3.6.2, meet GDC 4 and, are therefore, acceptable. The staff's conclusion is based on the evaluations in Section 3.12.3.7 of this report and the following:

The proposed pipe rupture locations are adequately determined, using the staff-approved criteria and guidelines. GE has sufficiently and adequately defined the design methods for high-energy mitigation devices and the measures to deal with the subsequent dynamic effects of pipe whip and jet impingement to provide adequate assurance that if the COL applicant completes the high-energy line break analyses, the ability of safety-related SSCs to perform their safety functions will not be impaired by the postulated pipe ruptures.

The provisions for protection against the dynamic effects associated with pipe ruptures of the RCPB inside the containment and the resulting discharging fluid provides adequate assurance that design-basis LOCAs will not be aggravated by the sequential failures of safety-related piping, and that the performance of the emergency core cooling system will not be degraded as a result of these dynamic effects.

The COL applicant is responsible for the arrangement of piping and restraints and the final design considerations for high- and moderate-energy fluid systems inside and outside the containment, including the RCPB. The COL applicant should use these staff-approved high-energy line break criteria and these guidelines to ensure that the SSCs important to safety that are in close proximity to the postulated pipe ruptures will be protected. Using these will ensure that the consequences of pipe ruptures will be adequately mitigated so that the reactor can be safely shut down and be maintained in a safe-shutdown condition in the event of a postulated rupture of a high- or moderate-energy piping system inside or outside the containment.

3.12.8 Leak-Before-Break Criteria

SSAR Section 3.6.3 and Appendix 3E provide a description of the evaluation procedures for an LBB methodology. Since no LBB analysis was submitted for staff review and approval, the use of the LBB approach has not been pre-approved by the staff in the ABWR design certification phase. Hence, it is a design option for the COL applicant to consider in lieu of performing high-energy line break analyses as discussed in Section 3.12.7 of this report.

GDC 4 permits the application of the LBB methodology to piping systems. It states, in part, that "dynamic effects associated with postulated pipe ruptures in nuclear power units may be excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping." The analyses referred to in GDC 4 (52 Federal Register, 41288-41295, October 27, 1987) should be based on specific plant data, such as piping geometry, materials, piping loads, and pipe support locations. The staff must review the LBB analyses for specific piping designs before the applicant can exclude the dynamic effects from the design basis for the piping system.

The staff concludes that COL applicants seeking approval of the LBB approach for high-energy piping systems in the ABWR standardized plant must submit to the NRC staff an LBB plant-specific analysis in accordance with GDC 4. The staff recognizes that the LBB technology is continually evolving, therefore, the staff will review LBB requests for the ABWR plant on a case-by-case basis, using the staff's methodology and acceptance criteria in effect at the time of the submittal.

3.12.9 Generic Piping Design ITAAC

In Section 3.3 of the CDM, dated June 17, 1992, GE provided its ITAAC for piping design. Table 3.3 therein identified 12 certified design commitments (design elements) for the ABWR piping design and the corresponding ITAAC. In a meeting held at GE's offices in San Jose, California, on January 11 through 21, 1993, the staff and GE reached agreement on the resolution of comments from the industry/NUMARC on the generic piping design ITAAC (also referred to as the Piping DAC). The piping design description has been substantially expanded to include additional certified design commitments. The ITAAC design commitments have been reduced in number to consolidate those design commitments that are implicit in the ASME Code, Section III, requirements and to eliminate some design criteria that were deemed not appropriate for inclusion into ITAAC. GE gave the NRC additions and changes to the SSAR to support the piping DAC/ITAAC changes.

The staff's evaluation of the certified design commitments and ITAAC for piping design is given in the following sections.

3.12.9.1 Fatigue

GE provided a certified design commitment that the piping will be designed for a fatigue life of 60 years. The COL applicant will perform a fatigue analysis for ASME Code Class 1 piping systems in accordance with the applicable requirements of ASME Code, Section III. Section III rules will be followed for ASME Code Class 2 and 3 piping, using a stress range reduction factor of 1.0 for those piping systems expected to experience less than 7000 thermal cycles in their 60-year design life.

The COL applicant will be required to ensure that the fatigue analysis meets the ASME Code requirements for the 60-year design life. The acceptance criteria for the fatigue design of ASME Code Class 1 piping will be that the cumulative usage factor is less than 1.0 as specified in the ASME Code, Section III.

The staff finds that a fatigue analysis of safety-related piping is a necessary certified design commitment to ensure the integrity of the RCPB and the ability of the piping systems to perform their safety function for a 60-year design life. The design acceptance criterion for a cumulative usage factor to be less than 1.0 is consistent with current ASME Code requirements for fatigue evaluation as stated in Subparagraph NB-3222.4. The COL applicant will consider in its fatigue analysis the environmental effects as discussed in Section 3.12.5.7 of this report.

The inspection of the ASME Code certified stress report, including the fatigue analyses ensures that the ASME Code requirements for fatigue will be satisfied. However, in the DFSER, the staff reported that an additional certified design commitment is needed for any ASME Code Class 2 and 3 piping system that is expected to experience 7000 or more thermal stress cycles in its 60-year design life. For any such piping, the COL applicant should use a stress reduction factor of less than 1.0 as required by Subparagraph NC/ND-3611.2 of the ASME Code, Section III. In addition, if an ASME Code Class 1 fatigue evaluation is required for ASME Code Class 2 and 3 piping systems, as discussed in Section 3.12.5.8 of this report, a cumulative usage factor of 1.0 should be met and environmental effects should be considered. This was DFSER Open Item 14.1.3.3.9.1-1.

Section 3.9.3.1 of the SSAR provides information on loading combinations, design transients, and stress limits for ASME Code Class 1, 2, and 3 components and supports. SSAR Section 3.9.3.1.19 states that the Class 1, 2, and 3 piping is constructed in accordance with the ASME Boiler and Pressure Vessel Code, Section III. For Class 1 piping, stresses are calculated on an elastic basis and evaluated in accordance with NB-3600 of the ASME Code, Section III. For Class 2 and 3 piping, stresses are calculated on an elastic basis and evaluated in accordance with NC/ND-3000 of the Code. From GE's commitment, the staff concluded that any Class 2 or 3 piping that is expected to experience 7000 or more thermal stress cycles must be designed to meet the allowable expansion stress range defined in NC/ND-3611.2(e) and based on the appropriate stress range reduction factor given in Table NC/ND-3611.2(e)-1 for the total number of full temperature cycles in its 60-year design life. The staff also concluded that for any piping system in which an ASME Code Class 1 fatigue evaluation will be performed, the evaluation must be performed in accordance with the requirements of NB-3653 and a cumulative usage factor of 1.0 must not be exceeded, considering the total number of stress cycles in its 60-year design life. In SSAR Section 3.9.3.1.1.7, GE commits to performing additional evaluations for environmental effects on fatigue design. The staff evaluation of this issue is discussed in Sections 3.12.5.7 and 3.12.5.8 of this report.

On the basis of these commitments, the staff concludes that there is reasonable assurance that the design of the ABWR piping systems will be adequately evaluated for fatigue effects.

3.12.9.2 Pipe-Mounted Equipment Allowable Loads

GE provided a certified design commitment that the loads imposed by the piping system on pipe-mounted equipment will meet the vendor allowable loads. The COL applicant will inspect the design documents and document that the as- designed interface loads meet the vendor's specified allowable loads. This was DFSER COL Action Item 14.1.3.3.9.2-1. SSAR Section 3.9.3.1.21, "Pipe-Mounted Equipment Allowable Loads," states that the COL applicant shall inspect the piping design reports and document that the pipe applied loads on attached equipment are less than the equipment vendor's specified allowable loads.

The staff finds it necessary to ensure that the calculated loads imposed by the piping on the equipment nozzles and other attachment interfaces are within the vendor's recommended allowable values. This verification will ensure that the equipment and supports will function as intended under normal operating, transient, and accident conditions.

On the basis of these commitments for a COL applicant to verify that the calculated piping loads are within the equipment and interface allowable loads, the staff concludes that there is reasonable assurance that the ABWR pipe-mounted equipment and piping attachment interfaces will adequately satisfy the vendor interface allowable limits to ensure that the equipment can perform its intended safety functions under normal, operating, transient, and accident loading conditions.

3.12.9.3 Piping Analysis Methods

GE provided a certified design commitment that would require that the analytical methods and load combinations be referenced or specified in a certified stress report. The COL applicant will use a suitable dynamic analysis method or an equivalent static load method in the analysis of the piping system.

The analytical methods to be used to complete the ABWR piping design will ensure the pressure integrity, structural integrity, and the functional capability of the piping system under normal operating and accident loading conditions and will use a suitable dynamic analysis or an equivalent static analysis method as approved by the staff. The analysis methods approved by the staff for the ABWR piping design are discussed in Section 3.12.3 of this report. The key analysis input parameters approved by the staff for the ABWR piping analysis are discussed in Section 3.12.5 of this report.

On the basis of these commitments for a COL applicant to use the above staff-approved analysis methods and input parameters for the ABWR piping analyses, the staff concludes that there is reasonable assurance that the ABWR analysis methods are adequate to ensure the pressure integrity, structural integrity, and functional capability of the piping.

3.12.9.4 High-Energy Line Break Analysis

GE provided a certified design commitment that would require an analysis demonstrating that safety-related systems, components, and structures are protected against the dynamic effects associated with the postulated rupture of high-energy piping systems.

The COL applicant will prepare a pipe rupture analysis report or an LBB analysis report to verify that the safety of the plant will not be adversely affected by the dynamic effects resulting from the postulated pipe breaks. For those impacted components needed to safely shut down the plant, the ASME Code requirements for faulted plant conditions and operability limits must be met. Pipe rupture mitigation devices (e.g., pipe whip restraints and jet impingement shields) will be used to restrain the whipping pipe and deflect the blowdown loads. The COL applicant will inspect to verify the existence of a pipe break analysis report or LBB report as stated in SSAR Section 3.6.5.1. The COL applicant will also inspect the as-built high-energy pipe break mitigation features as discussed in SSAR Section 3.6.4.

A pipe rupture analysis will be completed by the COL applicant to demonstrate that safety-related SSCs will be protected against the dynamic effects of a postulated pipe break, using the methods described in Sections 3.6.2 and 3.12.7 of this report. As an alternative, the COL applicant may submit a request for staff approval to eliminate breaks, using an LBB approach as discussed in Section 3.12.8 of this report.

On the basis of these commitments for a COL applicant to perform high-energy line break analyses, using the staff-approved analysis methods, and to verify the results of the analyses, the staff concludes that there is reasonable assurance that the safety-related SSCs in the ABWR are adequately protected against the dynamic effects of postulated HELBs.

3.12.9.5 Functional Capability

GE provided a certified design commitment that all ASME Code Class 1, 2, and 3 piping systems essential for the safe shutdown of the plant must be designed to ensure that they will maintain sufficient dimensional stability to perform their required function under all loading conditions. In Section 3.12.5.12 of this report, the staff evaluated the stress limits GE specified to ensure the functional capability of safety-related piping systems. In no case will the piping stress exceed the primary stress limits designated for Service Level D in the ASME Code, Section III. The Service Level D limits are 3.0 S_m (not to exceed 2.0 S_y) for ASME Code Class 1 piping and 3.0 S_h (not to exceed 2.0 S_y) for Class 2 and 3 piping.

The staff finds that the limits specified by GE to ensure functional capability of piping as discussed in Section 3.12.5.12 of this report are acceptable. The use of Service Level D limits (not to exceed 2.0 S_y) are consistent with the staff recommendations as documented in NUREG-1367, "Functional Capability of Piping Systems," for ensuring the functional capability of piping systems that were based on high-level dynamic tests sponsored by the EPRI and the NRC staff.

On the basis of these commitments for a COL applicant to limit piping stresses in the certified stress report to the design acceptance criteria discussed above, the staff concludes that there is reasonable assurance that the piping is capable of performing its safety function under all normal operating, transient, and accident conditions.

3.12.9.6 Analytical Modeling of Piping

GE provided a certified design commitment to verify the piping analysis modeling technique for the computer code to be used by the COL applicant to complete its piping stress analyses. The piping analysis model must address the key parameters needed to ensure adequate static and dynamic characteristics of the piping system. The key parameters for the piping model are discussed in Section 3.12.4.2 of this report. The computer program and the modeling techniques must be evaluated, using the NRC benchmark program discussed in Section 3.12.4.3 of this report.

The COL applicant will verify the sufficiency of the computer code and modeling techniques in conjunction with the piping benchmark program.

On the basis of these commitments for a COL applicant to verify that the piping benchmark results are within the acceptable range of values specified in the benchmark program, the staff concludes that there is reasonable assurance that the computer code and analytical modeling techniques to be used to complete the ABWR piping design and analyses are adequate.

3.12.9.7 ASME Code Classification

GE provided a certified design commitment that the ABWR piping, its appurtenances, and its supports must satisfy the ASME Code class, seismic category, and QG requirements commensurate with their classification. The COL applicant will review the ASME Code-required design documents for installed components to verify the completion of a certified stress report and related Code-required documents.

On the basis of this commitment for a COL applicant to verify the existence of Code-required documents, the staff concludes there is reasonable assurance that the piping and its subcomponents will be adequately designed, fabricated, and examined in accordance with the applicable ASME Code requirements.

3.12.9.8 Fracture Toughness

GE provided a certified design commitment that the piping systems made of ferritic material must not be susceptible to brittle fracture. Only intrinsically tough grades of ferritic materials will be used. The COL applicant will perform fracture toughness tests in accordance with the requirements of the ASME Code, Section III.

On the basis of these requirements for a COL applicant to verify that ferritic materials satisfy the requirements of ASME Code, Section III, the staff concludes that there is reasonable assurance that the material for piping systems will be adequately specified to preclude brittle fracture under pressure loadings for the expected service conditions.

3.12.9.9 Cracking in Stainless Steel Piping

GE provided a certified design commitment that the fabrication process for piping systems made of austenitic stainless steel will be selected to minimize the possibility of cracking during their 60-year design life. Special chemical, fabrication, handling, welding, and examination requirements will be satisfied to minimize the potential for cracking. The guidelines in NUREG-0313, Revision 2, will be followed as stated in SSAR Section 5.2.3.4.

On the basis of these commitments for a COL applicant to use the material and processes that satisfy the ASME Code and special requirements, the staff concludes that there is reasonable assurance that the austenitic stainless steel piping systems will be adequately fabricated to minimize the potential for cracking during service.

3.12.9.10 As-Built Piping Verification

GE provided a certified design commitment that the as-built piping system must be reconciled with the certified piping design. The COL applicant will perform an inspection to verify the pipe routing configurations, as well as the location, size, and orientation of piping supports, valves, and equipment, and to identify deviations from the as-designed condition as described in SSAR Section 3.9.3.1.20. The piping configuration and component location, size, and orientation will be within the specified tolerances. Deviations (outside the tolerances) will be evaluated to ensure that the vendor-allowable loads and ASME Code, Section III, stress limits are satisfied.

The tolerances to be used for the reconciliation of the as-built installation of piping systems will be obtained from the EPRI report, "Guidelines for Piping System Reconciliation (NCIG-05, Revision 1)," NP-5639 dated May 1988. The staff's acceptance of this approach is documented in a letter from G. Arlotto (NRC) to W. Weber (Nuclear Construction Issues Group) dated February 3, 1988. The staff's endorsement of the EPRI report is subject to the restriction that (1) the acceptable as-built piping tolerances not be increased beyond those stated in NP-5639 and (2) they be limited to the piping systems analyzed using linear-elastic methods and qualified on the basis of the staff-approved design criteria specified in this report.

On the basis of these commitments for a COL applicant to verify that the installation tolerances are satisfied and that all deviations are reconciled using the staff-approved methods and design acceptance criteria discussed in this report, the staff concludes that there is reasonable assurance that the ABWR piping systems will be constructed in accordance with the design documents.

3.12.9.11 Pressure Integrity

GE provided a certified design commitment that the ASME Code Class 1, 2, and 3 piping will be designed to the requirements of ASME Code, Section III, as discussed in Section 3.12.2.1 of this report to retain its pressure integrity for its 60-year design life.

The COL applicant will inspect the ASME Code-required documents to verify that ASME Code, Section III, requirements are satisfied for ASME Code Class 1, 2, and 3 piping systems. The COL applicant will perform hydrostatic pressure tests of the ASME Code Class 1, 2, and 3 piping in accordance with the ASME Code requirements. In addition, the COL applicant will ensure that the low-pressure piping systems that interface with the RCS pressure are adequately designed to withstand full RCS pressure in the event of an intersystem LOCA as discussed in Section 3.12.5.19 of this report.

On the basis of these commitments for a COL applicant to perform these inspections and tests, the staff concludes that the ASME Code Class 1, 2, and 3 piping systems will be designed, constructed, and tested to ensure their pressure integrity in service under normal operating, testing, transient, and accident conditions.

3.12.9.12 Interferences

GE provided a certified design commitment that piping will be designed with adequate clearances to preclude interferences with nearby SSCs resulting from piping displacements.

The COL applicant will verify that the maximum calculated pipe deflections under normal operating, transient, and accident conditions do not exceed the minimum specified clearances between the piping and nearby SSCs as stated in SSAR Section 3.9.3.1.22.

On the basis of these commitments for a COL applicant to verify that maximum calculated pipe deflections are within the minimum specified clearances or do not impact nearby SSCs, the staff concludes that there is reasonable assurance that the piping deflections under normal operating, transient, and accident conditions will not cause interferences with nearby SSCs.

3.12.9.13 Erosion-Corrosion

GE provided a certified design commitment that piping systems will be designed to minimize the effects of erosion-corrosion. Erosion-corrosion will be controlled as discussed in SSAR Section 5.2.3.2.2.3. The EPRI-developed code CHECMATE will be used for two-phase environments to predict corrosion rates and to identify areas where design improvements to the material selection, hydrodynamic conditions, oxygen content, and temperature may be required to ensure adequate margin for extended piping performance.

On the basis of this certified design commitment for a COL applicant to evaluate the erosion-corrosion effects in piping systems, the staff concludes that there is reasonable assurance that the effects of erosion-corrosion are minimized in the piping design for the 60-year plant life.

3.12.9.14 Conclusions

In the DFSER, the staff requested that GE revise the CDM for piping design to ensure that all its certified design commitments in Table 3.3 of the CDM are included in the design description. This was DFSER Open Item 14.1.3.3.9.13-1. In a revision to Section 3.3, "Piping Design," of the Tier 1 CDM document, GE expanded the design description to include the certified design commitments. GE has also included this information in the final CDM. On the basis of this evaluation, DFSER Open Item 14.1.3.3.9.13-1 is resolved.

The staff concludes that the certified design commitments and ITAAC for piping design incorporate the staff-approved piping design criteria and analysis methods for ensuring that the piping systems will be adequately designed to perform their safety-related functions for all postulated combinations of normal operating, operating transient, and accident conditions, and provide reasonable assurance that the piping systems are built in conformance with the certified design.

3.12.10 Overall Conclusions

The staff concludes that GE has provided sufficient information in the SSAR for the staff to reach a safety determination in ABWR piping and pipe support design. The staff's conclusion is based on the following:

(1) GE satisfies 10 CFR Part 50 requirements by identifying applicable codes and standards, design and analysis methods, design transients and load combinations, and design limits and service

conditions to ensure adequate design of all safety-related piping and pipe supports in ABWR for their safety functions.

(2) GE satisfies 10 CFR Part 52 requirements by providing reasonable assurance that the piping systems will be designed and built in accordance with the certified design. The implementation of these preapproved methods and satisfaction of the acceptance criteria will be verified through the performance of the ITAAC by the COL applicant to ensure that the as-constructed piping system are in conformance with the certified design for their safety functions.

(3) GE satisfies 10 CFR Part 100, Appendix A, requirements by designing the safety-related piping systems, with reasonable assurance to withstand the dynamic effects of earthquakes with appropriate combination of other loads of normal operation and postulated events with adequate margin for ensuring their safety functions.

Any change to the commitments involving the piping analysis methodology discussed in Section 3.12 of this report would involve an unreviewed safety question and, therefore, requires NRC review and approval prior to implementation. Any requested change to these commitments must either be specifically described in the COL application or be submitted for license amendment after COL issuance.

4 REACTOR

4.1 General

The reactor assembly consists of the reactor pressure vessel, pressure containing appurtenances that include control rod drive (CRD) housings, in-core instrumentation housing, and the head vent and spray assembly. The reactor pressure vessel includes the reactor internal pump (RIP) casing and flow restrictors in each of the steam outlet nozzles and the shroud support and pump deck that form the partition between the RIP suction and discharge. The design and description of the reactor pressure vessel are discussed in Section 5.3 of this report.

The major reactor internal components are the core (fuel, channels, control blades, and instrumentation), the core support structure (including the shroud, top guide, and core plate), the shroud head and steam separator assembly, the steam dryer assembly, the feedwater spargers, and the core flooding spargers. Except for the Zircaloy in the reactor core, these reactor internals are stainless steel or other corrosion-resistant alloys. The fuel assemblies (including fuel rods and channel), control blades, shroud head and steam separator assembly, and steam dryers and in-core instrumentation dry tubes are removable when the reactor vessel is opened for refueling or maintenance.

A fuel and control rod design and core loading pattern typical of many currently operating boiling water reactors (BWRs) was used as the basis for the core design for the first cycle and for the system response analysis for standard safety analysis report (SSAR) Chapters 6 and 15. These elements of the core design meet criteria approved by the Nuclear Regulatory Commission (NRC) as presented in SSAR Appendices 4B, 4C, and 4D.

4.2 Fuel System Design

The staff reviewed the fuel system design in accordance with Standard Review Plan (SRP) Section 4.2, which includes the acceptance criteria of General Design Criteria (GDC) 10, 12, and 27, as discussed below and the reference to fuel designs approved by the NRC or to fuel that meets acceptance criteria approved by the NRC for GE Nuclear Energy (GE) fuel (NEDE-31152P, "GE Fuel Bundle Designs Evaluated with GESTAR - Mechanical Analysis Bases," SSAR Appendix 4B and SSAR Reference 4.2-2).

(1) GDC 10 requires that acceptable fuel design limits be specified that are not to be exceeded during normal operation, including the effects of anticipated operational occurrences.

(2) GDC 12 requires that power oscillations which could result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed.

(3) GDC 27 requires that the reactivity control systems have a combined capability, in conjunction with poison addition by the emergency core cooling system, of reliably controlling reactivity changes under postulated accident conditions, with appropriate margin for stuck rods.

The fuel for the advanced boiling water reactor (ABWR) is similar in design, including most geometrical and material details, to the fuel commonly used for reloading most BWR cores. The fuel design bases, limits, analysis methodologies, and evaluations for the ABWR also are the same as those used and approved for initial loading and reloading of previous BWR cores. The ABWR fuel and core design provides the basis for the representative first cycle power distributions and rod patterns as presented in SSAR Appendix 4A and as used in the safety analyses of SSAR Chapters 6 and 15. The ABWR Cycle 1 core design, along with the fuel and control rod design (see below) are designated Tier 2 items and no inspections, tests, analysis, and acceptance criteria (ITAAC) are involved. However, pertinent elements of the fuel and control rod design criteria (SSAR Appendices 4B and 4C) are Tier 1 items and are included in the design descriptions. GE submitted the design description for the fuel and control rod. The adequacy and acceptability of these design descriptions are evaluated in Section 14.3 of this report.

Although the areas are Tier 2, the staff review has concluded that the fuel and control rod design criteria (of SSAR Appendices 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 this 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 design is requested for the first cycle, the COL applicant will be required to submit for staff review that specific fuel, control rod, and core design analysis and corresponding safety analysis described in SSAR Chapters 6 and 15. The review for the fuel, control rod, and core design will be based on fuel and control rod design criteria as described below.

Although it is not specifically referenced in the SSAR, GE customarily presents relevant generic information relating to core initial and reload cycle designs and analyses, including methodology related to fuel and control rod thermal-mechanical, nuclear, and thermal-hydraulic phenomena, in the licensing topical report NEDE-24011-P, "General Electric Standard Application for Reactor Fuel" (GESTAR II) (proprietary). Currently approved GE BWR fuel designs are described in the GE report NEDE-31152P, "General Electric Fuel Bundle Designs Evaluated with GESTAR-Mechanical Analysis Bases" (proprietary). GE states in SSAR Section 4.2 that BP 8 x 8R fuel is used for the reference core design. The BP 8 x 8R is a standard fuel design frequently used in current BWRs for cycle reloads. An eight-by-eight array of prepressurized fuel pins and a barrier coating on the inner surface of the fuel cladding are used. A discussion and NRC staff approval of features of the fuel may be found in GESTAR II, Supplement for United States, Appendix C, and in Reference 4.2-2 of the ABWR SSAR. The fuel assembly descriptions are in Section AY of NEDE-31152P, Volume 3, Revision 3. These reviews covered all of the areas in SRP Section 4.2. Compliance of the fuel design with the fuel design criteria (discussed below) is presented in SSAR Appendix 4D, and the design is shown to meet all requirements. This fuel is acceptable for the ABWR.

GE provided the primary reference to acceptable fuel design for the ABWR in SSAR, Appendix 4B. This appendix provides a set of acceptance criteria to be satisfied by new fuel designs for the ABWR. These criteria were developed from preceding generic work in response to the NRC staff generic request that such acceptance criteria be established by each fuel vendor. With the NRC approval of the generic criteria, new fuel designs (or changes) satisfying the criteria would not require explicit staff review for current reactors.

In response to the NRC staff generic request, GE submitted Amendment 22 to GESTAR II, containing proposed fuel licensing acceptance criteria for current BWRs. These criteria include considerations of fuel thermal-mechanical, nuclear, and thermal-hydraulic aspects of design analyses. The NRC staff and the Committee to Review Generic Requirements reviewed and approved these criteria. The staff safety evaluation report (SER) accepting these criteria can be found in GESTAR II, Revision 10, Supplement for United States, Appendix C. However, the staff has noted deficiencies in the generic fuel licensing criteria. In its audit summary, "Audit Team Audit of GE II Fuel Design Compliance With NEDE-24011-P-A," dated March 25, 1992, the staff noted the lack of a burnup limit requirement in the fuel criteria at that time. A parallel fuel burnup problem area (DFSER Confirmatory Item 4.2-1) for the ABWR is discussed below.

For the ABWR, GE has not directly referenced GESTAR II or the fuel criteria amendment. Instead GE has provided SSAR Appendix 4B which provides a similar set of criteria for the ABWR. Since the Appendix 4B criteria are essentially identical to the criteria approved by the staff for GESTAR II, they are generally acceptable. However, some additions or restrictions are necessary. The staff review of the Appendix 4B criteria and the staff audit of the generic criteria have indicated that the following restrictions are necessary for the long-term use of the criteria:

- NRC-approved analytical models and analysis procedures of General Criterion (1) to be used without further review must be limited to those referenced in GESTAR II, Revision 10, or previous revisions. Methods developed and approved in later GESTAR II revisions will not automatically apply to the ABWR and will have to be specifically reviewed and approved for ABWR use.

- Fuel burnup limits must be specified and justified on the basis of material properties versus exposure data for each fuel type used in the ABWR and may be extended only with NRC review and approval. This was identified as DFSER Confirmatory Item 4.2-1.

In response to the staff request for burnup limit and review statements in the fuel criteria, GE stated that such a criterion is unnecessary and not a safety issue. However, they proposed the following statement for the SSAR.

> Burnup limits will be specified for each fuel type used in the ABWR. The current maximum exposure limit for any GE fuel design is 70 Giga watts days metric ton of uranium (GWd/MTU) (6,048 MJ/gu (3,214 * 10EG BTU)LbU peak pellet exposure (~60 GWd/5,184 MJ/gU (2,755*10E6 BTU/LbU or 60 GWd/MTU) rod average exposure). Any extension to this maximum exposure limit in excess of 10 GWd/MTU will be submitted to the NRC for review and approval based on the available supporting materials properties vs. exposure information and planned surveillance program. In no event will the GE fuel design maximum exposure limit required by the NRC be lower than the maximum of all exposure limits approved by the NRC for LWR fuel vendors.

The staff found that the proposed fuel burnup limit in this submittal is higher than that previously approved for GE (5,184 MJ/gU (2,755*10E6 BTU/LbU or 60 GWd/MTU), peak pellet), that an unreviewed extension of 10 GWd/MTU is excessive, and that limits approved for other vendors do not necessarily apply to GE fuel without specific review for GE. The staff considers the burnup limit a safety question and has several fuel operating concerns at burnup levels above those currently approved for BWRs (about 5,184 MJ/gU (2,755*10E6 BTU/LbU or

60 GWd/MTU) peak-pellet burnup). These concerns impact normal operation, off-normal transients, and accidents.

A brief summary of the concerns are:

- no prototypical LWR operating data above about 5,357 MJ/gU (2,847*10E6 BTU/LbU or 62 GWd/MTU)
- no fuel transient data above about 3,974 MJ/gU (2,112*10E6 BTU/LbU or 46 GWd/MTU)
- significant drop in cladding ductility observed at about 5,184 MJ/gU (2,755*10E6 BTu/LbU or 60 GWd/MTU)
- decrease in fuel thermal conductivity and changes in other physical properties
- changes in loss-of-coolant accident (LOCA) rod behavior at higher burnup levels
- fission gas release

Other issues that need to limits be addressed on a design specific basis for an extension in fuel burnup are the following:

- assembly and cladding corrosion
- fuel rod and assembly axial growth
- grid spacer spring relaxation

Since GE has provided a fuel burnup limit, the staff considers DFSER Confirmatory Item 4.2-1 resolved. However, GE has been requested to augment its proposed fuel design criteria for the ABWR to include fuel burnup limits and to indicate that these limits may be extended only with NRC review and approval. In its later submittal on this subject, GE omitted previous objections that the staff has not imposed explicit burnup limits in the past. (They had appeared indirectly in the maximum average planar linear heat generation rate (MAPLHGR) TS maximum burnup listing, but this would effectively disappear in the core operating limit report.) However, they proposed a peak burnup limit of 6,048 MJ/gU (3,214*10E6 BTU/LbU or 70 GWd/MTU (peak pellet) which is greater than the limit previously approved by the staff (60), and they also proposed that extensions to approved values that would not need staff review and approval should be at least 10 GWd/MTU, which the staff considers to be highly excessive. This was an Open Item F4.2-1 for the ABWR review. In response to this open item, GE provided changes to the fuel licensing acceptance criteria Section 4B.3(2)(j), "Submittal Supporting Accelerated ABWR Schedule-Response to Open Item F4.2-1," dated February 4, 1994, which now states that (1) fuel burnup limits will be specified for fuel used in the ABWR design, (2) the current limit for the ABWR fuel is 60 GWd/MTU rod average exposure, and (3) any extension of this limit will be submitted to the NRC for review and approval. These changes provide an acceptable resolution of the need for burnup restrictions indicated in the staff review. The 60 GWd/MTU limit is acceptable based on the staff review of high performance data for GE fuel during the NRC audit of the fuel design process for the GE 11 fuel referenced above. The data supporting the high burnup performance that were examined during the audit included GE 8 x 8 fuel of the type used in the ABWR reference core. This submittal resolved Open Item F4.2-1.

With approval of the ABWR fuel criteria, new ABWR fuel designs (or changes) satisfying the criteria would not require explicit staff review, other than that required by its use by a COL applicant for the first cycle core loading.

Similar to the presentation of the ABWR fuel design, GE has provided a specific design for the control rod. This design was used in the safety analyses of SSAR Chapters 6 and 15. GE also has proposed control rod design criteria, similar in concept to those for the fuel designs, to be used as a basis for the proposed control rods or future new design submittals. Just as for the fuel design, the specified control rod design used in the ABWR safety analyses will constitute, based on the staff review and approval, an approved design that may be used by the COL applicant for the first cycle without further staff review. If the COL applicant changes the design, the staff will require new submittals for review and approval.

The ABWR control rod design has, for the most part, the same geometrical and material design characteristics of those approved and used for current reactors for the first cycle and for replacement, including current improvements for corrosion-cracking control. The ABWR design differs significantly from current control rod designs only in that it does not provide a velocity limiter for the rod drop accident event. GE believes that this event is not credible because information on the blade-drive uncoupling signal will be provided in the new ABWR rod drive design. In any case, the velocity limiter has become much less significant with axial reactivity shaping Gadolinia burnable poison in the fuel than with the early fuel designs that required the velocity limiter. Because of the burnable poison distribution or, later in the cycle, burnup distribution, most of the available reactivity increase from the dropped rod is inserted in a short distance and appears in the transient without significant dependence on the velocity limiter action. NRC consultant analyses (Brookhaven National Laboratory (BNL) - NUREG-36891, "Effects of Rod Worth and Drop Speed on the BWR Off-Center Rod Drop Accident," and ANS Transactions, November 1985) have shown little sensitivity to drop velocity. Thus removal of the velocity limiter is acceptable.

In SSAR Appendix 4C, GE submitted the set of control rod licensing acceptance criteria for the ABWR. These criteria describe the safety-related functional performance requirements for the control rods. The staff reviewed these criteria in Appendix 4C and proposed additions and modifications necessary to provide acceptable criteria. The changes which were submitted in the GE response were (1) removal of a statement from the general criteria indicating that a control rod design meeting the criteria did not require specific NRC review and (2) indicating that surveillance programs are to be implemented when changes in design features could impact the control function. Also added to the bases for the criteria were (1) inclusion of irradiation effects to the stress and strain limits, (2) further details of inspection of lead depletion rods with new design features, and (3) inclusion of crudding, crevices, and stress corrosion effects upon control rod material. The GE responses were responsive and satisfactory, and these criteria, as revised, were acceptable. This was DFSER Confirmatory Item 4.2-2. GE incorporated the appropriate changes have been in SSAR Amendment 31. The staff finds it to be acceptable. Therefore, DFSER Confirmatory Item 4.2.2 is resolved.

Current GE control rod designs that have been reviewed and approved by the NRC are suitably adaptable for the ABWR (when modified by the elimination of the velocity limiter). GE has provided a description of the reference control rod blade design in SSAR Section 4.2. GE has provided an evaluation of the design via a comparison with the Appendix 4C criteria. GE has revised SSAR Section 4.2.3.2.2.1 to indicate they have completed the evaluation of the control rod design, based on the criteria of SSAR Appendix 4C. GE has stated that the control rod evaluations described in Section 4.C.3 have been completed for the reference control rod design, and the criteria are satisfied. The control rod evaluation, performed by GE using proprietary Japanese data, was not submitted for staff review but is available for audit. It was not necessary to audit this data because the reference control rod blade design is similar to previously approved GE control rod designs. As discussed above, the staff finds GE's conclusions to be reasonable. The control rod design and evaluation were Open Item 14 in the draft safety evaluation report (DSER), and DFSER Confirmatory Item 4.2-3, respectively. Both items are now resolved.

The staff concludes that with approval of the specific fuel design type and of the fuel licensing acceptance criteria of SSAR Section 4B, as indicated above, and with the staff acceptance of the control rod design criteria in SSAR Section 4C, the ABWR core design approach meets the requirements of SRP Section 4.2 and is acceptable.

4.3 Nuclear Design

The staff reviewed the nuclear design in accordance with SRP Section 4.3 which includes the relevant requirements of the GDC related to the reactor core and reactivity control systems. The relevant requirements are as follows:

(1) GDC 10 requires that acceptable fuel design limits be specified that are not to be exceeded during normal operation, including the effects of anticipated operational occurrences.

(2) GDC 11 requires that in the power operating range, the prompt inherent nuclear feedback characteristics tend to compensate for a rapid increase in reactivity.

(3) GDC 12 requires that power oscillations which could result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed.

(4) GDC 20 requires automatic initiation of the reactivity control systems to assure that acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and to assure automatic operation of systems and components important to safety under accident conditions.

(5) GDC 25 requires that no single malfunction of the reactivity control system (this does not include rod ejection or dropout) causes violation of the acceptable fuel design limits.

(6) GDC 26 requires that two independent reactivity control systems of different design be provided, and that each system have the capability to control the rate of reactivity changes resulting from planned, normal power changes. One of the systems must be capable of reliably controlling anticipated operational occurrences. In addition, one of the systems must be capable of holding the reactor core subcritical under cold conditions.

(7) GDC 27 requires that the reactivity control systems have a combined capability, in conjunction with

poison addition by the emergency core cooling system, of reliably controlling reactivity changes under postulated accident conditions, with appropriate margin for stuck rods.

(8) GDC 28 requires that the effects of postulated reactivity accidents neither result in damage to the reactor coolant pressure boundary greater than limited local yielding, nor cause sufficient damage to impair significantly the capability to cool the core.

In SSAR Section 4.3, GE describes how the ABWR meets GDC 10, 11, 12, 20, 25, 26, 27, and 28 and the other requirements of SRP Section 4.3 by direct reference to the fuel licensing acceptance criteria in SSAR Appendix 4B. Appendix 4B contains proposed criteria on fuel design and on related neutronic and thermal-hydraulic aspects of the fuel design. As discussed in Section 4.2 of this report, the staff approved these criteria in its review of the ABWR fuel system design.

As discussed in Section 4.2 of this report, GE used a specific core design for the first cycle and for the ABWR system response analysis and provided specific designs of first core fuel assembly enrichment patterns based on the core design. GE has also developed, with staff approval, a slightly revised fuel assembly enrichment pattern with neutronic parameters falling within the bounds of the reference design but with decreased local power peaking. This is to be used for the first cycle as required by the staff review of the fuel misorientation event, as discussed in Section 15.3(2) of this report. In SSAR Appendix 4A, GE describes use of an example control rod pattern throughout a cycle and the resulting power distributions for the cycle. The core operating limits on process variables determined from the reference core safety evaluation are incorporated in the core associated technical specifications (TS). SSAR Chapters 6 and 15 provide the analyses to satisfy the acceptance criteria for all design-basis transients and accidents initiated from worst-case steady-state operating conditions within TS operating limits for the design life of the core.

The core design and control rod pattern operations are generally similar to current BWR designs. They only differ in relatively small ways, in details of the geometry and operating limits, which result in small differences in neutronic parameters and characteristics. These differences are generally in a direction of a more conservative core neutronic design and operation than, for example, a BWR/6. They are designed with slightly lower average power densities. The ABWR control rod pitch will be slightly greater than current designs and this will result in a six percent larger core water to UO_2 volume ratio (a moderation ratio increase), which in turn will result in a smaller (absolute magnitude) void reactivity coefficient throughout the range of power operating conditions. Both the lower average power density and smaller void reactivity coefficient tend to improve core thermal hydraulic stability and pressurization transient response. The moderation ratio increase tends to make the end-of-cycle low power (low temperature) moderator temperature reactivity coefficient less negative and possibly slightly positive. But the increase is not sufficient to cause an operational problem during startup or shutdown or a reactivity insertion problem, even assuming the maximum potential integrated positive reactivity is available for rapid insertion into the core.

In the operations examples, the ABWR control rods are withdrawn in patterns that are generally similar to current reactor withdrawal patterns using the banked position withdrawal sequence (BPWS). The primary difference is that, throughout the withdrawal patterns, multiple rods will be withdrawn simultaneously. Rather than pulling rods individually to provide the step banked patterns of BPWS, the rods in a group will be withdrawn simultaneously by the electric-drive motor systems. Similar to current BPWS patterns, about 1/8 of the 205 control rods in each of the 4 rod groups will be withdrawn to a 50-percent rod density checkerboard pattern (to about hot critical conditions). Because of the rod distribution in the patterns, the total group reactivity worths will not be very large (about 2 percent delta-K each for groups 2, 3, and 4 withdrawal when criticality might be expected). Operational control with this reactivity magnitude, and its accompanying differential reactivity worth, will be straightforward and not significantly different from current operation. Groups 3 and 4 (covering cold to hot critical) will be operated in a jog mode to avoid any approach to period scram levels. Beyond 50-percent rod density, the groups will be divided into groups of four or eight rods that will moved simultaneously, with patterns similar to BPWS and operation similar to BWR/6s. The simultaneous withdrawal for the first 50-percent groups will reduce the maximum reactivity worth, which could be associated with a (postulated) rod drop accident to insignificant levels. In the power range, the ABWR examples use a "control cell core" strategy, which, combined with axial zoning of fuel enrichment and burnable poison, will result in very little movement of the control rods over most of the cycle in the normal power operation range. These various control rod operational characteristics are either very similar to current approved operation or are improvements. The operation and characteristics are acceptable.

Design criteria for the core neutronics for the ABWR are included in the fuel design criteria in SSAR Appendix 4B. The nuclear criteria are 4B.4, 1 through 8. These criteria provide for fuel and moderator reactivity coefficients, shutdown margin, and fuel storage. reactivity. These criteria are in GESTAR Amendment 22 and are discussed in Section 4.2 of this report in connection with the fuel design. They are acceptable.

The staff concludes that on the basis of (1) meeting the relevant fuel licensing acceptance criteria of SSAR Appendix 4B, (2) the general similarity to current operating GE BWR cores, and (3) the additional conservatism relative to current BWRs in the area of power density and less negative void coefficient, the ABWR nuclear design meets the requirements of SRP Section 4.3 and is acceptable.

4.4 Thermal Hydraulic Design

The NRC staff reviewed the thermal-hydraulic design of the ABWR reactor core in accordance with SRP Section 4.4. SSAR Section 4.4 describes how the ABWR meets GDC 10 and the other requirements of SRP Section 4.4 by direct reference to the fuel licensing acceptance criteria in SSAR Appendix 4B. As discussed in Section 4.2 of this report, SSAR Appendix 4B contains criteria on fuel design and related neutronic and thermal-hydraulic aspects of the fuel design. The staff approved these criteria after its review of the ABWR fuel system design.

GE provided a specific first cycle core design for the ABWR and used this design for system response analysis. As discussed in Section 4.2 of this report, the fuel used in this design is the same as currently approved fuel used in existing BWRs and the core fuel arrangement is similar to current fuel loadings. Alternate fuel loading designs for the first or subsequent cycles will have to conform to the fuel criteria of SSAR Appendix 4B, or be specifically reviewed and approved by the staff; for the first cycle any changes will have to be reviewed and approved by the NRC, along with any changes necessary for the analyses of SSAR Chapters 6 and 15. These criteria include the requirements for providing new thermal-hydraulic data such as critical power ratio correlations and limits on fuel stability characteristics.

The reference core design is generally similar to current BWR designs. It only differs in relatively small ways in details of the geometry and operating limits, and which result in small differences in neutronic and thermal-hydraulic parameters and characteristics. These differences generally reflect a more conservative core neutronic and thermal-hydraulic design and operation than, for example, a BWR/6. The core has a with slightly lower average power density. The fuel bundle and overall core parameters for this design fall within the range of applicability of existing critical power correlations.

The only significant new design feature of the recirculation flow system is the use of the reactor internal pumps (RIPs). This system is discussed in Section 5.4.1 of this report. This change from current GE BWR design does not produce a significantly different power/flow operation map for the ABWR than that produced by current GE BWRs, although map parameters are slightly changed in some areas. GE provided a power/flow map, corresponding to the core design (discussed in SER Sections 4.2 and 4.3 of this report), to be used in systems response analyses. Each COL applicant should provide a plant-specific power/flow map at the time of application and for core reloads. This was DFSER COL Action Item 4.4-1. GE has also included this action item in the SSAR and the staff finds it to be acceptable. Flow lines and control rod lines of the power/flow map generally will be within the range of operation currently permitted for BWR/6s, although natural circulation occurs at a slightly lower flow. The ABWR is expected to operate with a minimum pump speed of 30 percent of nominal full-flow speed, providing a minimum (all pumps operating) flow of about 40 percent. The ABWR design replaces the usual cavitation restriction region with a steam separation limit region to provide for acceptable moisture carryover. There is an interlock to reduce RIP speed to prevent operation in this restricted region. The maximum expected flow for the RIP system is about 115 percent of normal full flow. For the submitted design power/flow map, GE did not consider operating above the 102-percent power control rod line. These power/flow parameters generally are within expected bounds and present an acceptable region for normal operation. As indicated in SSAR Section 4.4.3.2, for normal operation at least 9 of the 10 RIPs are required to be operating. GE has provided a power/flow map for both 9 and 10 RIP operation.

In addition to the above boundaries of the power/flow map, there is a restricted region in the low-flow/high-power areas of the map that is intended to eliminate possible problems with thermal-hydraulic stability. (This is Region III on the map, generally considered to be approximately above the 80-percent control rod line and below the 40-percent flow line.) Operation will not be permitted in this region. Automatic startup logic will be programmed to block rod withdrawal in this region during low power and startup operation and to insert rods to withdraw from the region if entered inadvertently from higher power/flow regions.

In addition to this automatic control, the ABWR has several design features intended to improve the stability status of the reactor that are not present in current reactors. These features are (1) smaller inlet orifices (loss coefficient doubled) to increase single-phase pressure drop, (2) wider control rod pitch to increase flow area and increase the moderator/fuel ratio and reduce the void reactivity coefficient, (3) more steam separators to reduce the two-phase pressure drop, (4) multiple RIPs on multiple power supplies along with minimum pump speed logic to reduce the likelihood of significant flow loss, and (5) regional local power range monitor (LPRM) and average power range monitor neutron flux time histories available for operator display to detect oscillations.

The control rod insertion to withdraw from Region III, if entered, will be provided by the selected control rod run-in system. This system is described in SSAR Section 7.7.1.2. Multiple control rods will be automatically inserted (simultaneously) to move to a stable region (e.g., from the 100-percent control rod line to below 25-percent power if two or more RIPs trip and flow is below the trip set point). Set points for power and flow will be adjustable and may be changed if stability analyses require it.

The NRC, in parallel with work by the BWR Owners' Group (BWROG) stability sub-committee, has for some time been reviewing generic questions relating to BWR stability. This review is still in progress. Questions concerning the need and the methods for improved stability control so as not to exceed core thermal-hydraulic safety limits and to understand and control, if necessary, adverse affects of power oscillation during anticipated transient without scram (ATWS) events are essentially complete. The ABWR design features indicated above are desirable for improving inherent stability and the system for control rod blocking and insertion to provide exclusion from Region III are consistent with some of the proposed long term solutions (LTS) for current BWRs. The BWROG has developed several (LTS that have been approved by the NRC for application to operating reactors. The region exclusion system for the ABWR is similar to the BWROG LTS Option IA. In response to a staff question requesting review of the recent LTS work and the possibility of its application to the ABWR, GE has stated that in addition to the region exclusion system, the BWROG Option III system, the oscillation power range monitor, which is based on the detection of oscillation signals by the LPRMs, will be implemented in the ABWR design (see SSAR Section 7.6.1.1.2.2). This LTS methodology has been accepted by the staff (letter from A. Thadani, NRC, to L. England, BWROG, "Acceptance for Referencing of Topical Reports NEDO-31960 and NEDO-31960 Supplement 1, 'BWR Owners Group Long-Term Stability Solution Licensing Methodology'") and is considered to be acceptable for the ABWR.

In the DFSER, the staff stated that the information presented by GE and the staff review of the subject of instability during ATWS transients was not complete. This was DFSER Open Item 4.4-1. GE subsequently submitted a report describing their analyses of oscillations during limiting ATWS transients susceptible to thermal-hydraulic instability and possibly large oscillations (Ref: GE ATWS Stability study, February 19, 1993, GE transmitted by letter dated February 22, 1993). GE performed calculations using the TRACG code, which has been used for stability studies for operating BWRs and has been audited by the staff. The ABWR calculations were reviewed by the staff and its consultants, and the stability characteristics of the ABWR have been explored using the Oak Ridge National Laboratory LAPUR code, commonly used by the staff for stability analyses. The LAPUR calculations showed, in accord with GE findings, that the ABWR is more stable than most current BWRs and will have smaller power oscillations, if any, at comparable operating conditions. The GE TRACG calculations appear to bound the worst expected ATWS conditions for instability and do not result in exceeding ATWS fuel failure limit criteria. The ABWR mitigating actions are similar to the BWROG proposed actions for current BWRs, and in SSAR Section 4.4.3.7 states that the ABWR emergency procedure guidelines will incorporate any changes recommended by the BWROG committee on thermal-hydraulic stability. There is automatic feedwater reduction and boron injection via the standby liquid control system. The feedwater runback and lowering of reactor level below the feedwater spargers are very effective in reducing oscillations magnitudes. The TRACG modelling and input for these calculations were found to be acceptable. The staff concludes that the issue of instability induced large power oscillations during ATWS has been properly addressed for the ABWR. GE has also included this information in the SSAR and the staff find it to be acceptable. On the basis of this evaluation, this item is resolved.

The ABWR design, as initially presented, did not include a loose-parts monitoring system (LPMS). However, in response to the staff position that an LPMS is required, GE submitted an LPMS general description, including a design basis, system description, system operation, safety evaluation, test, and inspection and application. This system is designed in conformance with Regulatory Guide (RG) 1.133, "Loose-Parts Detection Program for the Primary System of Light-Water-Cooled Reactors," Revision 1. The system includes sensors (accelerometers located at natural loose parts collection regions, e.g., steam outlet nozzle, feedwater inlet nozzle, control drive housings), signal conditioning, signal analysis, alarms, and calibration. The sensitivity is such that a sensor will be able to detect a metallic part between 0.1-14 kg (0.25 to

30 lbs) with a kinetic energy of 0.7 joule (0.5 ft-lb) on an inside surface within 1 m (3 ft) of a sensor. There will be provisions for online channel checks and functional tests and offline calibration. The system is designed to meet the seismic and environmental operability recommendations of RG 1.133, Revision 1. GE has provided an ITAAC for the LPMS as part of its Tier 1 certified design material submittal. It provides for design commitment for detector locations and sensitivity with appropriate accompanying inspections, tests, and criteria. In the DFSER, the staff stated that, as DFSER Open Item 4.4-2, Certified Design Commitment 1 in the LPMS ITAAC should be expanded to explicitly state that the LPMS design is consistent with the requirements of RG 1.133, Revision 1. This has been done; therefore, DFSER Open Item 4.4-2 is resolved. GE has submitted the Design Description, the ITAAC for the LPMS. The adequacy at acceptability of the ABWR Design Description and ITAAC are evaluated in Section 14.3 of this report. On the basis of this evaluation, this open item is resolved.

Core flow patterns are expected to be uniform at the core inlet during normal operations as a result of flow distributions from the downcomer through the RIPs, into the lower plenum, up through the orifices of the lower core plate, and into the fuel assemblies. TS will require for normal operation that at least 9 of the 10 RIPs are operating. Operation with fewer than 9 RIPs operating will require supporting analyses and justification by the COL applicant. This was DFSER TS Item 4.4-1. GE asserts that, with the allowed number of RIPs inoperable, pump operation will be close to normal and bounded by one recirculation loop operation in current jet pump BWRs (with, in effect, half the pumps out) for which there are no restrictions other than similar-type LOCA power-density restrictions. The restriction to no fewer than 9 RIPs operating is a recent modification of SSAR Section 4.4 and of the TS, and therefore, DFSER TS Item 4.4-1 is resolved. The restriction and specification are acceptable. The staff requested GE to provide existing flow test information. This was DFSER Confirmatory Item 4.4-1. GE has provided references in SSAR Section 4.4 to such information on current reactors relevant to the above restricted modes of operation. GE has also included this information in the SSAR and the staff finds it to be acceptable. Therefore, DFSER Confirmatory Item 4.4-1 is resolved.

4.5 Reactor Materials

4.5.1 Control Rod Drive System Structural Materials

The acceptance criteria used as the bases for the staff's evaluation of control rod drive (CRD) structural materials are SRP Section 4.5.1. The CRD structural materials are acceptable if they meet the relevant requirements of: GDC 1 as it relates to structures, systems, and components important to safety being designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed; GDC 14 as it relates to the reactor pressure boundary being 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 26 as it relates to the control rod being capable of reliably controlling reactivity changes so that specified acceptance fuel design limits are not exceeded; and Section 50.55a, of Title 10 of the Code of Federal Regulations, Part 50 as it relates to structures, systems, and components shall be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety functions to be performed.

The SSAR states that the properties of the materials selected for the ABWR CRD mechanism will be equivalent to those given in Appendix I to Section III of the American Society of Mechanical Engineers (ASME) Code; Parts A, B, and C of Section II of the ASME Code; and RG 1.85. No cold-worked austenitic stainless steels except those with controlled hardness or strain are employed in the CRD system. All materials used in this system will be selected for their compatibility with the reactor coolant as described in Articles NB-2160 and NB-3120 of the ASME Code. The materials selected as identified in SSAR Section 4.5.1 will be resistant to stress corrosion in a BWR environment. The controls imposed on the austenitic stainless steel of the CRD mechanism conform to the recommendations of RGs 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal," (Revision 3) and 1.44, "Control of the Use of Sensitized Stainless Steel," (Revision 0).

All materials selected for application in CRD mechanism components will conform with the ASME Code Section III or RG 1.85. Fabrication and heat treatment practices performed in accordance with the Code and regulatory guide provide added assurance that stress corrosion cracking will not occur during the design life of the components. Both martensitic and precipitation-hardening stainless steels will be given tempering or aging treatments in accordance with staff positions. Cleaning and cleanliness control will be in accordance with American National Standards Institute Standard N 45.2.1-1973, "Cleaning of Fluid Systems and Associated Components During Construction Phase of Nuclear Power Plants," and RG 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants," Revision 0.

The information in the SSAR meets the criteria in SRP Section 4.5.1. The staff concludes that the structural materials for the CRD mechanism conform to the staff's regulatory guidance to ensure that the requirements of GDC 1, 14, and 26 of Appendix A to 10 CFR Part 50 and the requirements of 10 CFR 50.55a are satisfied. This is acceptable.

4.5.2 Reactor Internal Materials

The acceptance criteria used as the bases for the staff's evaluation of reactor internal materials are SRP Section 4.5.2. The reactor internals are acceptable if the design, fabrication, and testing of the materials used in the reactor internal and core support structures meet the code and standards commensurate with the safety function to be performed so that the relevant requirements of GDC 1 and Section 50.55a of Title 10 of the Code of Federal Regulations, Part 50 are met.

The SSAR states that the requirements of GDC 1 and 10 CFR 50.55a will be met with regard to ensuring that the design, fabrication, and testing of the materials used in the reactor internal and core support structures are of high quality standards and adequate for structural integrity. The controls imposed on components constructed of austenitic stainless steel will satisfy the recommendations of RGs 1.31 and 1.44.

The materials to be used for the construction of components of the reactor internals and core support structures were identified in SSAR Section 4.5.2 by specification and found to be in conformance with the requirements of NG-2000 of Section III and Parts A, B, and C of Section II of the ASME Code. Extensive tests and satisfactory performance have shown that the specified materials are compatible with the BWR environment. The controls imposed on the reactor coolant chemistry satisfy the chemistry limits specified in EPRI report NP-4947, "BWR Hydrogen Water Chemistry Guidelines 1987 Revision," December 1988. This will provide reasonable assurance that the reactor internal and the core support structures will be adequately protected during operation from conditions that could lead to stress corrosion, including irradiation-assisted stress corrosion cracking and loss of component structural integrity.

The materials selection, fabrication practices, examination and testing procedures, and control practices provide reasonable assurance that the materials used for the reactor internal and core support structures will be maintained in a metallurgical condition that will preclude inservice deterioration. Conformance with the requirements of the ASME Code constitutes an acceptable basis for meeting, in part, the requirements of GDC 1 and 10 CFR 50.55a.

The staff finds that the information in the SSAR related to reactor internal materials meets the criteria of SRP Section 4.5.2 and is, therefore, acceptable.

4.6 Functional Design of Fine Motion Control Rod Drive System

The staff reviewed the fine motion control rod drive (FMCRD) system in accordance with SRP Section 4.6. The staff performed an audit review of each of the areas listed in the "Areas of Review" portion of the SRP section according to the guidelines provided in the "Review Procedures" section of the SRP section. Conformance with the acceptance criteria formed the basis for the staff's evaluation of the CRD system with respect to the applicable regulations of 10 CFR Part 50.

The ABWR incorporates electric-hydraulic FMCRDs, which provide electric fine rod motion during normal operation and hydraulic pressure for scram insertion. Fine motion during normal operation is provided by a ball nut and spindle arrangement driven by the electric stepper motor. In response to a scram signal, the control rods are inserted hydraulically via the stored energy in the scram accumulator similar to the current operating BWR CRDs.

A scram signal is also given simultaneously to insert the FMCRDs 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 FMCRD and recirculation flow control system (RFCS) are designed to control reactivity during power operation. Reactivity will be controlled in the event of fast transients by automatic rod insertion. During ATWS conditions, the internal recirculation pumps will be tripped automatically. In the event the reactor cannot be shut down with the control rods, the operator can actuate the standby liquid control system (if not automatically started) that pumps a solution of sodium pentaborate into the primary system. The evaluation of the functional design of the standby liquid control system is addressed in Section 9.3.5 of this report. This evaluation resolved Open Item 15 from the DSER (SECY-91-153). Compliance with the ATWS rule is discussed in Section 15.5 of this report.

Reactivity in the core will be controlled by the FMCRD by moving control rods interspersed throughout the core. These rods will control the reactor's overall power level and will provide the principal means of quickly and safely shutting down the reactor.

GE submitted a proprietary failure modes and effects analysis for the FMCRD system. The single-failure analysis of the FMCRD and hydraulic control unit (HCU) components indicates that the system design is satisfactory. This resolved Open Item 17 from the DSER (SECY-91-153). A supply pump (with a spare pump on standby) will provide the HCUs with water from the condensate treatment system and/or condensate storage tank to supply CRD purge water and to supply the purge water to the RIPs and reactor water cleanup pumps. The supply pump also will provide water to a scram accumulator in each HCU to maintain the desired water inventory. When necessary, the accumulator will force water into the drive system to scram the control rods connected to that HCU; the volume of water in the scram accumulator will be sufficient to scram two rods. A single failure in an HCU would result in the failure of two rods only. The failed rods would not be adjacent; they would be sufficiently separated so that the reactivity effect would essentially be the same as for the failure of one rod in current BWRs. Therefore, adequate shutdown margin exists with a single HCU failure even though the HCU is shared by two drives.

The FMCRD is designed to permit periodic functional testing during power operation with the capability to independently test individual scram channels and motion of individual control rods. The FMCRD is also designed so that failure of all electrical power or instrument air will cause the control rods to scram, thereby protecting the reactor. This satisfies the protection system failure mode requirements of GDC 23.

Preoperational tests of the CRD hydraulic system will be conducted to verify the capability of the system. Startup tests will be conducted over the range of temperatures and pressures from shutdown to operating conditions to determine compliance with applicable TS. Each rod that is partially or fully withdrawn during operation will be exercised one notch at least once each week. After each refueling shutdown, control rods will be tested for compliance with scram time criteria from the fully withdrawn position.

The FMCRD is designed to control reactivity under normal operating conditions and during anticipated operational occurrences. This capability is demonstrated by the safety analyses discussed in SSAR Chapter 15 (including effects of stuck rods). This CRD system also will be capable of holding the core subcritical under cold shutdown conditions. The RFCS will be capable of accommodating reactivity changes during normal operating conditions. The standby liquid control system will be capable of bringing the reactor subcritical under cold shutdown conditions in the event the control rods cannot be inserted.

These protection and reactivity control systems, taken together, satisfy the requirements of GDC 26, 27, and 29 pertaining to reactivity control system redundancy and capability, combined reactivity control system capability, and protections against anticipated operational occurrences.

The control rod system design incorporates appropriate limits on the potential amount and rate of reactivity increase. Control rod withdrawal sequences and patterns will be selected to achieve optimum core performance and low individual rod worths. The rod control and information system (RCIS) will reduce the chances of withdrawal other than by the preselected rod withdrawal pattern. The RCIS function will assist the operator with an effective backup control rod monitoring routine that enforces adherence to established control rod procedures for startup, shutdown, and low-power-level operations.

A malfunction in the FMCRD could result in a reactivity change. GE demonstrated in SSAR Chapter 15 that the FMCRD system will limit these postulated transients to within acceptable fuel design limits, as required by GDC 25.

The control rod mechanical design incorporates a brake system and ball check valve, which will reduces the chances of rapid rod ejection. This engineered safeguard will protect against a high reactivity insertion rate from a potential control rod ejection. Normal rod movement and the rod withdrawal rate will be limited through the FMCRD.

GE adopted an internal CRD housing support to replace the support structure of beams, hanger rods, grids, and support bars used in current BWR designs. This system will use the outer tube of the drive to provide support. This tube will be welded to the drive middle flange and will attach by a bayonet lock to the guide tube base. The guide tube, supported by the housing extension, will prevent downward movement of the drive in the event of housing failure. The CRD housing support is designed to prevent ejection of a CRD and attached control rod.

The FMCRD is designed to detect separation of the control rod from the drive mechanism. Two redundant and separate Class 1E switches will be 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 will cause an immediate rod block and will initiate an alarm in the control room, thereby reducing the chances of a rod drop accident. The ABWR control rod design does not include a velocity limiter because of the design features described. The design features of the reactivity control system will limit the potential amount and rate of reactivity increase to ensure that GDC 28 is satisfied for postulated reactivity accidents.

The safety concerns associated with a pipe break, given in NUREG-0803, "Safety Concerns Associated With a Pipe Break in the BWR Scram System," are not applicable for the ABWR. The ABWR design does not include scram discharge volume piping. The water displaced by the CRD during the scram will be routed to the reactor pressure vessel.

Nitrogen charge pressure for the CRD accumulator is substantially increased in the ABWR design. The initial N_2 charge pressure will be 1,4817 kPa (2,134 psig) compared with a maximum of 10,446 kPa (1,500 psig) for the operating BWRs. GE's calculations have verified that even with initial N_2 charge pressure of 12,859 kPa (1,850 psig), there is sufficient differential pressure to drive the CRD fully with a reactor pressure of 9,584 kPa (1,375 psig). (The peak calculated reactor pressure for a main steam isolation valve closure event during ATWS is less than 9,584 kPa (1,375 psig)). The calculations have shown that at 4.53 seconds, when the CRD is fully inserted, the accumulator N_2 charge pressure is calculated as 10,625 kPa (1,526 psig) with reactor pressure at 9,584 kPa (1,375 psig). There is a remaining differential pressure of 166 kPa (151 psid), demonstrating that accumulator pressure is more than adequate for the rod insertion.

GE submitted the LaSalle test report on FMCRD in-plant test program by letter dated October 12, 1989. This report is proprietary. The in-plant test shows that the basic design of the FMCRD is acceptable. There is significant operating experience with the FMCRDs. In Europe, 2,700 drives are in service with over 15,000 drive years of experience. Thus, FMCRDs in BWRs, are a proven technology. Open Item 16 in the DSER (SECY-91-153) was resolved in the DFSER.

GE submitted the design description and the ITAAC for the FMCRD system. This was DFSER Open Item 4.6-1. 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 open item is resolved.

The staff concludes that the functional design of the reactivity control system conforms to the requirements of GDC 23, 25, 26, 27, 28, and 29 with regard to demonstrating the ability to reliably control reactivity changes under normal operation, anticipated operational occurrences, and accident conditions, including single failures. The design of the reactivity control system conforms to the applicable acceptance criteria of SRP Section 4.6, and is acceptable.

5 REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

5.1 Introduction

The reactor coolant systems (RCS) and connected subsystems are evaluated in the following sections.

5.2 Integrity of Reactor Coolant Pressure Boundary

5.2.1 Compliance With Code and Code Cases

The staff reviewed the measures used to provide and maintain the integrity of the reactor coolant pressure boundary (RCPB) and other pressure-retaining components and their supports that are important to safety for the design lifetime of the plant.

5.2.1.1 Compliance With 10 CFR 50.55a

According to 10 CFR 50.55a, components important to safety are subject to the following:

(1) RCPB components must meet the requirements for Class 1 (Quality Group (QG) A) components specified in ASME Code, Section III, except for those components that meet the exclusion requirements of 10 CFR 50.55a(c)(2).

(2) Components classified as QG B and C must meet the requirements for Class 2 and 3 components, respectively, specified in ASME Code, Section III.

SSAR Table 3.2-1 classifies the pressure-retaining components of the RCPB as ASME Code, Section III, Class 1 components. These Class 1 components are designated QG A in conformance with Regulatory Guide (RG) 1.26, "Quality Groups Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," Revision 3. The staff reviewed the QG A RCPB components in accordance with SRP Section 5.2.1.1, as discussed below.

In addition to the QG A components of the RCPB, certain lines that will perform a safety function and that meet the exclusion requirements of 10 CFR 50.55a(c)(2) are classified as QG B in accordance with Position C.1 of RG 1.26, Revision 3, and will be constructed as ASME Code, Section III, Class 2 components. The staff's review of these components and other pressure-retaining components that will be constructed to ASME Code, Section III, Class 2 and Class 3 specifications, is discussed in Section 3.2.2 of this report.

SRP Section 5.2.1.1 recommends that safety analysis reports for both construction permits and operating licenses contain a table identifying the ASME component code, code edition, and applicable addenda for all ASME Code, Section III, Class 1 and 2 pressure vessel components, piping, pumps, and valves in the RCPB. SSAR Section 5.2.1.1 states that the ASME Code edition, applicable addenda, and component dates will be in accordance with 10 CFR 50.55a and that ASME Code, Section III, will be used for the design of ASME Code Class 1, 2, and 3 pressure retaining components and their supports. The specific edition and addenda are given in SSAR Tables 1.8-21 and 3.2-3. The ASME Code is considered Tier 1 information; however, the specific edition and addenda are considered Tier 2 information partly because of the continually evolving design and construction practices (including inspection and examination techniques) of the Code. Fixing a specific edition and addenda during the design certification stage might result in inconsistencies between design and construction practices during the detailed design and construction stages. The ASME Code involves a consensus process to reflect the evolving design and construction practices of the industry. Although reference to a specific edition of the Code for the design of ASME Code Class components and their supports is suitable to reach a safety finding during the design certification stage (as reflected in the discussion of the reactor pressure vessel (RPV) design in Section 5.2.4 of this report), it is necessary that the construction practices and examination methods of an updated Code that would be effective at the COL stage be consistent with the design practices established at the design certification stage.

To avoid this potential inconsistency for the advanced boiling water reactor (ABWR) pressure-retaining components and their supports, it is appropriate that the ASME Code be specified as Tier 1 information and the specific edition and addenda as Tier 2 information so that the COL applicant has the option to revise or supplement the referenced Code edition with portions of the later Code editions and addenda and still ensure consistency between the design and construction practices. In this manner, consistency with the latest design, construction, and examination practices also is ensured. However, the staff finds that there might be a need to fix certain design parameters from a specific Code edition or addenda during its design certification review particularly when that information is important for establishing a significant aspect of the design or is used by the staff to reach its final safety determination. Such considerations, if necessary, are reflected in the various sections of this report.

Therefore, all ASME Code, Class 1, 2, and 3 pressure-retaining components and their supports shall be designed in accordance with the requirements of ASME Code, Section III, using the specific edition and addenda given in the SSAR. The COL applicant should ensure that the design is consistent with the construction practices (including inspection and examination methods) of the ASME Code edition and addenda in effect at the time of the COL application, as endorsed in 10 CFR 50.55a. The COL applicant should identify in its application the portions of the later code editions and addenda for NRC staff review and approval. This was DFSER COL Action Item 14.1.3.3.2.1-1. The resolution of this item is discussed in Section 3.12.2.1 of this report.

Since the above position is not totally consistent with ASME Code, Section III, Subsection NCA-1140, "Use of Code Editions, Addenda, and Cases," the following comparison is provided:

NCA-1140 Rules	Design Certification Position
NCA-1140(a)(1) - Under the rules of this Section, the owner or his designee shall establish the Code edition and addenda to be included in the Design Specifications. All items of a nuclear power plant may be constructed to a single Code edition and addenda, or each item may be constructed to individually specified Code editions and addenda.	(1) The vendor must specify the Code edition and addenda in the SSAR during design certification. The COL applicant may update the Code edition or addenda (or portions thereof) referenced in the SSAR using a process similar to that specified in 10 CFR 50.59 without prior NRC approval unless the proposed change involves a change to the certified design or an unreviewed safety question. The COL applicant may update the referenced edition or addenda for all items of a nuclear power plant or a specific item if the construction practices (including fabrication, inspection, and examination methods) are not compatible with the design requirements. All changes to the referenced edition and addenda must be documented and maintained by the COL applicant and must be available for audit.
NCA-1140(a)(2) - In no case shall the Code edition and addenda dates established in the Design Specifications be earlier than 3 years prior to the date that the nuclear power plant construction permit application is docketed.	(2) The specific Code edition and addenda are required to be established during design certification except for Section XI requirements related to inservice inspection (ISI), inservice testing, and system pressure tests, which must meet the requirements of 10 CFR 50.55a(f) and (g). Further discussions and any exceptions to 10 CFR 50.55a(f) and (g) are noted in Sections 5.2.4 and 6.6 of this report.
NCA-1140(b) - Code editions and addenda later than those established by (a) above may be used by mutual consent of the owner or his designee and certificate holder. For Division 2 design and construction, the consent of the designer shall also be obtained. Specific provisions within an edition or addenda later than those established in the design specifications may be used, provided that all related requirements are met.	(3) This is acceptable subject to the conditions noted in Position (1) above.
NCA-1140(c) - Code Cases are permissible and may be used beginning with the date of approval by the ASME Council (and the American Concrete Institute for Division 2 design and construction). Only Code Cases that are specifically identified as being applicable to this Section may be used.	(4) As discussed in Section 5.2.1.2 of this report, only those Code Cases identified in RG 1.84, Revision 24, or 1.85, Revision 24, as specified in the SSAR may be used. The COL applicant may submit for staff review and approval future Code Cases that are endorsed in future revisions of RGs 1.84 and 1.85 with its COL application, provided these cases do not involve a change to the certified design or an unreviewed safety question.

NCA-1140 Rules	Design Certification Position
NCA-1140(d) - Code Cases may be used by mutual consent of the owner or his designee, and the certificate holder on or after the date permitted by (c) above. For Division 2 design and construction, the consent of the designer shall also be obtained.	(5) Code Cases to be used in the design of the standard plant must be identified by the vendor in the SSAR during design certification.
NCA-1140(e) - Existing materials previously produced and certified in accordance with Code editions and addenda earlier than the one specified for construction of an item may be used, provided all of the following requirements are satisfied. (i) The material (NCA-1220) meets the applicable requirements of a material specification permitted by paragraph 2121 of the applicable subsection of the Section III edition and addenda specified for construction. (ii) The material meets all the requirements of Article 2000 of the applicable Subsection of the Section III edition and addenda specified for construction. (iii) The material was produced under the provisions of a quality system program that had been accepted by the society or qualified by a party other than the society (NCA-3820), in accordance with the requirements of the latest Section III edition and addenda issued at the time the material was produced. Material exempted from portions of the provisions of NCA-3800 by paragraph 2610 of the applicable Subsection of Section III may be used, provided the requirements of (i) and (ii) above are met.	(6) Does not apply to design certification.
NCA-1140(f) - Code editions, addenda (including the use of specific provisions of editions addenda permitted by (b) and (e) above), and Cases used shall be reviewed by the owner or his designee for acceptability to the regulatory and enforcement authorities having jurisdiction at the nuclear power plant site.	(7) The vendor must specify the Code edition and addenda in the SSAR during design certification. Use of later editions and addenda (or portions thereof) and Code Cases not approved during design certification must be reviewed by the COL applicant or its designee for acceptability to the regulatory and enforcement authorities having jurisdiction at the nuclear power plant site.

5.2.1.1.1 Conclusion

The staff concludes that the construction of all ASME Code, Class 1, 2, and 3 components and their supports will conform to the appropriate ASME Code editions and addenda and the Commission's regulations and that component quality will be commensurate with the importance of the safety function of all such components and their supports. This constitutes an acceptable basis for satisfying GDC 1 and is acceptable.

5.2.1.2 Applicable Code Cases

SSAR Table 5.2-1 identifies specific ASME Code Cases that will be applied in the construction of pressure-retaining ASME Code, Section III, Class 1, 2, and 3 components. The staff's review of this table is based on the guidelines in RG 1.84, "Design and Fabrication Code Case Acceptability - ASME Section III, Division 1," and RG 1.85, "Materials Code Case Acceptability - ASME Section III, Division 1." All ASME Code Cases that have been either conditionally or unconditionally endorsed by the staff are discussed in one of these RGs, as applicable. Table 5.2-1 of the SSAR lists 15 code cases that will be used in the design of the ABWR. All of these code cases have been endorsed by the staff and are included in one of the above RGs.

In SSAR Sections 3.9.3.4 and 3.9.3.5, Code Case N-476, "Class 1, 2, 3, and MC Linear Component Supports - Design Criteria for Single Angle Members, Section III, Division 1, Subsection NF," is referenced as augmenting ASME Subsection NF rules for the design of component supports. As stated in Section 3.9.3.3 of this report, the staff finds this code case acceptable. Therefore, it will be an acceptable addition to SSAR Table 5.2-1.

The only acceptable ASME Code Cases that may be used for the design of ASME Code, Class 1, 2, and 3 piping systems in the ABWR standard plant are those either conditionally or unconditionally approved in RGs 1.84 and 1.85 in effect at the time of design certification. However, the COL applicant may submit, with its COL application, future code cases that are endorsed in RGs 1.84 and 1.85 at the time of the application provided they do not alter the staff's safety findings on the ABWR certified design. In addition, the COL applicant should submit those Code Cases that are applicable to RG 1.147, "Inservice Inspection Code Case Acceptability - ASME Section XI, Division 1," which is in effect at the time of the COL application.

The staff concludes that all of the Code Cases in SSAR Table 5.2-1 either meet the guidelines of RG 1.84 or 1.85 or have been reviewed and endorsed by the staff and are acceptable for use on the ABWR design. Compliance with the requirements of these Code Cases will result in a component quality that is commensurate with the importance of the safety functions of these components, constitutes the basis for satisfying GDC 1, and is acceptable.

5.2.2 Overpressure Protection

The staff evaluated overpressure protection in the ABWR in accordance with SRP Section 5.2.2, which states that the acceptance criteria also are based on GDC 31 as it relates to the fracture behavior of the RCPB. This review area is addressed in Section 5.3.1 of this report. SRP Section 5.2.2 also states that overpressure protection during low-temperature operation need not be considered for BWRs, since BWRs never operate in water solid conditions. Hence, overpressure protection during low-temperature conditions is not addressed for the ABWR.

The RCPB is designed with a pressure relief system to:

- prevent the pressure in the RCPB from rising beyond 110 percent of the design value
- provide automatic depressurization if small breaks in the nuclear system should occur together with failure of the high-pressure core flooder (HPCF) and reactor core isolation cooling (RCIC) system. (This depressurization will allow operation of the low-pressure flooder systems to protect the fuel barrier.)

To be acceptable, the pressure relief system must permit verification of its operability and must withstand adverse combinations of loadings and forces resulting from normal, upset, emergency, and faulted conditions.

Overpressure protection in the ABWR will be provided using 18 safety/relief valves (SRVs); of which 8 are part of the automatic depressurization system (ADS). The 18 SRVs are divided into six nominal pressure set point groups and mounted on the four main steamlines (MSLs) between the reactor vessel and the first isolation valve inside the drywell. The SRVs will discharge through piping to the suppression pool. The design of the ABWR pressure relief system is similar to that for BWR 4, 5, and 6 plants.

The SRVs are classified as QG A and seismic Category I, as shown in SSAR Table 3.2. The SRVs are designed to meet RGs 1.26, Revision 3, and 1.29, "Seismic Design Classification," Revision 3.

The nominal pressure set points of the SRVs will be distributed in six valve groups with a minimum set point of 7.9 MPa (1,149.2 psig) and a maximum of 8.2 MPa (1,189.0 psig) in the safety mode of operation. The nominal pressure set points of the SRVs in the relief mode of operation will be at a minimum of 7.5 MPa (1,089.5 psig) and maximum of 7.9 MPa (1,139.2 psig). The SRVs can also be operated in the relief mode by remote-manual controls from the main control room. Four SRVs can also be operated from the remote shutdown panel. The effects of flow-induced SRV discharge line backpressure on the performance of the SRV are addressed by sizing the line to ensure that the steady-state backpressure does not exceed 40 percent of the SRV inlet pressure. These sizing criteria control the effective backpressure buildup and maintain the required force balance needed to keep the SRV open and to permit proper blowdown.

In the DFSER, the staff noted that before the valves are installed, the SRV manufacturer will test them hydrostatically according to ASME Code, Section III requirements. During startup testing, opening response time and set pressure tests will be conducted to verify that design and performance requirements have been met. This was DFSER COL Action Item 5.2.2-1. Since these tests are being addressed in Chapter 14 of the SSAR, they need not be specified as a COL action item. Therefore, COL Action Item 5.2.2-1 is resolved.

GDC 15 defines the basis for overpressurization protection in a nuclear reactor. It requires that the RCPB design conditions not be exceeded during any condition of normal operation, including anticipated operational occurrences. To satisfy this criterion, the overpressurization protection system for the ABWR is designed in compliance with ASME Code, Section III, which requires that the maximum pressure reached during the most severe pressure transient be less than 110 percent of the design pressure. For the ABWR, that pressure limit is 9.5 MPa (1,375.3 psig). GE analyzed the series of transients that would be expected to require SRV actuation to prevent overpressurization. The analysis was performed using the computer-simulation model ODYNA. ODYNA is the ABWR version of ODYN incorporating changes unique to the ABWR such as reactor internal pumps (RIPs). ODYN is described in GE Topical Report NEDO-24154. The staff reviewed ODYN and found it acceptable as documented in "Safety Evaluation for the General Electric Topical Report Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors, NEDO-24154 and NEDO-24154-P Volumes I, II, III, June 1980." The staff performed an audit of ODYNA and found the changes to be acceptable. The acceptability of ODYNA for ABWR transient analysis is discussed in Section 15.1 of this report. The analyses show that the maximum pressure will remain below the 9.5-MPa (1,375.3-psig) limit. For the most severe transient (i.e., closure of all main steam isolation valves (MSIVs) with a high neutron flux scram), the maximum vessel bottom pressure is calculated to be 8.8 MPa (1274.4 psig) when all 18 SRVs are assumed to operate in the safety mode. The analysis assumed that the plant was operating at 102.7 percent of rated steam flow of 7.85 × 10E+6 kg/hr (about 17 × 10E+6 lb/hr) and a vessel dome pressure of 7.2 MPa (1,040 psig). This is consistent with SRP Section 5.2.2 and is acceptable.

GE based the sizing of the SRVs on the initiation of a reactor scram by the high-neutron flux scram, which is the second safety-grade scram signal from the reactor protection system following MSIV closure. The staff believes that the qualification and redundance of reactor protection system equipment, coupled with the fact that the reactor vessel pressure is limited to less than 110 percent of design pressure, provides adequate assurance that the reactor vessel integrity will be maintained for the limiting transient event.

The staff evaluation of TMI-2 Action Items (NUREG-0737 requirements that are incorporated into 10 CFR 50.34(f)(1)(vi), (1)(v), (2)(vi), 2(x), and (2)(xi)), as related to SRVs, is discussed in Chapter 20 of this report.

GE performed the overpressure protection analysis for a core loading pattern, which is described in Chapter 4 of the SSAR.

GE originally submitted the design description and the inspection, test, analysis, and acceptance criteria (ITAAC) for SRVs. At the time the DFSER was issued, the ITAAC review was in progress. Therefore, this was DFSER Open Item 5.2.2-1. GE has since submitted revised design description and ITAAC in its certified design material (CDM). The adequacy and acceptability of the CDM are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

The staff concludes that the pressure relief system, in conjunction with the reactor protection system, will provide adequate protection against overpressurization of the RCPB and that the overpressurization system is acceptable and meets the relevant requirements of GDC 15.

5.2.3 Reactor Coolant Pressure Boundary Materials

The staff reviewed RPCB materials in accordance with SRP Section 5.2.3. The RPCB materials are acceptable if they meet the requirements of (1) GDC 1 and 30 as related to quality standards for design, fabrication, erection and testing; (2) GDC 4 as related to compatibility of components with environmental conditions; (3) GDC 14 and 31 as related to extremely low probability of rapidly propagating fracture and gross rupture of the RCPB; (4) Appendix B to 10 CFR Part 50 as related to onsite material cleaning control; (5) Appendix G to 10 CFR Part 50 as related to material testing and acceptance criteria for fracture toughness of the RCPB; and (6) 10 CFR 50.55a as related to quality standards and fracture toughness.

In the DSER (SECY-91-153), the staff noted that the materials used for the construction of RCPB components had been identified by specification and were in conformance with Section III of the ASME Code and NUREG-0313. However, the staff requested that GE use Revision 2 (not Revision 1 as proposed by GE) of NUREG-0313. GE revised the SSAR (Amendment 14) to reference NUREG-0313, Revision 2. GE's compliance with the provisions of the Code for material specifications and conformance with NUREG-0313, Revision 2, satisfy the quality standards of GDC 1 and 30 and 10 CFR 50.55a.

In the DFSER, the staff noted that the construction materials for the RCPB had been identified in SSAR Table 5.2-4 and were compatible with the primary coolant water, which will be chemically controlled in accordance with appropriate technical specifications (TS). This was DFSER TS Item 5.2.3-1.

GE addressed DFSER TS Item 5.2.3-1 in its April 16, 1993, submittal, "Response to TS Items in ABWR DRAFT FSER," which stated that the reactor coolant chemistry limits had been removed from the improved TS (NUREG-1433 and -1434) and thus would not be included in the ABWR TS. The reactor coolant chemistry limits are to be controlled by appropriate administrative controls outside of TS. This is acceptable. Therefore, this item is resolved.

The RCP materials of construction identified in the SSAR that will be exposed to the reactor coolant have been identified and are compatible with the primary coolant water, which will be chemically controlled to maintain adequate water purity. This compatibility has been proven by extensive testing and satisfactory performance. This includes conformance with the recommendations of RG 1.44, "Control of Sensitized Stainless Steel," Revision O, and with the staff guidelines of NUREG-0313, Revision 2. General corrosion of all materials, except unclad carbon and low-alloy steels will be negligible. For unclad carbon and low-alloy steel, GE has provided conservative corrosion allowances for all exposed surfaces in accordance with ASME Code, Section III. This compatibility with the reactor coolant and compliance with the ASME Code satisfy the requirements of GDC 4 as related to the compatibility of components environmental conditions.

The main source of radiation buildup in operating plants is cobalt-60, which is formed by neutron activation of cobalt-59. GE reduced the cobalt content in alloys to be used in high-fluence areas such as fuel assemblies and control rods. It replaced cobalt-base alloys, used for pins and rollers in control rods, with non-cobalt alloys. This will reduce occupational exposure from cobalt-60 during operation and maintenance of plant components.

The materials to be used for the construction of the RCPB are compatible with the thermal insulation used in these areas and conform to the recommendations of RG 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steels," Revision 0. They satisfy the requirements of GDC 14 and 31 as they relate to the prevention of RCPB failure.

The ferritic steel tubular products and the tubular products fabricated from austenitic stainless steel that GE proposes to use will be examined nondestructively in accordance with the provisions of ASME Code, Section III. This is acceptable and satisfies the quality standards of GDC 1 and 30 and 10 CFR 50.55a.

The fracture toughness tests, required by the ASME Code and augmented by Appendix G to 10 CFR Part 50, provide reasonable assurance that adequate safety margins against nonductile behavior or rapidly propagating fracture can be established for all pressure-retaining components of the RCPB. The use of Appendix G to ASME Code, Section III, and the results of fracture toughness tests performed in accordance with the ASME Code and NRC regulations in establishing safe operating procedures provide adequate safety margins during operations, testing, maintenance, and postulated accident conditions. This satisfies the requirements of GDC 14 and 31 and 10 CFR 50.55a regarding the prevention of RCPB fracture and of gross rupture. The use of low-alloy steel is restricted to the RPV. The controls imposed on preheat temperatures for welding of ferritic steels conform to the requirements of ASME Code, Section III, and provide reasonable assurance that components made from ferritic steels will not crack during fabrication. These controls also minimize the possibility of subsequent cracking due to the retention of residual stresses in the weldment and satisfy the quality standards requirements of GDC 1 and 30 and 10 CFR 50.55a.

The controls imposed on electroslag welding of ferritic steels are not necessary because electroslag welding will not be used for RCPB components.

The controls imposed on the welding of RCPB materials under conditions of limited accessibility are in accordance with the recommendations of RG 1.71, "Welder Qualification for Areas of Limited Accessibility," Revision 0, and provide assurance that proper requalification of welders will be required in accordance with the welding conditions. These controls also satisfy the quality standards requirements of GDC 1 and 30 and 10 CFR 50.55a. The controls imposed on stainless steel weld cladding are not necessary because the use of low-alloy steels is restricted to the RPV.

The controls to avoid stress corrosion cracking in RCPB components constructed of austenitic stainless steels conform to the recommendations of RGs 1.44 and 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Plants," and NUREG-0313, Revision 2. Cold work of austenitic stainless steel is controlled by applying limits on hardness, bend radii, and surface finish on ground surfaces. The controls to be followed during material selection, fabrication, examination, and protection, in accordance with these recommendations, in order to prevent excessive yield strength, sensitization, and contamination provide reasonable assurance that the RCPB components of austenitic stainless steels will be in a metallurgical condition that minimizes susceptibility to stress corrosion cracking during service. These controls meet the requirements of GDC 4 pertaining to the compatibility of components with environmental conditions and those of GDC 14 pertaining to the prevention of leakage and failure of the RCPB.

Since hydrogen water chemistry will be used, a hydrogen injection of less than 1 part per million (ppm) in the feedwater will also be used to minimize intergranular stress corrosion cracking (IGSCC) of the reactor internals in the ABWR. To suppress IGSCC, the reactor coolant conductivity will be maintained below 0.3 microsiemen/cm and sufficient hydrogen will be added to the feedwater to reduce the electrochemical potential below -0.23 volt (standard hydrogen electrode). These controls will further ensure that the materials exposed to the reactor coolant will not be subject to IGSCC. This is acceptable.

The staff concludes that the RCPB materials are acceptable because of the above reasons and because they meet the requirements of GDC 1, 4, 14, 30, and 31 of Appendix A to 10 CFR Part 50; of Appendices B and G to 10 CFR Part 50; and of 10 CFR 50.55a.

5.2.4 Inservice Inspection and Testing of the Reactor Coolant Pressure Boundary

SSAR Section 5.2.4 and Table 5.2-8 describe certain commitments and plans for the preservice inspection (PSI) and inservice inspection (ISI) programs. GE discussed basic inspection concepts and general ASME Code provisions because it had reasoned that the requirements of the NRC regulations might be controlled by the date of order of each specific component subject to examination. The staff review was performed in accordance with SRP Section 5.2.4, except as discussed below.

Throughout the service life of the plant, the COL applicant will have the overall responsibility for the ISI of the RCPB although other organizations also will contribute to the examination activity (e.g., the reactor vendor, the architect-engineer and the inspection agency). The staff and GE representatives discussed the issue of ISI requirements during meetings on March 25 and 26, 1992, as documented in an NRC meeting summary dated April 28, 1992. It was determined that GE is responsible for designing the RPV for accessibility to perform PSI and ISI. For all ASME Code, Class 1, 2, and 3 components, the development of the PSI and ISI programs is the responsibility of the COL applicant.

Pursuant to 10 CFR 50.55a(g)(3), "Components which are classified as ASME Code Class 1 shall be designed and be provided with access to enable the performance of inservice examination of such components and shall meet the preservice examination requirements set forth in Section XI of editions of the ASME Boiler and Pressure Vessel Code and Addenda applied to the construction of the particular component." The applicable construction code should be determined by Paragraph NCA-1140 of ASME Code, Section III, but the construction code that is selected must be incorporated by reference in 10 CFR 50.55a(b).

Throughout the service life of a nuclear power facility, components (including supports) that are classified as ASME Code Class 1 must meet the requirements in ASME Code, Section XI, that become effective after the editions specified in 10 CFR 50.55a(g)(2) and are incorporated by reference in 10 CFR 50.55a(b), to the extent practical within the limitations of design, geometry, and materials of construction of the components. The inservice examination of components conducted during the initial 10-year interval must conform to the requirements of the latest edition and addenda of ASME Code, Section XI, incorporated by reference in 10 CFR 50.55a(b) on the date 12 months before the date of issuance of the operating license. The regulations also require that the ISI program be updated for each subsequent 10-year interval to comply with the ASME Code incorporated by reference in 10 CFR 50.55a.

Compliance with the requirements of the regulations for the PSI and ISI must be based on commitments for or by the COL applicant. The applicable ASME Code or Codes for both design and ISIs are not known at the time of design certification. In addition, the regulation permits the COL applicant the option to change to a newer edition of ASME Code, Section XI, by component, during the construction of the plant.

The staff review of the information in the SSAR emphasized design access and the use of an effective nondestructive examination (NDE) methodology. Since the PSI requirements are established and known at the time each component is ordered, 10 CFR 50.55a(g) does not have provisions for "relief requests" for impractical examination requirements. ASME Code, Section XI has provisions to use certain shop and field examinations in lieu of the onsite preservice examination. Therefore, the NRC staff concluded in the DFSER that the COL applicant must incorporate plans for NDE during construction in order to meet all access requirements of the regulations. This was DFSER COL Action Item 5.2.4-1.

GE responded to this COL item by including additional information in SSAR Section 5.2.6.2, "Plant-Specific ISI/PSI." This section states that the COL applicant will submit the complete plant-specific ISI/PSI program to the NRC including references to the edition and addenda of ASME Code, Section XI, that will be used for selecting components subject to examination, a description of the components exempt from examination by the applicable code, and isometric drawings used for the examination. This is acceptable. Further, SSAR Section 5.2.4.2 states that all items within the Class 1 boundary are designed to provide access for the examinations required by ASME Code, Section XI, IWB-2500. This is also acceptable.

ASME Code, Section XI, states that the PSI should be conducted with equipment and techniques equivalent to those that are expected to be used for subsequent ISIs. Ultrasonic testing of RCPB components will improve in the near future, as indicated by ASME Code, Section XI, Appendix VII, "Qualification of Nondestructive Examination Personnel for Ultrasonic Examination," and Appendix VIII, "Performance Demonstration for Ultrasonic Examination Systems." The NRC has referenced in 10 CFR 50.55a(b) the ASME Code, Section XI edition that includes the published Appendix VII. In addition, the NRC staff has established a technical contact to coordinate the implementation of Appendix VIII. Therefore, the staff concluded in the DFSER that GE should include in its SSAR PSI program provisions that ultrasonic testing be performed in accordance with Appendices VII and VIII pursuant to 10 CFR 50.55a(g)(3). This was DFSER Open Item 5.2.4-1. However, on further review the staff determined that this open item could be resolved by the action taken by the COL applicant.

GE responded to this open item by including additional information in SSAR Section 5.2.6.2. This section states that the COL applicant will submit, for staff review, the complete plant-specific ISI/PSI program, including references to the edition and addenda of ASME Code, Section XI, that will be used for selecting components subject to examination, a description of the components exempt from examination by the applicable code, and isometric drawings used for the examination. This is acceptable. Therefore, DFSER Open Item 5.2.4-1 is resolved.

The requirements for the initial ISI program will be determined by the ASME Code in effect 1 year before issuance of the COL. The COL applicant is required to meet the ISI requirements "to the extent practical within the limitations of design, geometry and materials of construction of the components." The regulations have provisions for staff evaluation, on written request, of new ASME Code requirements determined by a licensee to be impractical for its facility.

In the DSER (SECY-91-355), the staff found that the SSAR information pertaining to compliance with 10 CFR 50.55a(g), design access, PSI requirements, and proposed methodology for ISI was unacceptable. This was DSER Outstanding Issue 1. In SSAR Section 5.2.4, GE addressed the topics discussed in SRP Section 5.2.4. GE described the access provisions for examining the major components of the RCPB including the reactor vessel, closure head, RPV studs, nuts and washers, reactor vessel support skirt, piping, pumps, valves, and component supports. GE stated that all items within the Class 1 boundary are designed to provide access for the examinations required by ASME Code, Section XI, IWB-2500. This is acceptable and resolved DSER Outstanding Issue 1.

SSAR Table 5.2-1 lists the applicable ASME Code Cases for the major RCPB components. In a letter dated March 11, 1992, GE stated that the Code addenda requirements for the ABWR plants will comply with 10 CFR 50.55a, except for the RPV, because as of that date the 1989 Edition of the ASME Code had not been referenced by the regulation. The 1989 Edition was subsequently referenced by 10 CFR 50.55a. As a result of a meeting on March 25 and 26, 1992, it was determined that the COL applicant would be responsible for the development of the PSI and ISI programs for all ASME Code, Class 1, 2, and 3 components. Since the applicable ASME Code edition for the PSI and ISI programs at the time of the COL application cannot be determined at the time of design certification, pursuant to 10 CFR 50.55a(g), the development of the PSI and ISI programs for ASME Code, Class 1, 2, and 3 components for the ABWR was DFSER COL Action Item 5.2.4-2. In the DSER (SECY-91-153), the staff noted that its reviews of the PSI and ISI programs were in progress. These were DSER Outstanding Issues 20 and 23. In response to the above items, GE included additional information in the SSAR to state that the COL applicant will submit the complete plant-specific ISI/PSI program to address a number of concerns including those discussed above. This is acceptable. The staff concludes that the SSAR enables the COL applicant to meet the requirements of the NRC regulations defined in 10 CFR 50.55a, except for the RPV, which is evaluated below as an individual component pursuant to 10 CFR Part 52. Therefore, DSER Outstanding Issues 20 and 23 are resolved.

GE committed to design the RPV so that PSI based on the requirements of ASME Code, Section XI, 1989 Edition, can be performed. This edition is specified in the RPV ITAAC, Section 2.1.1, "Reactor Pressure Vessel System." The RPV shell welds are designed for 100 percent accessibility for both PSI and ISI. The RPV nozzle-to-shell welds will be 100 percent accessible for PSI but might have limited areas that will not be accessible from the outer surface for inservice volumetric examination using current examination techniques. The staff will review the ISI program in accordance with the ASME Code edition in effect and the ISI techniques available at the time of the COL application.

The staff reviewed GE's use of the 1989 Edition of ASME Code, Section XI, and evaluated the extent of examination, design access, methodology for NDE, and personnel qualifications. Subarticle IWB-2200 "Preservice Examination" of the 1989 Edition states that examinations required for Class 1 components shall be completed before initial plant startup. In addition, these PSIs should be extended to include essentially 100 percent of the pressure- retaining welds in all Class 1 components except in those "components exempt from examination" as defined by IWB-1220(a), (b), or (c). GE's use of the 1989 Edition of ASME Code for design of the RPV should not significantly change the extent of examination required by 10 CFR 50.55a(g).

GE committed to design the RPV with essentially 100 percent of the shell welds accessible for both PSI and ISI. The RPV nozzle-to-shell welds, a different examination category defined by ASME Code, Section XI, will be 100 percent accessible for PSI. The staff assumes that PSI of the nozzle-to-shell regions will be accomplished by shop and field inspections during construction as permitted by the Code. Although the RPV nozzle-to-shell welds might have limited areas that will not be accessible from the outer surface for inservice volumetric examination using current examination techniques, the actual Code edition for ISI, as defined by 10 CFR 50.55a(g), cannot be determined until the construction of the ABWR plant is essentially complete. Therefore, the staff will review all examination limitations with the COL application. This is acceptable because the concepts described above are equivalent or superior to examinations performed on existing operating BWRs. However, the staff concluded in the DFSER that the COL applicant must monitor the construction of structural supports around the RPV and the installation of auxiliary equipment to ensure that the design access that will be provided is not adversely affected. This was DFSER COL Action Item 5.2.4-3.

GE responded to this item in SSAR Section 5.2.6.2, which states that the COL applicant will submit the complete plant-specific ISI/PSI program to the NRC including references to the edition and addenda of ASME Code, Section XI, that will be used for the selection of components subject to examination, a description of the components exempt from examination by the applicable code, and isometric drawings used for the examination. This is acceptable. GE also stated that the COL applicant will submit plans for preservice examination of the RPV welds to address the degree of compliance with RG 1.150, "Ultrasonic Testing of Reactor Vessel Welds During Preservice and Inservice Examinations," Revision 1. Further, SSAR Section 5.2.4.2 states that all items within the Class 1 boundary are designed to provide access for the examinations required by ASME Code, Section XI, IWB-2500. This is also acceptable.

The SSAR Section 5.2.4.3.2.1 states that ultrasonic testing of the reactor vessel welds will be performed in accordance with RG 1.150 and that personnel performing ultrasonic examinations shall be qualified in accordance with ASME Code, Section XI, Appendix VII. Ultrasonic examination systems will be qualified in accordance with an industry-accepted program for implementing ASME Code, Section XI, Appendix VIII. GE does not consider the supplemental examinations recommended in GE service information letters (SILs) and rapid communication service information letters (RICSILs) for previous BWR designs applicable to the ABWR. In the ABWR design either the components addressed by the SIL or RICSIL (e.g., jet pumps) have been eliminated or the need for the examination no longer exists because of the elimination of crevices and the use of materials resistant to the known degradation mechanisms (e.g., IGSCC) on which the SIL and RICSIL examinations were based. GE's proposed methodology for NDE of the reactor vessel and personnel qualifications are acceptable because the concepts described above have been successfully used in existing operating BWRs.

Periodic examinations and hydrostatic testing of pressure-retaining components of the RCPB by the COL applicant in accordance with ASME Code, Section XI, and 10 CFR Part 50 will provide reasonable assurance that structural degradation or loss of leaktight integrity during service will be detected in time to permit corrective action before the safety functions of a component are compromised. Compliance with the PSI and ISI requirements of ASME Code and 10 CFR Part 50 constitutes an acceptable basis for satisfying GDC 32.

As a vendor-specific issue, GE based its design of the RPV on the 1989 Edition of the ASME Code, which is a national standard referenced by 10 CFR 50.55a(b). The staff reviewed the design of the RPV using the 1989 Edition of the ASME Code to determine if the design is technically acceptable. In the DFSER, the staff concluded that GE's proposal related to the examination of the RPV will be technically acceptable, if it is properly described in the SSAR and the Tier 1 document. GE was asked to revise the SSAR to indicate that it had based its design of the RPV on the 1989 Edition of the ASME Code and that PSI will be performed in accordance with that same code edition, rather than the code in effect at the time of COL application (as required by 10 CFR 50.55(g). In addition, a discussion of the PSI and the 1989 Edition of the Code will be added to the RPV Tier 1 document design information. This was DFSER Confirmatory Item 5.2.4-1.

GE responded to this item in SSAR Section 5.2.4 by stating that the design for performing PSI on the reactor vessel shall be based on the requirements of ASME Code, Section XI, 1989 Edition. For the required preservice examination, the reactor vessel shall meet the acceptance standards of Section XI, IWB-3510. The RPV shell welds are designed for 100-percent accessibility for both PSI and ISI. The RPV nozzle-to-shell welds will be 100-percent accessible for PSI but might have limited areas that will not be accessible from the outer surface for performing inservice examination techniques. The staff reviewed revised SSAR Section 5.2.4 and found it acceptable. After a discussion with the applicant, the staff decided that PSI requirements will include ultrasonic examination in addition to the radiographic examination required by Section III of the Code. These requirements are included in the RPV Tier 1 design description. However, the specific edition of the ASME Code (1989) is considered Tier 2 information as previously discussed in Section 5.2.1.1 of this report. This item is resolved.

5.2.5 Reactor Coolant Pressure Boundary Leakage Detection

The staff reviewed the RCPB leakage detection systems in accordance with SRP Section 5.2.5. Staff acceptance of the RCPB leakage detection systems is based on the design meeting the requirements of GDC 2 as it relates to the ability of systems to maintain and perform their safety functions following an earthquake and meeting the requirements of GDC 30 as it relates to the detection, identification, and monitoring of the source of reactor coolant leakage. Conformance with GDC 2 is based on the design meeting the guidelines of RG 1.29, Positions C.1 and C.2 while conformance with GDC 30 is based on the design meeting the guidelines of RG 1.45, Positions C.1 through C.9.

A limited amount of leakage is to be expected from components forming the RCPB. Leakage is classified into two types, identified and unidentified. Components such as valve stem packing, pump shaft seals, and flanges will not be completely leaktight. Because some leakage is expected from them, it will be collected and monitored. This leakage from the selected components (1) is considered identified leakage and (2) its total flow rate will be monitored separately from unidentified leakage (which may be symptomatic of an unexpected failure of the RCPB). Items (1) and (2) above are requirements of Position C.1 of RG 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," Revision 0. The sensitivity of detection methods (3.79 L/min (1 gpm)) and leakage limits (ranging from 3.79 L/min (1 gpm) to 19 L/min (5 gpm)) for unidentified leakage will be selected to detect and correct potential through-wall flaws (cracks) in the RCPB before such cracks can grow sufficiently to threaten the safety of the plant.

Reactor coolant leakage for the ABWR involves leakage within the drywell, leakage external to the drywell (i.e., in the equipment areas in the reactor building, the main steam tunnel, and the turbine building) and intersystem leakage. These are discussed below.

(1) <u>Leakage Within the Drywell</u>

<u>Identified Leakage:</u> Within the drywell, the drywell equipment drain sump will collect leakage from the reactor vessel head flange inner seal, stem inner packing for large remote power-operated valves, and other known leakage sources. The sump is designed with two pumps, timers, and

instrumentation for control room monitoring. The sump instrumentation and timers will monitor identified leakage by measuring the rate of change in sump level and the sump's fill-up and pump-out times. The ABWR design provides control room indication and alarm capabilities. The sump level monitoring instrument and the fill-up and/or pump-out timer will activate an alarm in the control room when the total leak rate reaches 95 L/min (25 gpm). Different parameters will monitor leakage from individual sources. The leakage from the head flange inner seal will be monitored by means of a seal drain line pressure instrument. Safety/ relief valve (SRV) and valve stem leakage will be monitored by temperature sensors provided on each SRV discharge line. These monitors will continuously indicate and/or record leakage in the control room. In addition, the monitors will trip and activate an alarm in the control room if leakage from the monitored components is detected. The stems for large power-operated valves will be equipped with drain lines to the sump. A remote-operated solenoid valve installed in each drain line can be used during plant operation, in conjunction with additional sump instrumentation, to identify leaking valve packing and to isolate leakage flow from the valve stem's inner seal.

Unidentified Leakage: Within the drywell, the drywell floor drain sump will collect unidentified leakage from sources such as control rod drives, valve flanges, closed cooling water for reactor services (e.g., cooling of RIP motor), condensate from the drywell atmosphere coolers, and other leakage not collected in the drywell equipment drain sump. The sump is designed with two pumps, timers, and instrumentation for control room monitoring. Sump fill-up and pump-out times will be monitored. The instrumentation will activate an alarm in the control room when preset limits are reached. The rate of change in sump level will be indicated continuously in the control room. Other primary detection methods for small unidentified leaks will include increases in condensate flow rate from the drywell air coolers and increases in the radioactivity count level for noble gases, iodines, and particulates in the drywell atmosphere. The flow rate for condensate, which results from condensation of thermally hot leakage inside the drywell, will be monitored by flow instrumentation provided on the common drain line for the condensate from all the coolers. The radioactivity count levels will be monitored by drywell radiation monitors. These variables will be continuously indicated and/or recorded in the control room. The monitors will activate alarms in the control room when their preset limits are reached. The sensitivity and response time for all these primary detection systems is 3.79 L/min (1 gpm) or its equivalent in less than 1 hour, thus satisfying Positions C.2 and C.5 of RG 1.45, Revision 0. Secondary detection methods will include monitoring the drywell temperature and pressure for gross unidentified leakage. These variables will be recorded in the control room and will activate control room alarms when they are high. Excessive leakage inside the drywell (e.g., during a loss-of-coolant accident (LOCA)) will be detected by high drywell pressure, low reactor vessel pressure, or high steamline flow (for breaks downstream of the flow elements). The instrumentation for these monitored variables will trip, activate alarms and isolate the appropriate valves when their predetermined limits are exceeded. Position C.3 of RG 1.45, Revision 0, states that at least three methods of leak detection should be used in the design. Position C.7 of RG 1.45, Revision 0, states that indicators and alarms for each leakage detection system should be provided in the main control room. Also, it states that procedures for correcting various indications to a common leakage equivalent should be available to reactor operators. The leakage detection methods described above meet Position C.3 of RG 1.45, Revision 0, and are acceptable. The instrumentation and alarms discussed in the SSAR meet the indicator and alarm provisions of Position C.7 of RG 1.45, Revision 0 and are acceptable. The applicant referencing the ABWR design will provide the procedures to convert leakage indicators to a common leakage equivalent.

The total leakage rate from the RCPB consists of all leakage, identified and unidentified, that flows into the drywell floor drain and equipment drain sumps. GE specifies an identified leakage rate limit of 95 L/min (25 gpm) and an unidentified leakage rate limit of 3.79 L/min (1 gpm) from the RCPB. GE also states that the total leakage rate from both categories from the RCPB is limited to 95 L/min (25 gpm).

Position C.9 of RG 1.45, Revision 0, requires that the TS include limiting conditions for identified and unidentified leakage and address the availability of various instruments to ensure adequate coverage. The technical specification limits on RCS leakage are 3.79 L/min (1 pgm) for unidentified leakage and 98.4 L/min (26 gpm) for total leakage. Since the above limits are either equal to or less than current BWR standard TS limits for RCPB leakage limits, the system meets Position C.9 of RG 1.45, Revision 0, and is acceptable. This was DFSER TS Item 5.2.5-1. GE has provided the applicable TS. Therefore, DFSER TS Item 5.2.5-1 is resolved.

(2) Leakage External to the Drywell

The areas designated as external to the drywell that are monitored for reactor coolant leakage include the equipment areas in the reactor building, the main steam tunnel, and the turbine building. The process piping for each monitored system is located in separate compartments or rooms, where feasible, to facilitate the detection of the leak through area temperature indications. Each leakage detection system has the capability to detect leak rates that are less than the established limits for identified and unidentified leakage from the RCPB. Many detection and monitoring systems will be used to detect leakage external to the drywell.

- Equipment Areas in the Reactor Building

  Leakage from unknown or unidentified sources, including shutdown cooling system piping, reactor water cleanup system (CUW) piping, process instrumentation piping, and control rod drive hydraulic control unit (HCU) piping is collected in several reactor building floor drain sumps. Identified leakage from known sources will be collected in reactor building equipment drain sumps.

  Pumps, timers, and instrumentation used for processing and monitoring the reactor building equipment and floor drain sumps will be the same as those provided for the drywell equipment and floor drain sumps. Sump levels and sump fill-up and pump-out times will be monitored. Alarms will be initiated in the control room when the applicable set points are reached.

  In addition to the above parameters (i.e., sump levels and sump fill-up and pump-out times), other parameters also will be used for detecting leakage. Equipment rooms that house the RCIC, RHR, and CUW components will be monitored by dual-element thermocouples that sense high ambient temperature. These sensors will indicate possible reactor coolant leaks in these areas. The high-temperature condition will be indicated, alarmed, and recorded in the control room. In some cases, a high-temperature condition will provide isolation signals to close appropriate valves. These monitors will be suitable to detect leakage of 95 L/min (25 gpm) or less into the monitored areas. In addition to area temperature monitoring, leakage external to the drywell originating from specific systems can be detected by monitoring other parameters such as low steam pressure and high steam flow in the RCIC system and main steamlines (MSLs), high RCIC exhaust line diaphragm pressure, and high differential flow between suction and discharge lines of the CUW system. These detection systems will activate alarms in the control room and/or initiate closure of applicable valves when the preset limits for the corresponding parameters are reached.

- Main Steam Tunnel

  Leakage within the main steam tunnel will be detected by monitoring area temperatures and radiation. MSL area temperature will be monitored by thermocouples located in the area of the main steam and RCIC pipelines. All temperature elements will be located and shielded so that they will be sensitive to ambient air temperature rather than the radiated heat from hot equipment. The monitors will alarm in the control room and provide isolation signals to the MSL and the MSL drain line isolation valves as well as the CUW isolation valves. If the leakage is from the feedwater system, the system will be isolated manually. These monitors will be suitable to detect leakage of 95 L/min (25 gpm) or less into the monitored areas. MSL radiation will be monitored by the process radiation monitoring system (PRMS). The PRMS trip functions will include isolation of the MSIVs and reactor scram. The PRMS will initiate control room readouts and alarms.

- Turbine Building

  Reactor coolant leakage in the turbine building will be detected by monitoring MSL pressure, main condenser vacuum, and turbine building ambient temperature in areas traversed by the MSL. These monitors will alarm and indicate in the control room and, in some cases, will also provide signals for isolating the MSIV and MSL drain lines.

Large leaks external to the drywell will be detected by monitoring MSL flow rate, reactor vessel level, RCIC steamline flow rate, and low pressure on the RCIC steamline or the RCIC turbine. Abnormal conditions detected by any of the above monitors will alarm in the control room, and isolation of appropriate system(s) will be initiated.

(3) Intersystem Leakage

Position C.4 of RG 1.45, Revision 0, states that provisions should be made in the design to monitor systems connected to the RCPB for signs of intersystem leakage.

The ABWR design provides for monitoring intersystem leakages (i.e., leakages from the reactor coolant system (RCS) into other connected systems). Detection of leakage into systems connected to the RCS is necessary because inter-system leakage could result in damage to those systems. The systems of concern are:

- HPCF system
- RCIC system
- RHR shutdown cooling (SDC) system
- RHR low-pressure core flooder (LPFL)
- secondary side of RHR heat exchangers
- secondary side of RIP heat exchangers
- secondary side of CUW heat exchangers
- secondary side of fuel pool cooling (FPC) heat exchangers

In response to request for additional information (RAI) Question (Q)430.2c, GE submitted information regarding the detection of intersystem leakage for systems connected directly to the RCS. In the DFSER, the staff stated that this information should be incorporated into SSAR Section 5.2.5. This was identified as Confirmatory Item 5.2.5-1. GE added a reference to the response to RAI Q430.2c in the text of SSAR Section 5.2.5.2.2(11). Since the response is presented in SSAR Chapter 20 and is adequately referenced in Section 5.2.5, Confirmatory Item 5.2.5-1 is resolved. GE's response to the RAI is summarized below.

The ABWR high-pressure safety injection system consists of two HPCF divisions (Divisions B and C) and the RCIC system. There is no connection between the RCS and the inlet suction of HPCF divisions B and C. Both divisions will draw their suction flow from the condensate storage pool or suppression pool, and not from the RCS. The discharge lines of the HPCF divisions connect to the RCS through normally closed discharge check valves and injection valves. Substantial (potential) leakage through the closed discharge check valves and closed injection valves into either of the discharge lines would result in pressurization of both the discharge line and the HPCF pump suction line, leading to a control room alarm indicating high HPCF B (or C) suction pressure. Significant pressurization of the suction piping would result in a discharge to the suppression pool via a pressure relief valve.

There is no connection between the RCS and the suction or discharge of the RCIC system. The system will draw its suction flow from the condensate storage pool or suppression pool, and not from the RCS. The discharge line of the RCIC system connects to feedwater line B through normally closed discharge check and injection valves and is not considered to be connected to the reactor system pressure boundary. Therefore, the provisions of SRP Section 5.2.5 are not applicable to the RCIC system.

Each of the three suction lines for the shutdown cooling mode of the RHR system connect to the RCS through a shutdown cooling line suction valve and through both an inboard and an outboard containment isolation valve. Substantial (potential) leakage through both of the closed containment isolation valves into any of the suction lines would be detected by pressure sensors located between the outboard containment isolation valve and the key-locked closed RHR shutdown cooling mode suction valve. High pressure in this section of piping would result in a control room alarm. Significant pressurization of this section of piping would result in a discharge to the suppression pool via a pressure relief valve.

In the LPFL mode of RHR, suction flow will be drawn from the suppression pool through normally opened RHR pump suction valves, and not from the RCS.

The RHR discharge lines will be used by all three RHR divisions to return flow to the reactor for both the shutdown cooling mode or LPFL mode of operation. The discharge lines normally will be filled with water by the RHR discharge line fill pumps. Only RHR divisions B and C will discharge directly into the RPV; division A will discharge into feedwater line A. Therefore, intersystem leakage is only postulated to occur in RHR divisions B and C. The discharge lines of RHR connect to the RPV through normally closed discharge check valves and injection valves. Substantial (potential) leakage through the closed discharge check valves and closed injection valves into either of the discharge lines would result in pressurization of the discharge line, leading to a control room alarm. Significant pressurization of the discharge piping would result in a discharge to the suppression pool via a pressure relief valve.

Radiation monitors will detect reactor coolant leakage into the reactor building cooling water (RCW) system, which will provide cooling water to the RHR heat exchangers, the RIP heat exchangers, the CUW nonregenerative heat exchangers (NRHXs), and the fuel pool pooling and cleanup (FPC) heat exchangers. At least two process radiation monitoring channels will monitor leakage into each of the two common cooling water headers that will receive RCW return flow from the above heat exchangers. Each channel will alarm on high-radiation conditions that indicate process leakage into the CUW system.

The staff concludes that intersystem leakage applicable to the ABWR, such as that into the RHR, HPCF, and RCIC systems, would be highly unlikely because it would have to occur through closed check valves and/or closed containment isolation valves. All potential intersystem leakage from the RCPB will be into closed systems, normally filled with water. Therefore, indicators for abnormal water levels or flows in the affected areas will not be used for monitoring intersystem leakage. For these systems, high-pressure alarms will be used. Thus, the ABWR design satisfies the requirements of Position C.4 of RG 1.45, Revision 0, and is acceptable.

Position C.8 of RG 1.45, Revision 0, states that leakage detection systems should be equipped with provisions to permit testing for operability and calibration during plant operation.

The ABWR leakage detection systems are equipped with features to permit operability testing and calibration during plant operation using the following methods:

- simulation of input signals into trip units
- comparison of methods (e.g., comparison of airborne particulate monitoring or air cooler condensate flow with sump fill-up rate)
- comparison of channels when more than one channel is used for any one detection method (e.g., area temperature monitoring)

These methods meet Position C.8 of RG 1.45, Revision 0, and are acceptable.

However, in the DFSER, the staff noted that the COL applicant should provide information regarding the sensitivity of detection methods and leakage limits for unidentified leakage and procedures and graphs for converting various indicators into a common leakage equivalent to meet the procedures portion of Position C.7 of RG 1.45. This was DFSER COL Action Item 5.2.5-1. GE has included this action in SSAR Section 5.2.5.9. This is acceptable.

GE originally submitted the design description and the ITAAC relating to RCPB leakage detection. At the time the DFSER was issued, the ITAAC review was in progress. Therefore, this was DFSER Open Item 5.2.5-1. GE has since provided revised design description and ITAAC. The adequacy and acceptability of the design description and the ITAAC are evaluated in Chapter 14.3 of this report. Therefore, DFSER Open Item 5.2.5-1 is resolved.

The leakage detection systems proposed for the ABWR design meet RG 1.45, Revision 0, Positions C.1 through C.5, C.8, and C.9. RG 1.29, Revision 3, states in Positions C.1 and C.2 that systems required for monitoring systems important to safety should be designed to meet seismic Category I standards and to withstand the effects of a safe shutdown earthquake (SSE) without failure. RG 1.45, Revision 0, indicates in Position C.6 that the airborne particulate monitoring system should function when subjected to an SSE, and the leakage detection system should be capable of performing its functions following seismic events not requiring shutdown. The ABWR leakage detection systems are designed to remain functional following seismic events. Specifically, the drywell airborne particulate radioactivity monitoring system is designed to seismic Category I standards. Thus, the requirements of GDC 2, "Design Basis for Protection Against Natural Phenomena," and the guidelines of RG 1.29, Revision 3, Positions C.1 and C.2, and of RG 1.45, Revision 0, Position C.6, are satisfied.

Indicators and alarms for each leakage detection system are provided in the control room. Level and flow from floor and equipment drain sumps, both internal and external to the drywell, will be monitored for leakage. Condensate flows from the drywell air cooler also will be monitored and will alarm in the control room on high flow.

The staff concludes that the leakage detection systems provided to detect leakage from components of the RCPB furnish reasonable assurance that structural degradation that may develop in pressure-retaining components will be detected on a timely basis. Thus, corrective action can be taken before such degradation becomes sufficiently severe to jeopardize the safety of the system, or before the leakage increases to a level beyond the capability of the makeup system to replenish the loss. On the basis of the information in the SSAR and the evaluation above, the staff concludes that the systems are in conformance with the guidelines of RG 1.29, Revision 3, Positions C.1 and C.2, and RG 1.45, Revision 0, Positions C.1 through C.9, and satisfy the requirements of GDC 2 and 30. Therefore, the RCPB leakage detection systems meet the acceptance criteria of SRP Section 5.2.5 and are acceptable.

5.3 Reactor Vessel

5.3.1 Reactor Vessel Materials

The materials specified for construction of the reactor vessel and its appurtenances have been identified by specification on SSAR Table 5.2-4 and found to be in conformance with Section III of the ASME Code. GE has identified special requirements with regard to the control of residual elements that the staff considers acceptable. Compliance with the provisions of the Code for material specifications satisfies GDC 1 and 30 and 10 CFR 50.55a.

Ordinary processes will be used for the manufacture, fabrication, welding, and NDE of the reactor vessel and its appurtenances. Fabrication processes and NDEs will be performed in accordance with the requirements specified in the ASME Code. Compliance with these ASME Code provisions meets GDC 1 and 30 and 10 CFR 50.55a.

In the DSER (SECY-91-153), the staff noted that GE should discuss the onsite cleaning and cleanliness controls for austenitic stainless steel. This was DSER Outstanding Issue 21. As discussed in SSAR Section 5.3.1, the controls to be used during all stages of welding to prevent contamination and sensitization that could cause stress corrosion cracking in austenitic stainless steel conform with the recommendations of RGs 1.37, Revision 0, and 1.44, Revision 0, and NUREG-0313, Revision 2. These controls will provide reasonable assurance (1) that austenitic stainless steel components will be properly cleaned on site, thus satisfying Appendix B to 10 CFR Part 50, and (2) that welded components will not be contaminated or excessively sensitized before or during the welding process, thus satisfying GDC 1, 4, and 30 and 10 CFR 50.55a. This is acceptable and resolved DSER Outstanding Issue 21.

When components of austenitic stainless steels are welded, ASME Code controls will be supplemented by conformance with the recommendations of RG 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal," Revision 3, and NUREG-0313, Revision 2. These controls will provide reasonable assurance that the welds will not contain microcracks and will be resistant to IGSCC and satisfy GDC 1, 14, and 30 and 10 CFR 50.55a as related to the integrity of the RCPB.

When components of low-alloy steels are welded, ASME Code controls will be supplemented by conformance with the recommendations of RG 1.50, "Control of Preheat Temperature for Welding of Low-Alloy Steel," Revision 0. Low-alloy steel components either will be held for an extended time at preheat temperatures to ensure the removal of hydrogen, or preheat will be maintained until postweld treatment. This will provide reasonable assurance that components made from low-alloy steels will not crack during fabrication and will minimize the potential for subsequent cracking. Adherence to the recommendations of RG 1.43, "Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components," Revision 0, is not necessary because the RPV specifications require that all low-alloy steel be produced in accordance with fine grain practice. This will provide reasonable assurance that underclad cracking will not occur during the weld cladding process. These controls satisfy GDC 1 and 30 and 10 CFR 50.55a.

Integrity of the reactor vessel studs and fasteners is ensured by conformance with the recommendations of RG 1.65, "Materials and Inspections for Reactor Vessel Closure Studs," Revision 0, which satisfies GDC 1, 30, and 31 and 10 CFR 50.55a and the requirements of Appendix G to 10 CFR Part 50.

The staff reviewed the fracture toughness of the reactor vessel materials, the RCPB materials, and the materials surveillance program for the reactor vessel beltline region according to SRP Section 5.3.1.

GDC 31 require that the RCPB be designed with sufficient margin to ensure that, when stressed under operating, maintenance, and test conditions, the boundary behaves in a nonbrittle manner and the probability of rapidly propagating fracture is minimized. GDC 32 require that the RCPB be designed to permit an appropriate material surveillance program for the RPV. The fracture toughness requirements for the ferritic materials of the RCPB are defined in Appendices G and H to 10 CFR Part 50.

The edition and addenda of the ASME Code that are applicable to the design and fabrication of the reactor vessel and RCPB components are specified in 10 CFR

50.55a. The ASME Code edition and addenda that are applicable to the ABWR are specified in SSAR Tables 1.8-21 and 3.2-3.

In the ABWR SSAR, GE states that the reactor vessel will comply with the fracture toughness requirements of Appendix G to 10 CFR Part 50. However, the staff concluded in the DFSER that to confirm this conclusion, the COL applicant must provide fracture toughness data based on the limiting reactor vessel materials. This was DFSER COL Action Item 5.3.1-1.

GE responded to this item in SSAR Section 5.3.4 by stating that the COL applicant will provide fracture toughness data based on the limiting reactor vessel materials. This is acceptable.

Appendix G, "Protection Against Nonductile Failures," to Section III of the ASME Code is used, together with the fracture toughness test results required by Appendices G and H to 10 CFR Part 50, to calculate the pressure-temperature limits for the ABWR reactor vessel. The fracture toughness tests required by the ASME Code and Appendix G to 10 CFR Part 50 provide reasonable assurance that adequate safety margins against the possibility of nonductile behavior or rapidly propagating fracture can be established for all pressure-retaining components of the RCPB. These methods will provide adequate safety margins during operating, testing, maintenance, and anticipated transient conditions. Compliance with these code provisions and NRC regulations is acceptable and satisfies GDC 31.

The materials surveillance program will be used to monitor changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region resulting from exposure to neutron irradiation and the thermal environment, as required by GDC 32. The ABWR surveillance program, which must comply with Appendix H to 10 CFR Part 50 and American Society for Testing and Materials (ASTM) E-185-82, "Standard Recommended Practices for Surveillance Tests for Nuclear Reactor Vessels," requires that fracture toughness data be obtained from material specimens that are representative of the limiting base weld and heat-affected-zone materials in the beltline region. These data will permit the determination of the conditions under which the vessel can be operated with adequate margins of safety against fracture throughout its service life.

In the SSAR, GE states that the reactor vessel materials surveillance program will comply with the requirements of Appendix H to 10 CFR Part 50 and ASTM E-185-82. However, the staff concluded in the DFSER that to confirm this conclusion, the COL applicant must provide details of its surveillance program including the specific materials in each surveillance capsule, the capsule lead factors, the withdrawal schedule for each capsule, the neutron fluence to be received by each capsule at the time of its withdrawal, and the vessel end-of-life peak neutron fluence. This was DFSER COL Action Item 5.3.1-2.

GE responded to this item in SSAR Section 5.3.4 by stating that the COL applicant will submit the following information to the NRC: the specific materials in each surveillance capsule, the withdrawal schedule for each surveillance capsule, the neutron fluence to be received by each capsule at the time of its withdrawal, and the vessel end-of-life peak neutron fluence. This is acceptable.

The materials surveillance program will generate information on the effects of irradiation on material properties so that changes in the fracture toughness of the material in the ABWR reactor vessel beltline region can be properly assessed and adequate safety margins against the possibility of vessel failure can be provided. The surveillance program must generate sufficient information to permit the determination of conditions under which the reactor vessel will be operated with an adequate margin against rapidly propagating fracture throughout its service lifetime. On the basis of a 40-year design life, ASTM E-185-82 recommends that three materials surveillance capsules be installed in the reactor vessel beltline. However, in the DFSER, the staff noted that for the ABWR, the design life is expected to be increased to 60 years. Accordingly, GE needed to reassess the number of materials surveillance capsules to be provided to account for the additional 20-year increase in the expected life of the vessel. The staff noted that GE should revise its SSAR to address changes to the materials surveillance program, including the number and location of the vessel surveillance capsules, for the ABWR reactor vessel to account for the 60-year design life of the plant. This was DFSER Open Item 5.3.1-1.

GE addressed this open item in the SSAR. Specifically, GE committed to provide four surveillance capsules instead of the three previously proposed capsules. GE also required that the capsules be placed to produce a lead factor of approximately 1.2 to 1.5. Furthermore, GE specified that the weld metal specimens be made from the same heat of weld wire and lot of flux and that the same welding practice as that for the beltline weld be used. GE also calculated that the predicted end-of-license adjusted reference temperature of the reactor vessel will be less than 38 °C (100 °F). The staff finds this response acceptable because these requirements will ensure that the surveillance program will generate sufficient information to determine the conditions under which the reactor vessel will be operated throughout its 60-year service lifetime. Therefore, this open item is resolved.

Compliance with Appendix H to 10 CFR Part 50 and ASTM E-185-82 ensures that the surveillance program will be capable of monitoring radiation-induced changes in the fracture toughness of the reactor vessel material and will satisfy the requirements of GDC 32 regarding the design of an appropriate materials surveillance program for the reactor pressure vessel.

5.3.2 Pressure-Temperature Limits

The acceptance criteria for reviewing the pressure-temperature limits are given in SRP Section 5.3.2 and Appendices G and H to 10 CFR Part 50. Appendices G and H describe the operating conditions that require pressure-temperature limits and provide the bases for these limits, specifically requiring that pressure-temperature limits provide safety margins at least as great as those recommended in ASME Code, Section III, Appendix G. Appendix G to 10 CFR Part 50 requires additional safety margins whenever the reactor core is critical (except for low-level physics tests) for the materials in the closure flange and beltline regions.

The staff reviewed the pressure-temperature limits that will be imposed on the RCPB during the following operations and tests to ensure that there will be adequate safety margins against nonductile behavior or rapidly propagating failure of ferritic components as required by GDC 31:

- preservice hydrostatic tests
- inservice leak and hydrostatic tests
- heatup and cooldown operations
- core operation - criticality

Appendices G and H to 10 CFR Part 50 require the applicant to predict the amount of increase in reference temperature, RT_{NDT}, resulting from neutron irradiation. The increase in RT_{NDT} resulting from neutron irradiation is then added to the initial RT_{NDT} and the margin to establish the adjusted reference temperature. The staff's recommended method for calculating the increase in RT_{NDT} resulting from neutron irradiation is contained in RG 1.99, "Radiation Embrittlement of Reactor Vessel Materials," Revision 2. The relationships in the guide were derived by statistical analysis of 216 material data points that were reported from the testing of irradiated materials.

These materials were contained in surveillance capsules and had been irradiated inside U.S. commercial nuclear reactor vessels. As more surveillance data become available, this guide may need additional revision, since the relationship between the increase in RT_{NDT} and neutron fluence is empirically derived from analysis of material surveillance data on U.S. commercial nuclear reactor vessels.

The pressure-temperature curves in SSAR Figure 5.3-1 are system hydrotest limits with fuel in the vessel, non-nuclear heating limits, and nuclear (core critical) limits. In the DFSER, the staff noted that these three limits were different from the suggested limits on preservice hydrostatic tests, inservice leak and hydrostatic tests, heatup and cooldown operations, and core operation in SRP Section 5.3.2. The pressure-temperature curves in the SSAR were generic and were not valid for any specific effective full-power years. The staff found that the general shapes of the curves are acceptable, but the curves should be used as a reference only. For approval of any COL, COL applicants must submit plant-specific calculations of RT_{NDT}, stress intensity factors, and pressure-temperature curves similar to those in RG 1.99 and SRP Section 5.3.2. This was DFSER COL Action Item 5.3.2-1.

GE responded to this item in the SSAR, Section 5.3.4, by stating that the COL applicant will submit plant-specific calculations of RT_{NDT}, stress intensity factors, and pressure-temperature curves similar to those in RG 1.99 and SRP Section 5.3.2. This is acceptable.

In response to the staff's questions, GE submitted a calculation of the RT_{NDT} shift for the vessel plate and weld metal. The calculation followed RG 1.99, closely. The calculated RT_{NDT} was low (-22 °C (-8 °F) for the weld metal and -8.4 °C (17 °F) for the plate) because of the low neutron fluence and the low copper and nickel contents used in the calculation.

GE predicted the neutron fluence at end of life to be 6×10^{17} n/cm^2, which is low in comparison to the existing BWR design. In the DFSER, the staff noted that GE should submit additional information to show how the fluence value of 6×10^{17} n/cm^2 was predicted. This was DFSER Open Item 5.3.2-1.

GE responded to this item in SSAR Section 5.3.2.1.5. It stated that fast neutron fluence for the ABWR vessel was evaluated using the Oak Ridge National Laboratory code on the CRAY X-MP Super Computer based on an eighth core symmetry fixed source model. The neutron source was based on a three-dimensional nodal fuel model of ABWR for an integrated equilibrium core with a 26-group neutron spectrum. The results were found to be reasonable in comparison to the BWR/6 calculations performed earlier by the Department of Energy (DOE). In evaluating the relative fluence, the power level and shroud-to-vessel water thickness were taken into account. The incorporation of integral pumps increased the annulus between the shroud and the vessel walls for the ABWR. This led to an order of magnitude reduction in expected fluence. Therefore, this item is resolved.

The staff concludes that the pressure-temperature limits imposed on the RCS for operating and testing conditions to ensure adequate safety margins against nonductile or rapidly propagating failure will be in conformance with the fracture toughness criteria of Appendix G to 10 CFR Part 50 and Section III, including Appendix G, of the ASME Code. The use of operating limits, based on the criteria in SRP Section 5.3.2, provides reasonable assurance that nonductile or rapidly propagating failure will not occur and constitutes an acceptable basis for satisfying the requirements of 10 CFR 50.55a and GDC 1, 14, 31, and 32.

5.3.3 Reactor Vessel Integrity

Although the staff reviewed most areas separately in accordance with its review plans, reactor vessel integrity is of such importance that a special summary review of all factors relating to it was warranted. The staff reviewed the fracture toughness of the ferritic materials for the reactor vessel and the RCPB, the pressure-temperature limits for operation of the reactor vessel, and the materials surveillance program for the reactor vessel beltline. The acceptance criteria and references that are the bases for this evaluation are given in SRP Section 5.3.3.

The staff reviewed the information in each area to ensure that no inconsistencies existed that would reduce the certainty of vessel integrity. The areas reviewed and the sections of this report in which they are discussed are given below.

- RCPB materials design (Section 5.2.3)
- ISI and testing of RCPB (Section 5.2.4)
- reactor vessel materials fabrication methods (Section 5.3.1)
- pressure-temperature limits and operating conditions (Section 5.3.2)

The staff concludes that the structural integrity of the reactor vessel is acceptable and meets the requirements of GDC 1, 4, 14, 30, 31, and 32 of Appendix A to 10 CFR Part 50; Appendices B, G, and H to 10 CFR Part 50; and 10 CFR 50.55a.

The basis for this conclusion is that the design, materials, fabrication, inspection, and quality assurance requirements for the ABWR plant satisfy the NRC regulations and regulatory guides and the rules of the ASME Code, Section III. The stringent fracture toughness requirements of the regulations and the ASME Code, Section III will be met, including requirements for surveillance of vessel materials properties throughout service life, in accordance with Appendix H to 10 CFR Part 50. Also, operating limitations on temperature and pressure will be established for the plant in accordance with Appendix G, "Protection Against Nonductile Failure," to ASME Code Section III and Appendix G to 10 CFR Part 50.

The integrity of the reactor vessel is assured because the vessel

(1) will be designed and fabricated to the high standards of quality required by ASME Code and by any pertinent Code Cases;

(2) will be made from material of controlled and demonstrated high quality;

(3) will be subjected to extensive preservice inspection and testing to provide assurance that the vessel will not fail because of material or fabrications deficiencies;

(4) will operate under conditions and procedures and protective devices that provide assurance that the vessel design conditions will not be exceeded during normal reactor operation, maintenance, testing, and anticipated transients;

(5) will be subjected to periodic inspection to demonstrate that the high initial quality of the reactor vessel has not deteriorated significantly under service conditions;

(6) will be subjected to surveillance to account for neutron irradiation damage so that the operating limitation may be adjusted.

5.4 Components and Subsystem Design

5.4.1 Reactor Recirculation System

This system is not addressed in the SRP. However, since the ABWR reactor recirculation system is unique, as compared to the current BWR designs, an evaluation of the system is provided.

A significant change in the ABWR from current BWR designs is the elimination of the external loops and the incorporation of RIPs for reactor coolant recirculation. The containment volume is reduced as a consequence of using RIPs instead of external pumps because the external large pipes of the reactor recirculation system are eliminated. (Rupture of large-bore external pipes in the lower part of the reactor vessel is eliminated as the design-basis accident). There is no pipe larger than approximately 5 cm (2 in.) below the core; thus, the probability of fuel uncovery is reduced during a loss-of-coolant (LOCA). This improves plant safety performance. The use of RIPs requires a vessel with a larger diameter, approximately 706 cm (278 in.), to allow pump impeller removal. However, this results in reduced neutron flux at the vessel beltline and reduced reactor pressure rates during transients. Full-power operation will be possible with one RIP out of service, improving plant availability. Maintenance of RIPs will be easier than that of current recirculation pumps with less radiation exposure to plant personnel.

The reactor recirculation system consists of 10 RIPs with their shafts, impellers, and diffusers internal to the reactor vessel and will be removed from above for service. The RIPs are mounted vertically onto and through the pump nozzles which are arranged in an equally spaced ring pattern on the bottom head of the RPV. The RIPs are single-stage, vertical pumps driven by variable-speed induction motors. The pump speed will be changeable by varying the voltage and frequency output of the individual pump motor's electrical power supply. The RIPs will provide recirculation flow from the downcomers through the lower plenum and up through the lower grid, the reactor core, and steam separators. The flow rate will range from minimum flow, established by the pump characteristics, to above the maximum flow required to obtain rated reactor power.

The RIP motors are variable-speed, four-pole, ac induction wet-motor type. The operating speed of the pump motor will depend on the variable voltage and the variable frequency output of the adjustable speed drives. The RIP motors will be cooled by circulating water in the motor cavity through the shell side of the recirculation motor heat exchanger. Hot water in the shell side of the heat exchanger will be cooled by the reactor building cooling water system. There is one heat exchanger per motor. A clean purge flow will be provided by the control rod drive system to prevent reactor water from entering the motor cavity region, thereby preventing any impurity buildup. Also, anti-reverse rotation devices will be installed on the motor shaft to prevent possible damage of the motor caused by reverse pump flow. For service, the motors will be removed from below the RPV.

The staff discusses acceptability of the flow coastdown characteristics to maintain fuel thermal margins during abnormal operational transients in Chapter 15 of this report. The RIP inertia is significantly less than the pump inertia for operating BWRs (i.e., about 0.7-second inertia time constant for the ABWR compared with about 3 to 5 seconds for operating BWRs). The ABWR design also uses motor generator (MG) sets as pass-through energy devices on 6 of the 10 RIPs. The MG sets will be powered from two separate electrical distribution systems and used to lengthen the coastdown of two groups of three RIPs on loss of offsite power (LOOP), thus preventing the departure from nucleate boiling on an all-pump trip. No new safety concerns are associated with the use of RIPs, as compared with external recirculation loops. There is significant operating experience with the RIPs. In Europe, nearly 100 pumps are operating in BWRs, approaching 600 pump-years of experience. Few major forced outages have been reported as a result of pump-related problems. Thus, RIP operation did not require the development of new technology.

Tests conducted to verify that the core flow pattern for 9-pump operation is not significantly different from 10-pump operation showed that the flow distribution to the reactor core is uniform even if one RIP is idle.

GE originally submitted the design description and the ITAAC for the recirculation flow control system. At the time the DFSER was issued, the ITAAC review was in progress. Therefore, this was DFSER Open Item 5.4.1-1. GE has since provided revised design description and ITAAC. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

Since the operating limits such as maximum critical power ratio are calculated in Chapter 15 of this report for abnormal transients using RIP coastdown and are found to be acceptable, the staff concludes that the ABWR reactor recirculation system is acceptable.

5.4.2 Steam Generators

This section is not applicable because the ABWR does not use steam generators for power generation.

5.4.3 Reactor Coolant Piping

This section does not apply because the reactor internal pumps (RIPs) are located inside the reactor pressure vessel (RPV) and there is no major external reactor coolant piping connected to it.

5.4.4 Main Steam Line Flow Restrictions

The functional requirements of the main steam line restrictors are reviewed in Section 15.4.3 of this report.

5.4.5 Main Steam Isolation Valve Leakage Control System

This section is not applicable to the ABWR because a main steam isolation valve leakage control system is not used. This is discussed in Section 10.3 of this report.

5.4.6 Reactor Core Isolation Cooling System

The staff evaluated the reactor core isolation cooling (RCIC) system for conformance to SRP Section 5.4.6. The staff's review criteria are based on meeting the following:

(1) GDC 4 as related to dynamic effects associated with flow instabilities and loads.

(2) GDC 5 as related to structures, systems, and components important to safety not being shared among nuclear power units unless it can be demonstrated that sharing will not impair its ability to perform its safety function.

(3) GDC 29 as related to the system being designed to have an extremely high probability of performing its safety function in the event of anticipated operational occurrences.

(4) GDC 33 as related to the system capability to provide reactor coolant makeup for protection against small breaks in the reactor coolant pressure boundary so the fuel design limits are not exceeded.

(5) GDC 34 as related to the system design being capable of removing fission product decay heat and other residual heat from the reactor core to preclude fuel damage or reactor coolant pressure boundary overpressurization.

(6) GDC 54 as related to piping systems penetrating primary containment being provided with leak detection and isolation capabilities.

Unlike most current BWR designs, the RCIC system in the ABWR is a part of the emergency core cooling system (ECCS). The initiation logic is diversified by adding a high drywell pressure input as well as by maintaining the typical system initiation on RPV Level 2. In the ABWR design, system reliability is improved by including a bypass line to the turbine steam inlet valve (F045 dc powered) to provide for a smoother turbine start and reduce the possibility of an overspeed trip. Since the RCIC system is a part of the ECCS, full-flow testing capability is provided using the safety-related suction source (i.e., suppression pool).

The RCIC system is designed as a high-pressure reactor coolant makeup system that will start independent of the ac power supply. All motor-operated valves will be dc operated, except the inboard steam isolation valves. Steam supply inboard isolation valve F035 and inboard bypass valve F048 will be powered from ac power sources; however, valve F035 will normally open and fail as is; therefore, loss of ac power will not prevent RCIC system operation. Inboard bypass valve F048 will be closed during the system operation; and hence, loss of ac power will not prevent RCIC system operation. The system will provide sufficient water to the reactor vessel to cool the core and to maintain the reactor in a standby condition if the vessel becomes isolated from the main condenser and experiences a loss of feedwater flow. The system also is designed to maintain reactor water inventory, in the event of a loss of normal feedwater flow, while the vessel is depressurized to the point at which the RHR system can function in the shutdown cooling mode.

In reviewing Amendment 32 to the SSAR, the staff noted that the previously described RCIC system capability, without ac power, of "8 hours" had been changed to "at least 2 hours." The staff raised a concern to GE that changing the capability from 8 hours to 2 hours would result in a measurable increase in the core damage frequency estimate as related to station blackout. To clarify the changed position, GE stated and the staff agreed that an RCIC system capability of up to 8 hours could only be adequately demonstrated during startup tests when plant steam is available after fuel loading. SSAR Section 5.4.6 stated that the RCIC system is designed to perform its function without ac power for at least 2 hours with a capability up to 8 hours. It further stated that the COL applicant will provide analyses for the as-built facility to demonstrate the 8-hour capability. This is acceptable and resolved the staff's concern.

The RCIC system consists of a steam-driven turbine-pump unit and associated valves and piping capable of delivering makeup water to the reactor vessel through the feedwater system. The steam supply to the RCIC turbine is taken from main steamline B at the upstream side of the inboard MSIV. The steam supply to the RCIC turbine will be ensured even if MSIVs are closed. Fluid removed from the reactor vessel following a shutdown from power operation will be normally made up by the feedwater system and supplemented by inleakage from the control rod drive system. If the feedwater system is inoperable, the RCIC system will start automatically when the water level in the reactor vessel reaches the Level 2 (L2) trip set point or will be started by the operator from the control room. The system is capable of delivering rated flow within 29 seconds of initiation. Primary water supply for the RCIC system comes from the condensate storage tank (CST), and a secondary supply comes from the suppression pool.

The RCIC system design operating parameters, as shown in SSAR Figure 5.4-9b, are consistent with expected operational modes.

Essential components of the RCIC system are designated seismic Category I (in accordance with RG 1.29, Revision 3) and QG B (in accordance with RG 1.26, Revision 3), as discussed in Section 3.2 of this report. The proposed initial test programs are discussed in Chapter 14 of this report.

The RCIC system is housed in the reactor building, which will provide protection against wind, tornados, floods, and other weather phenomena. Compliance with the requirements of GDC 2 is discussed in Section 3.8 of this report. In addition, the system is protected against pipe whip inside and outside the containment, as required by GDC 4 and as discussed in Section 3.6 of this report.

The HPCF and RCIC systems are located in different rooms of the reactor building for additional protection against common-mode failures. Different energy sources will be used for pump motivation (steam turbine for RCIC pump, electric power for HPCF pumps) and different power systems for control power. This diversity conforms to SRP Section 5.4.6.

To protect the RCIC pump from overheating, the RCIC system contains a miniflow line that will discharge into the suppression pool when the line to the reactor vessel is isolated. When sufficient flow to the vessel is achieved, a valve in the miniflow line will automatically close, thus directing all flow to the reactor.

The makeup water system connection to the RCIC system will maintain the pump discharge line in a filled condition up to the injection valve. The makeup water system operation eliminates the possibility of an RCIC pump discharging into a voided pipe and minimize waterhammer effects. A high point vent is provided, and the system will be vented periodically. The RCIC system includes a full-flow test line with water return to the suppression pool for periodic testing. The periodic tests will be performed according to ASME Code, Section XI, as required by the TSs. The staff requires, for any emergency core cooling system (ECCS), that the TSs include a system functional test at least every refueling outage, with simulated automatic actuation and verification of proper automatic valve position to verify that the RCIC pump will develop a minimum flow of 182 m^3/hr (800 gpm). This was DFSER TS Item 5.4.6-1. GE has submitted the ABWR TS which include the functional testing; this is acceptable. Therefore, DFSER TS Item 5.4.6-1 is resolved.

The suction piping of the RCIC system is designed for low pressure. A relief valve, therefore, is provided to protect against overpressurization of the line from the high-pressure piping. The suction pressure piping was upgraded from 1.37 MPa (200 psig) to 2.82 MPa (410 psig) for resolving the generic issue pertaining to interfacing systems loss-of-coolant-accident (ISLOCA) as discussed in Chapter 20 of this report.

Suitable provisions will be provided for isolation of the RCIC system from the RCS by one testable valve and a closed dc-powered valve in the RCIC system discharge line, and two normally open motor-operated valves with appropriate closure signals to terminate the leakage of the pipe as a result of a break outside containment. Inservice testing of pumps and valves is discussed in Section 3.9.6 of this report.

The RCIC system will have controls that will shut down the system if operating conditions exceed certain limits. A leak detection system is provided to detect leakage in the RCIC system.

The CST level transmitter will be supported and mounted in such a way that automatic suction transfer to the suppression pool from the non-seismic tank will take place without failure during a seismic event.

The RCIC system design meets RG 1.1, "Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps," Revision 0.

In the DSER (SECY-91-153), the staff identified Outstanding Issue 24, which required an evaluation of the TMI-2 Action Items as related to the ECCS. The staff's evaluation of these items (NUREG-0737 requirements that are incorporated into 10 CFR 50.34(f)(1)(v) and (1)(ix)), as related to the RCIC system, is provided in Chapter 20 of this report. Therefore, DSER Outstanding Issue 24 is resolved.

During an earlier review, the staff found that the testing of steam isolation valves (F035 and F036) leading to the RCIC turbines in currently operating BWRs did not include actual operating conditions such as a differential pressure of about 7,000 kPa (1,000 psig) and a temperature of 286 °C (546 °F) expected during a steam pipe break downstream of the valves. Thus, there is no verification that the isolation valves will close during a break as a result of dynamic steam flow forces. Generic Issue (GI)-87, "Failure of HPCI Steamline Without Isolation," addresses this concern. Tests performed as part of the NRC effort to resolve GI-87 have reinforced concerns about the operability of motor-operated valves (MOVs) under these design-basis conditions. The staff concerns regarding operability of MOVs are given in Generic Letter 89-10 (June 1989) and its supplements. GE assumed closure of these valves in the steamline break analysis. Therefore, this functional requirement is incorporated into the ITAAC. In the DFSER the staff noted that the COL applicant referencing the ABWR design should verify test data showing the steam isolation valves (F035 and F036) will isolate under actual operating conditions of a differential pressure at about 7,000 kPa (1,000 psig) and a temperature of 286 °C (546 °F). This was DFSER COL Action Item 5.4.6-1. Since verification of the valves performance by the preoperational and power ascention testing is discussed in Chapter 14 of the SSAR, it need not be specified as a COL action item.

GE originally submitted the design description and the ITAAC for the RCIC system. At the time the DFSER was issued, the ITAAC review was in progress. Therefore, this was DFSER Open Item 5.4.6-1. GE has since provided a revised design description and ITAAC. The adequacy and acceptability of the ABWR design description and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

The RCIC system meets GDC 4, 5, 29, 33, 34, and 54 as identified in SRP Section 5.4.6. Compliance with GDC 4 on protecting the system against dynamic effects associated with flow instabilities and loads is discussed in Section 3.6 of this report. Since the ABWR is a single-unit plant, GDC 5 are not applicable. The RCIC system meets GDC 29 and 34 because it is designed to performed its function without the availability of any ac power and, in conjunction with the high-pressure core flooder system, is designed to ensure an extremely high probability of accomplishing its safety function. The RCIC system is used to supply reactor coolant makeup for small leaks. Accordingly the system meets GDC 33. Compliance with GDC 54 is discussed in Section 6.2 of this report.

The staff concludes that the design of the RCIC system conforms to the Commission's regulations and is, therefore, acceptable.

5.4.7 Residual Heat Removal System

The staff evaluated the residual heat removal (RHR) system according to SRP Section 5.4.7. The staff's review criteria are based on meeting the following:

(1) GDC 2 with respect to the seismic design of systems, structures, and components whose failure could cause an unacceptable reduction in the capability of the residual heat removal system. Acceptability is based on meeting position C-2 of RG 1.29 or its equivalent.

(2) GDC 4 as related to dynamic effects associated with flow instabilities and loads (e.g., water hammer).

(3) GDC 5 which requires that any sharing among nuclear power units of structure, systems and components important to safety will not significantly impair their safety function.

(4) GDC 19 with respect to control room requirements for normal operations and shutdown.

(5) GDC 34 which specifies requirements for a residual heat removal system.

The RHR system consists of three independent loops - subsystems A, B, and C. Each loop contains a motor-driven pump, heat exchanger, piping, valves, instrumentation and controls. Each loop can take suction from either the RPV or the suppression pool and will be capable of discharging water to either the RPV or back to the suppression pool through a full-flow test line. Each shutdown cooling loop has its own heat exchanger, which will be cooled by the reactor building cooling water system. RHR subsystems B and C can be used for wetwell and drywell sprays. The RHR system will operate in the following modes:

(1) shutdown cooling

(2) suppression pool cooling

(3) wetwell and drywell spray cooling

(4) low-pressure flooder mode

(5) fuel pool cooling

(6) ac independent water addition

During all six modes of operation, the same major hardware components (e.g., RHR pump, heat exchanger) will be used. Shutdown cooling, fuel pool cooling, wetwell spray cooling, drywell spray cooling, and ac independent water addition will all be started manually. The low-pressure flooder mode and the suppression pool cooling mode will be started automatically. Modes (2), (3), and (4) are reviewed in Sections 6.2 and 6.3 of this report; Mode (5) is reviewed in Section 9.1.3 of this report.

The normal operational mode of the RHR system is the shutdown cooling mode, which will be used to remove decay heat from the reactor core to achieve and maintain a cold shutdown condition. Shutdown cooling will be started manually when the RPV is depressurized to about 931 kPa (135 psig). The heat removed in the RHR heat exchangers will be transported to the ultimate heat sink by the reactor building cooling water system.

There are three suction lines directly from the RPV for shutdown cooling rather than from the external reactor recirculation loop. This design is an improvement over the present single suction line in current operating BWRs because the single line is more vulnerable to a loss of shutdown cooling by single failure of valves in the line. The RHR shutdown cooling return line for RHR subsystem A is routed to the RPV through the feedwater system. The RHR shutdown cooling return lines for RHR subsystems B and C are routed to the RPV directly.

The RHR system also includes an ac-independent water addition subsystem that consists of piping and manual valves connecting the fire protection system to the RHR pump discharge line on loop C downstream of the pump's discharge check valve. This flow path will allow water to be injected into the reactor vessel and the drywell spray during postulated beyond-design-basis conditions where all ac power and all ECCS pumps are unavailable. Additionally, an external hookup outside the reactor building is provided so that a fire truck pump can be connected as an alternative water source.

In Branch Technical Position (BTP) RSB 5-1 of the SRP, the staff recommends that the valves provided in the suction line (valves F010 and F011) of the RHR system to isolate it from the RCS have independent and diverse interlocks to protect the RHR system. Although the ABWR design does not explicitly meet the guidance on diversity, it does meet the intent of the guidance to provide high reliability against inadvertent opening of the isolation valves. The pressure signal that provides the interlock function will be supplied from 2-out-of-4 logic, which has four independent pressure sensor and transmitter inputs. The independence is provided by each being in a separate instrument division. Furthermore, the inboard and outboard valves of the shutdown cooling system also will close on low reactor water level and will be powered from different electrical divisions. This meets the intent of BTP RSB 5-1 and, therefore, is acceptable. High-pressure/low-pressure system interaction protection functions are discussed in Section 7.6.1.3 of this report.

In the DSER (SECY-91-153), the staff noted that GE should describe in detail how the RHR system design meets the pressure relief requirements of BTP RSP 5-1. This was Outstanding Issue 22. This same issue was also identified as Outstanding Issue 31 in a follow up DSER (SECY-91-235). GE investigated potential overpressurization transients when the RHR system is not isolated from the RCS. GE also evaluated all the reactor nozzle penetrations to determine if they could be a source of high pressure into the RPV during shutdown. GE concluded that relatively cold water from the feedwater (condensate pump) system could be injected into the RPV through either of two valves in the feedwater line by inadvertent operator action or by a valve failing open. The condensate pumps are rated for approximately 49,210 L/min (13,000 gpm), which is one-third feedwater flow, with a shutoff head of approximately 4,137 kPa (600 psig). A pressure in the range of 2,758 to 3,447 kPa (400 to 500 psig) could occur upstream of the feedwater inlet valves in the RPV by inadvertent operation of the condensate pumps. This high pressure could lift the RHR pressure relief valve set at 2,275 kPa (330 psig). To provide protection against this situation, which would cause an overpressure condition in the RHR low-pressure piping, a design change was made by GE that will close all feed pump discharge and bypass valves by a reactor water level (L8) signal. The staff concludes that the pressure relief requirements of the RHR system satisfy BTP RSB 5-1 and are acceptable. This resolved DSER (SECY-91-235) Outstanding Issue 31.

Inservice testing of pumps and valves is discussed in Section 3.9.6 of this report. Relief valves are provided in each of the low-pressure lines that interconnect with the RCS to protect against overpressurization from RCS leakage.

The RHR pumps are motor-driven centrifugal pumps and are sized for the low-pressure flooder mode of operation. The available net positive suction head for the RHR pumps is adequate to prevent cavitation and to ensure pump operability in accordance with RG 1.1, Revision 0. Environmental qualification for long-term operability of the RHR pumps is discussed in SSAR Appendix 3I and evaluated in Section 3.11 of this report.

Three discharge line fill pumps are provided to maintain the RHR discharge header full of water and pressurized to reduce waterhammer during system initiation.

Each train of the RHR system can be tested during normal plant operation by pumping water from the suppression pool back into the pool. In the DFSER, the staff noted that the TS should include (1) verification of low-pressure flooder mode operability, (2) demonstration of the capability to start each pump from the control room, and (3) performance of a system functional test without requiring coolant injection into the reactor vessel. This was DFSER TS Item 5.4.7-1. GE has submitted the ABWR TS, which include the above three areas in Section 3.5.1. Hence, DFSER TS Item 5.4.7-1 is resolved.

The preoperational test program for the RHR system is discussed in Chapter 14 of this report.

The RHR system is designed to operate with or without offsite power. The RHR system will be controlled from the control room. RHR shutdown cooling subsystems A and B also can be operated from the Remote Shutdown Panel.

The staff concludes that the RHR system has the capability to bring the reactor to cold shutdown conditions in a reasonable time (within 36 hours as specified by SRP Section 5.4.7).

GE originally submitted the design description and the ITAAC for the RHR system. At the time the DFSER was issued, the ITAAC review was in progress. Therefore, this was DFSER Open Item 5.4.7-1. GE has since provided a revised design description and ITAAC. The adequacy and acceptability of the ABWR design description and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

The staff finds that the RHR system is designed to minimize the possibility of an ISLOCA in the following ways. The low-pressure systems at pump discharge directly interfacing with the RCS are designed with 3,447-kPa (500-psig) piping that provides for a rupture pressure of at least 6,895 kPa (1,000 psig). In addition, the high-/low-pressure motor-operated isolation valves have safety-grade, redundant pressure interlocks. Also the motor-operated valves will only be tested when the reactor is at low pressure. All inboard check valves on the RHR system are testable and have position indication.

As a part of the high-/low-pressure interface design review (Generic Issue 105, "Interfacing Systems LOCA at LWRs"), GE originally proposed to change the design pressure of the RHR suction piping from 1379 to 2827 kPa (200 psig to 410 psig). At the time the DFSER was issued, GE was still assessing this change and other interfaces in the design. Therefore, this was DFSER Open Item 5.4.7-2. The SSAR now describes design improvements to reduce the possibility of an ISLOCA. The adequacy and acceptability of the ABWR design with regard to the ISLOCA concerns are evaluated in Chapter 20 of this report. Therefore, this item is resolved.

RHR system vulnerabilities during shutdown and low-power operation are addressed in Section 19.4 of this report.

The RHR system meets the requirements of GDC 2, 4, 5, 19, and 34 as identified in SRP Section 5.4.7. It is designed to the seismic Category I recommendations of RG 1.29, as discussed in Section 3.2 of this report. It is housed in the reactor building for protection against the effects of flooding, tornados, hurricanes, and other natural phenomena. Compliance with GDC 2 is discussed in Section 3.2 of this report. Compliance with GDC 4 on protecting the system against dynamic effects associated with flow instabilities and loads are discussed in Section 3.6 of this report. Since the ABWR is a single-unit plant, GDC 5 are not applicable. The RHR system meets GDC 19 because its design includes necessary instrumentation and controls with respect to control room requirements. By meeting BTP RSB 5-1, the RHR system complies with GDC 34.

The staff concludes that the design of the RHR system conforms to the Commission's regulations and is, therefore, acceptable.

5.4.8 Reactor Water Cleanup System

The staff reviewed the reactor water cleanup system (CUW) description and piping and instrumentation diagrams in accordance with SRP Section 5.4.8. Staff acceptance of the design is based on compliance with the requirements of (1) GDC 1 as related to the design meeting standards commensurate with the system's safety function; (2) GDC 2 as related to the system being able to withstand the effects of natural phenomena; (3) GDC 14 as related to assuring the integrity of the RCPB; (4) GDC 60 as related to the capability of the system to control the release of radioactive effluents to the environment; and (5) GDC 61 as related to designing the system with appropriate confinement.

The CUW performs the following functions:

(1) Removes solid and dissolved impurities from the reactor coolant.

(2) Provides containment isolation which ensures that the major portion of the system is outside the RCPB.

(3) Discharges excess reactor water during startup, shutdown, and hot standby conditions to the radwaste system or the main condenser.

(4) Provides flow to the RPV head spray for rapid RPV cooldown.

(5) Minimizes RPV temperature gradients by maintaining circulation in the bottom head of the RPV during periods when the reactor internal pumps are not operating.

The CUW is a closed-loop system comprised of a regenerative heat exchanger, two nonregenerative heat exchangers (cooled by RCW), two filter-demineralizers, two circulating pumps, system isolation valves, and other piping, valves, and instrumentation. The system takes its suction from the shutdown cooling suction line of RHR loop "B" and the RPV bottom head drain line. CUW discharges back to the vessel via either the feedwater system or the RPV head spray line. The system capacity is 2 percent of rated feedwater flow, which is twice the capacity of previous BWRs.

The majority of the system (including all safety-related components) is located in the nondivisional quadrant of the secondary containment portion of the reactor building. The system is classified as non-safety related with the exception of the containment isolation valves. System piping and components in the drywell, up to and including the outboard containment isolation valves, form part of the RCPB are classified as seismic Category I, QG A. The remainder of the system is classified as nonseismic, QG C. Low-pressure piping in the backwash and precoat area downstream of the block valves is classified as QG D. Nonsafety components receive power from non-class 1E supplies. Based on this QG and seismic classification design information, the staff concludes that the CUW system meets the guidelines of RGs 1.26 and 1.29 and, therefore, the requirements of GDC 1 and GDC 2 as they relate to the CUW design meeting the standards commensurate with the system's safety function and the system being able to withstand the effects of natural phenomena.

The two safety-related containment isolation valves on the suction line of the CUW system and the isolation valve on the RPV head spray line receive isolation signals from the leak detection system (LDS). These valves are powered by class-1E power sources and automatically isolate on indications of low reactor water level, high ambient main steam tunnel area temperature, high mass differential flow, high ambient CUW equipment area temperature, or initiation of the standby liquid control (SLC) system. The isolation valves are designed and tested to ensure that they close during maximum flow and differential pressure conditions. Based on this information, along with the seismic and quality group classifications for that portion of the system up to and including the containment isolation valves discussed earlier, the staff concludes that the system meets the requirements of GDC 14 as it relates to assuring the integrity of the RCPB.

GE performed pressure and temperature and cooldown analyses for CUW breaks inside the secondary containment. The resulting pressures and temperatures were within the environmental qualification envelope for components which would be affected by the breaks and were within the design pressure and temperature limits for the primary and secondary containment structures (see Chapter 6 of this report).

The CUW suction line contains a flow restrictor inside the containment to monitor flow and to restrict flow during a line break, as well as a motor-operated shutoff valve for long-term system isolation. In addition, upon indication of high radiation levels in the secondary containment, SGTS actuates and processes potentially radioactive effluents before release to the environment. FDU vents are routed to the backwash receiving tank while piping vents and drains are routed to the low conductivity portion of the liquid radwaste system. Unidentified leakage from CUW piping is collected and monitored in the reactor building floor drain sumps. These design features, along with the automatic system isolation features discussed earlier, ensure that releases of radioactive effluents to the environment are limited. This satisfies the requirements of GDC 60 as it relates to controlling the release of radioactive material.

The filter demineralizers are of the pressure precoat type using filter aid and powdered mixed ion-exchange resins as a filter and an ion-exchange medium. Spent resins cannot be regenerated and will be sluiced from the filter-demineralizer unit to a backwash-receiving tank from which they will be transferred to the radwaste system for processing and disposal. Resins will be discarded on the basis of filter-demineralizer performance, as indicated by monitored effluent conductivity, differential pressure across the unit, and sample analysis. A strainer is installed in the effluent line of each filter demineralizer to prevent resins from entering the system in the event of failure of a filter-demineralizer resin support. Each strainer and filter-demineralizer vessel has a control room alarm that will be energized by high differential pressure. If differential pressure increases beyond the alarm point, the filter demineralizer will isolate automatically. Based on this information (along with the information discussed earlier regarding system vents and drains), the staff concludes that the CUW design meets the requirements of GDC 61 as it relates to designing the system with adequate confinement features.

Hydrogen water chemistry that conforms to with the guidelines contained in Electric Power Research Institute (EPRI) Report NP-4947-SR, "BWR Hydrogen Water Chemistry Guidelines," 1987 Revision, will be used. This document also includes hydrogen water chemistry limits, responses to out-of-specification water chemistry, chemical analysis methods, data management and surveillance schemes, and management philosophy required to establish and implement a successful water chemistry control program. Therefore, the CUW system design satisfies the reactor chemistry limits specified in RG 1.56, "Maintenance of Water Purity in Boiling Water Reactors," Revision 1, and EPRI Report NP-4947-SR.

The Advisory Committee on Reactor Safeguards (ACRS) performed an independent review of the CUW system and provided the results to both GE and the NRC staff. GE reviewed the results and provided responses to each of the ACRS concerns identified in the ACRS report. The staff has reviewed GE's responses and has concluded that the modified CUW design adequately addresses the issues raised during the ACRS independent review and documented in the report.

Major concerns in the ACRS report included (1) CUW configuration and design features, (2) CUW processing capacity, (3) quality group and seismic classification of components, (4) quality and consistency of information in the SSAR, (5) design of system isolation valves, and (6) role of the system in safety analyses.

(1) CUW Configuration and Design Features

The ACRS concerns in this area included (a) the letdown path for excess water during plant heatup, (b) reverse flow from the feedwater system into the main steam tunnel (MST) or secondary containment on a CUW pipe break, and (c) high-temperature isolation of the system's filter-demineralizer units (FDUs) and resin retention in the FDUs.

(a) LETDOWN PATH FOR EXCESS WATER

It was unclear whether the CUW system provided letdown to the condensate storage tank. GE clarified that CUW discharges excess water to the main condenser or to the liquid waste management system.

(b) REVERSE FLOW FROM THE FEEDWATER SYSTEM

GE modified the system design to ensure that reverse flow from the feedwater system to the secondary containment and steam tunnel would not occur as a result of a CUW line break in these areas. Previously, check valve G31-F014 and motor operated valve G31-F015 were located inside secondary containment with G31-F014 upstream of G31-F015. In this configuration, a break in the CUW line upstream of these valves, along with a failure of G31-F015 to close, could result in excessive blowdown of the CUW line into the secondary containment area. In response to this concern, GE modified the valve arrangement so that G31-F014 is now inside the MST next to the tunnel wall and G31-F015 is located against the outer MST wall. In this configuration, a CUW line break with a failure of G31-F015 will not result in excessive system blowdown into the secondary containment because check valves in the nuclear boiler system (NBS) (B21-F006A and B21-F006B) as well as G31-F014 are available to prevent reverse flow. In addition, two motor-operated valves in the NBS (B21-F007A and B21-F007B) are available to provide additional long-term manual isolation. These NBS valves are classified as safety-Class 2, Quality Group B, and seismic Category I.

The three containment isolation valves for CUW are designed to close on a signal from the Leakage Detection and Isolation System (LDS). The LDS provides this signal on low reactor water level, high ambient MST area temperature, high mass differential flow, high ambient CUW equipment area temperature, and SLCS actuation. Based on this information, the staff concludes that the modified CUW design provides assurance that excessive feedwater or CUW blowdown will not occur as a result of a CUW pipe failure inside secondary containment.

(c) HIGH-TEMPERATURE ISOLATION OF THE SYSTEM'S FILTER-DEMINERALIZER UNITS (FDUs) AND RESIN RETENTION IN THE FDUs

The ACRS report identified a concern regarding protection of the FDU resins from high temperature water. The high temperature setpoint for automatic bypass of the FDUs was not provided in the SSAR. GE later clarified that the FDUs are isolated and bypassed at 54 °C as shown on SSAR Figure 5.4-13, sheet 2.

(2) CUW Processing Capacity

The ACRS identified several concerns related to CUW flexibility and processing capacity. These included (a) increased processing capacity and (b) the effectiveness of the nonregenerative heat exchanger (NRHX) when used as an alternate means of decay heat removal.

(a) INCREASED PROCESSING CAPACITY

The ABWR design doubles the traditional CUW processing capacity from 1 percent in current BWRs to 2 percent in the ABWR. The ACRS report expressed concern regarding the impact of this increase. Specifically, the ACRS was concerned that design details such as resin bed volumes, resin types, and resin bed replacement criteria were not provided in the SSAR. Subsequently, GE clarified that the ABWR commits to maintaining the water chemistry requirements stated in the SSAR. The COL applicant will determine the proper components to meet the chemistry requirements.

(b) EFFECTIVENESS OF THE NON-REGENERATIVE HEAT EXCHANGER (NRHX) WHEN USED AS AN ALTERNATE MEANS OF DECAY HEAT REMOVAL

The ACRS report stated that additional heat exchanger parameters are needed to determine if the NRHX can be used as an effective means of removing decay heat during high pressure conditions with RHR unavailable. Subsequently, GE provided information which showed that the use of the NRHXs is not a primary means of accomplishing the heat removal function and that this method of heat removal has only a small impact on the PRA core damage frequency.

(3) Quality Group and Seismic Classification of Components

The ACRS report identified a concern relating to the conversion of Japanese safety classification codes to equivalent domestic codes. Specifically, ambiguities were identified in note 11 of SSAR Table 1.7-1 (this table correlates the Japanese and domestic codes). As a result, GE modified this table to remove any ambiguities.

(4) Quality and Consistency of Information in the SSAR

The ACRS expressed a concern regarding the quality and consistency of information not only in the SSAR section relating to CUW, but also in other SSAR sections. The staff has worked with GE throughout the review process to identify any ambiguities and inconsistencies found in the SSAR. These have largely been identified and corrected.

(5) Design of System Isolation Valves

The ACRS report identified a concern regarding the design adequacy of the system isolation valves. Specifically, current BWRs must meet Generic Letter 89-10 regarding reliable containment isolation valve closure against maximum differential pressure and flow. However, despite this, problems still exist. Demonstration of the reliability and capability of these valves through the use of in-service testing cannot be achieved since the worst-case conditions cannot be created in-situ. As a result, the assumptions used in the safety analysis and PRA may not be valid. In response, GE clarified that type-testing of all isolation valves are conducted in the shop at worst-case design conditions and each installed valve is tested in-situ. The requirements on these valves are provided in SSAR Section 3.9.6 and evaluated in Section 3.9.6 of this report. In addition, verifications are made as part of the ITAAC process.

(6) Role of the System in Safety Analyses

The ACRS report identified concerns related to system failure as it relates to the safety analysis. This included (a) maximum inventory loss and (b) isolability of the bottom head drain line.

(a) MAXIMUM INVENTORY LOSS.

The ACRS report stated that because of the uncertainty of whether the CUW line break was evaluated in Chapter 6 of the SSAR, it appeared that the total inventory loss from this break (assuming a 30-second CUW line isolation) may be more than that from a main steam line (MSL) break with subsequent MSIV isolation. Subsequently, GE clarified that the worst-case break outside containment (from a loss-of-mass perspective) occurred as a result of a feedwater line break. The CUW line break analysis assumed a 75-second isolation time (45 second time delay on system differential flow indication and 30 second closure time). In this conservative analysis the total inventory loss was less than that from a MSL break with subsequent MSIV isolation.

(b) ISOLABILITY OF THE BOTTOM HEAD DRAIN LINE

The original CUW design included a manually-operated valve on the reactor pressure vessel (RPV) bottom head drain line inside containment. The ACRS report identified a concern regarding isolability of this line. Subsequently, GE modified the design to make this a remote manually-operated shutoff valve and moved the valve to a position immediately upstream of the inboard containment isolation valve on the CUW suction line. In addition, the RPV bottom head drain line connects to the main CUW suction line at a "tee." The centerline of this connection is at least 389 mm (15.3 in.) above the top of active fuel. If an unisolated CUW line break should occur, the shutoff valve can be used to isolate the break. If this were to fail, the RPV water level would be maintained at the level of the "tee" connection. An isolated line break is discussed in SSAR Subsection 19.9.1.

Other concerns identified in the ACRS report related to the treatment of CUW in the PRA and the ITAAC. In both cases GE provided information which resolved the concerns.

Based on the information provided by GE in response to the concerns raised in the ACRS report, the staff concludes that the issues have been adequately addressed.

In the DFSER, the staff requested GE to provide adequate design description and inspections, tests, analyses, and acceptance criteria (ITAAC) relating to CUW. GE provided a revised set of design description and ITAAC. The adequacy and acceptability of the design description and ITAAC are evaluated in Section 14.3 of this report.

The reactor water cleanup system (CUW) will be used to aid in maintaining the reactor water purity and to reduce the reactor water inventory as required by plant operations. The review has included piping and instrumentation diagrams and process diagrams along with descriptive information concerning the system design and operation.

The staff concludes that the proposed design of CUW is acceptable and meets the relevant requirements of General Design Criteria 1, 2, 14, 60, and 61. This conclusion is based on the following:

(1) GE has met the requirements of GDC 1 by designing, in accordance with the guidelines of RG 1.26, the portion of the CUW extending from the reactor vessel to the outermost primary containment isolation valves to Quality Group A and by designing, in accordance with position C.2.C of RG 1.26, the remainder of the CUW outside primary containment (excluding the precoat unit) to Quality Group C.

(2) GE has met the requirements of GDC 2 by designing, in accordance with positions C.1, C.2, C.3, and C.4 of RG 1.29, the portion of the CUW extending from the reactor vessel to the outermost primary containment isolation valves to seismic Category I.

(3) GE has met the requirements of GDC 14 by meeting RG 1.56 in maintaining reactor water purity and material compatibility to reduce corrosion probabilities, and thus reducing the probability of RCPB failure.

(4) GE has met the requirements of GDC 60 and 61 by designing a system containing radioactivity with confinement and by venting and collecting drainage from CUW components through closed systems.

Based on this information, the staff concludes that the CUW design for the ABWR is acceptable.

6 ENGINEERED SAFETY FEATURES

6.1 Engineered Safety Features Materials

6.1.1 Metallic Materials

The staff reviewed the engineered safety features (ESF) materials in accordance with SRP Section 6.1.1. The engineered safety features materials are acceptable if they meet the requirements of: GDC 1 and Section 50.55a of 10 CFR Part 50 as they relate to quality standards being used for design, fabrication, erection, and testing of ESF components and the identification of applicable codes and standards; GDC 4, as it relates to compatibility of ESF components with environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents; GDC 14, as it relates to design, fabrication, and testing of reactor coolant pressure boundary so as to have extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture; GDC 31, a it relates to extremely low probability of rapidly propagating fracture and gross rupture of the reactor pressure coolant boundary; GDC 35, as it relates to assurance that the clad metal-water reaction is limited to negligible amounts; GDC 41 as it relates to control of the concentration of hydrogen in the containment atmosphere following postulated accidents to assure that the containment integrity is maintained; and Appendix B to 10 CFR Part 50 as it relates to the requirement that measures should be established to control the cleaning of material and equipment in accordance with work and inspection instructions to prevent damage or deterioration.

To meet the requirements of GDC 1, 14, and 31 of Appendix A to 10 CFR Part 50, the requirement of 10 CFR 50.55a and to ensure that the ESF materials perform the necessary safety function, the ASME Code and industry standards should be satisfied for the ABWR design. SSAR Table 5.2-4 lists the pressure-retaining materials and materials specifications for the reactor coolant pressure boundary components. SSAR Table 6.1-1 lists the pressure-retaining materials and materials specifications for the primary containment system, emergency core cooling systems (ECCSs) and their auxiliary systems, and the standby liquid control system. In the DSER (SECY-91-153), the staff stated that the ESF materials should satisfy GDC 1, 14, and 31 and 10 CFR 50.55a to ensure low probability of leakage, rapidly propagating failure, and gross rupture. To do so, the ESF materials selected should satisfy Appendix I, "Design Stress Intensity Values, Allowable Stresses, Material Properties, and Design Fatigue Curves," to Section III of the ASME Code and Parts A, B, and C of Section II of the code. In its transmittal of October 9, 1991, GE addressed this item by revising SSAR Section 6.1.1.1.1 to state that the ESF materials will satisfy Appendix I to Section III of the ASME Code and Parts A, B, and C of Section II of the code. This revision was acceptable and resolved part of Open Item 25 from the DSER (SECY-91-153).

To meet the requirements of GDC 4, 14, and 41, the water used in the ESF systems should be controlled to provide assurance that stress corrosion cracking of unstabilized austenitic stainless steel components will not occur. In the DSER (SECY-91-153), the staff made the following recommendations:

- GE should follow Electric Power Research Institute (EPRI) report, "Boiling Water Reactor (BWR) Water Chemistry Guidelines," NP-3589-SR-LD, April 1985, for the water used for ECCS and spray systems.
- GE should follow EPRI report, "Guidelines for Permanent BWR Hydrogen Water Chemistry Installations - 1987 Revision," NP-5283-SR-A, September 1987, for the design, construction, and operation of hydrogen water chemistry installations.
- GE should follow EPRI report, "BWR Hydrogen Water Chemistry Guidelines: 1987 Revision," NP-4947-SR, October 1988, for water chemistry limits, responses to out-of-specification water chemistry, chemical analysis methods, and data management and surveillance and management philosophy required to establish and implement a successful water chemistry control program.

In the DSER (SECY-91-153), the staff also stated that SSAR Sections 6.1.1.1.2 and 6.1.1.2 should have adhered to the above recommendations. This was DSER Open Item 25.

In its letter of March 11, 1992, GE responded to this item by revising SSAR Section 9.3.9 (SSAR Amendment 21) to state that the hydrogen water chemistry system will conform to the guidelines in EPRI reports NP-5283-SR-A and NP-4947-SR. This response was acceptable and resolved Open Item 25 from the DSER (SECY-91-153); however, because GE also might use pure water chemistry, the staff concluded in the DFSER that GE should commit to follow EPRI Report NP-3589-SR-LD. This was identified as DFSER Confirmatory Item 6.1.1-1. GE advised the staff in its letter of March 8, 1993, that compliance with EPRI report NP-3589-SR-LD was no longer appropriate because it had committed to use hydrogen water chemistry and was following the guidelines of EPRI report NP-4947-SR. The staff finds this commitment acceptable. This resolved DFSER Confirmatory Item 6.1.1-1.

SSAR Section 6.1.1.1.3 adheres to the materials specifications that satisfy Appendix I to Section III of the ASME Code and Parts A, B, and C of Section II of the code; therefore, Section 6.1.1.1.3 satisfies GDC 4, 14, and 35 and Appendix B to 10 CFR Part 50 as they pertain to materials specifications.

The controls to be placed on concentrations of leachable impurities in nonmetallic thermal insulation used on ESF components comply with the guidelines of RG 1.36, "Nonmetallic Thermal Insulation of Austenitic Stainless Steels," Revision 0. Therefore, by complying with RG 1.36, the ESF thermal insulation specifications meet the requirements of GDC 1, 14, and 31.

GE has met GDC 1, 14, and 31 and 10 CFR 50.55a with respect to ensuring an extremely low probability of leakage, of rapidly propagating failure, and of gross rupture, since the materials selected for the ABWR ESF satisfy Appendix I to Section III of the ASME Code and Parts A, B, and C of Section II of the code. During bending and fabrication, the bend radius, the material hardness, and the surface finish or ground surface are to be controlled. If the controls are not met, the materials are required to be solution heat-treated again. Fracture toughness specifications for the ferritic materials meet the requirements of the code.

The controls on the use and fabrication of the austenitic stainless steel of the systems satisfy the guidance of RG 1.31, "Control of Ferrite Content of Stainless Steel Weld Metal," Revision 3, and RG 1.44, "Control of the Use of Sensitized Stainless Steel," Revision 0. Fabrication and heat treatment practices performed in accordance with these requirements provide added assurance that the probability of stress corrosion cracking will be reduced during the postulated accident time interval.

Conformance with the code and RGs and with the staff positions mentioned above constitutes an acceptable basis for meeting the requirements of GDC 1, 4, 14, 35, and 41; Appendix B to 10 CFR Part 50; and 10 CFR 50.55a, according to which the systems are to be designed, fabricated, and erected so that they can perform their functions as required.

As discussed above, GE has met GDC 1, 14, 31, and 35 and Appendix B to 10 CFR Part 50 with respect to ensuring that the ABWR reactor coolant boundary and associated auxiliary systems will have an extremely low probability of leakage, of rapidly propagating failures, and of gross rupture.

Based on the above discussion, the staff concludes that the ESF metallic materials satisfy the requirements of GDC 1, 4, 14, 31, 35, and 41; and the requirements of 10 CFR Part 50.

6.1.2 Protective Coating Systems (Paints) and Organic Material

The staff performed its review in accordance with SRP Section 6.1.2. The protective coating systems are acceptable if the protective coatings that will be applied inside the containment meet the requirements of Appendix B to 10 CFR Part 50 as it relates to the quality assurance (QA) requirements for the design, fabrication, and construction of safety-related structures, systems, and components. The Appendix B to 10 CFR Part 50 requirements are satisfied if the protective coatings conform to the testing requirements of ANSI N101.2-1972, "Protective Coatings (Paints) for Light Water Reactor Containment Facilities," and the QA guidelines of RG 1.54, "Quality Assurance Requirements for Protective Coating Applied to Water-Cooled Nuclear Power Plants." This provides assurance that the protective coatings will not fail under design-basis-accident (DBA) conditions and generate significant quantities of solid debris that would impair the performance of the ESF.

GE stated that the coating system materials to be used on the exposed surfaces within the drywell will be qualified in accordance with ANSI N101.2-1972. The protective coating system for the containment will be applied in accordance with RG 1.54. However, GE has indicated that the COL applicant may use materials in relatively small areas in the drywell that will not conform to RG 1.54.

Since GE will meet the QA requirements of Appendix B to 10 CFR Part 50 and since the containment coating systems have been evaluated for suitability to withstand a postulated DBA environment, the staff concluded in the DFSER that the protective coating systems and their applications were acceptable.

However, the staff also concluded in the DFSER that the COL applicant referencing the ABWR design should indicate the total amount of protective coatings and organic materials used inside the containment that did not meet the requirements of ANSI N101.2-1972 and RG 1.54. This was identified as DFSER COL Action Item 6.1.2-1.

GE included this information in SSAR Section 6.1.3, which requires that the COL applicant indicate the total amount of protective coatings and organic materials used inside the containment that do not meet the requirements of ANSI N101.2 and RG 1.54. This is acceptable.

6.2 Containment Systems

The containment systems for the ABWR include a containment structure as the primary containment, a secondary containment (reactor building) surrounding the primary containment and housing equipment essential to safe shutdown of the reactor and fuel storage facilities, and supporting systems. The primary containment is designed to prevent the uncontrolled release of radioactivity to the environment with a leakage rate of 0.5 percent by weight per day at the calculated peak containment pressure related to the DBA. The secondary containment is designed to confine the leakage of airborne radioactive materials from the primary containment. SSAR Figure 6.2.1 shows the principal features of the ABWR containment.

6.2.1 Primary Containment Functional Design

The ABWR primary containment is designed with the following main features:

- A drywell that consists of two volumes: (1) an upper drywell (UD) volume surrounding the reactor pressure vessel (RPV) and housing the steam and feedwater lines and other connections of the reactor primary coolant system, safety/relief valves (SRVs), and the drywell heating, ventilation, and air conditioning (HVAC) coolers; and (2) a lower drywell (LD) volume housing the reactor internal pumps (RIPs), control rod drives, and under-vessel components and servicing equipment.

  The UD is a cylindrical, steel-lined, reinforced-concrete structure with a removable steel head and a reinforced-concrete steel diaphragm floor. The cylindrical RPV pedestal, which is connected rigidly to the steel diaphragm floor, separates the LD from the wetwell. Ten UD-to-LD connecting vents (DCVs), approximately 1 m x 2 m (3.3 ft x 6.6 ft) in cross-section, are built into the RPV pedestal. The DCVs extend downward through steel pipes with an inside diameter of 1.2 m (4 ft), each of which has three horizontal vent outlets into the suppression pool.

  The drywell, which has a net free volume of 7,350 m^3 (259,563 ft^3), is designed to withstand design pressure and temperature transients following a loss-of-coolant accident (LOCA) and also the rapid reversal in pressure when the steam in the drywell is condensed by ECCS flow during post-LOCA flooding of the RPV. A wetwell-to-drywell vacuum relief system will prevent backflooding of the suppression pool water into the LD and protect the integrity of the steel diaphragm floor slab between the drywell and wetwell and the drywell structure and liner. The drywell design pressure and temperature are 310 kilopascal (kPa) (45 pounds per square inch gauge (psig)) and 171 °C (340 °F), respectively. The design drywell-to-wetwell differential pressures are +172.4 kPa (25 psig) and -13.8 kPa (-2 psig). The design drywell-to-reactor building negative differential pressure is -13.8 kPa (-2 psig).

- A system of drywell-to-wetwell vent channels that will blow down from the drywell and discharge into the suppression pool following a LOCA. There are 30 vents in the vertical section of the LD below the suppression pool water level, each with a nominal diameter of 0.7 m (2.3 ft). These vents are arranged in 10 circumferential columns, each containing three vents. The three-vent centerlines in each column are located at 3.5 m (11.48 ft), 4.87 m (15.98 ft), and 6.24 m (20.48 ft) below the suppression pool water level when the suppression pool is at the low water level.

- A wetwell that consists of an air volume and suppression pool, with a net free-air volume of 5,960 m^3 (210,475 ft^3) and a minimum pool volume of 3,580 m^3 (126,427 ft^3) at low water level.

  The wetwell is designed for an internal pressure of 310 kPa (45 psig) and a temperature of 103.9 °C (219 °F). The design wetwell-to-reactor building negative differential pressure is -13.8 kPa (-2 psig). The suppression pool, which is located inside the wetwell annular region between the cylindrical RPV pedestal wall and the outer wall of the wetwell, is a large body of water that will serve as a heat sink for postulated transients and accidents and as a source of cooling water for the ECCS. In the case of transients that result in a loss of the ultimate heat sink, energy will be transferred to the pool by the discharge piping from the reactor system's SRVs. In the event of a LOCA in the drywell, the drywell atmosphere will be vented to the suppression pool through the system of drywell-to-wetwell vents.

This primary containment design basically uses combined features of the Mark II and Mark III designs, except that the drywell consists of UD and LD volumes. The vents to the suppression pool are a combination of the vertical Mark II and horizontal Mark III systems. The wetwell is similar to a Mark II wetwell.

6.2.1.1 LOCA Chronology

Following a postulated LOCA, the drywell pressure increases as a result of blowdown of the reactor coolant system. Pressurization of the drywell causes the water initially in the vent system to be accelerated into the pool until the vents are cleared of water. During this clearing process, the water leaving the horizontal vents forms jets in the suppression pool and causes water jet impingement loads on the structures within the suppression pool and on the containment wall opposite the vents.

During the vent-clearing transient, the drywell also is subjected to a pressure differential and the RPV pedestal wall experiences a vent-clearing reaction force.

Immediately following vent clearing, an air and steam bubble forms at the exit of the vent. The bubble pressure initially is assumed equal to the existing drywell and wetwell differential pressure. This bubble transmits a pressure wave through the suppression pool water and results in a loading on the suppression pool boundaries and on equipment located in the suppression pool.

As the air and steam flow from the drywell becomes established in the vent system, the initial vent exit bubble expands to equalize the suppression pool hydrostatic pressure. GE's large-scale pressure suppression test facility (PSTF) tests show that the steam fraction of the flow is condensed, but continued injection of drywell air and expansion of the air bubble results in a rise of the suppression pool surface. During the early stages of this process, the pool swells in a bulk mode (i.e., a slug of solid water is accelerated upward by the air pressure). Structures close to the pool surface experience loads as the rising pool surface impacts the lower surface of the structure. In addition to these initial impact loads, these structures experience drag loads as water flows past them. Equipment in the suppression pool also experiences drag loads.

Data from PSTF air tests indicate that after the pool surface has risen approximately 1.6 times the initial submergence of the top vent (which translates to 3.66 m (12 ft) above the initial pool surface for the Mark III design) the thickness of the water ligament could be as small as 0.61 m (2 ft) or less, significantly reducing the impact loads. This phase is referred to as "incipient breakthrough," that is, the ligament begins to break up. To account for possible nonconservatism in the test facility arrangement and instrumentation error bands, the staff has determined that the breakthrough height should be set at 5.5 m (18 ft) above the initial pool surface for the Mark III design. The staff's evaluation of the breakthrough height for the ABWR design is given in Section 6.2.1.6 of this report.

As the drywell air flow through the horizontal vent system decreases and the air/water suppression pool mixture experiences gravity-induced phase separation, pool upward movement stops and the fallback process starts. During this process, floors and other flat structures experience downward loading, and therefore, the containment wall theoretically can be subjected to a small pressure increase. However, this pressure increase has not been observed experimentally.

As the reactor blowdown proceeds, the primary system is depleted of high-energy fluid inventory and the steam flow rate to the vent system decreases. This reduced steam flow rate leads to a reduction in the drywell-to-wetwell pressure differential that, in turn, results in a sequential recovering of the horizontal vents. Suppression pool recovery of a particular vent row occurs when the vent stagnation differential pressure corresponds to the suppression pool hydrostatic pressure at that row of vents.

Toward the end of the reactor blowdown, the top row of vents is capable of condensing the reduced blowdown flow and the two lower rows are totally recovered. As the blowdown steam flow decreases to very low values, the water in the top row of vents starts to oscillate, causing what has become known as vent chugging. This action results in dynamic loads on the top vents and on the RPV pedestal wall opposite the upper row of vents. In addition, an oscillatory pressure loading condition can occur on the drywell and wetwell. Because this phenomenon is dependent on a low steam mass flux (the chugging threshold appears to be in the range of 49 kg/sec/m^2 (10 lb/sec/ft^2)), it is expected to occur for all break sizes. For smaller breaks, it may be the only mode of condensation that the vent system will experience.

The staff's evaluation of these LOCA-related pool dynamic loads is given in Section 6.2.1.6 of this report.

Shortly after onset of a DBA, the ECCS pumps automatically start and pump suppression pool water into the RPV. This water floods the reactor core, and the water starts to cascade into the drywell from the break. When this occurs will depend on the size and location of the break. Because the drywell is full of steam at the time of vessel flooding, the sudden introduction of cool water causes rapid steam condensation and drywell depressurization. When the drywell pressure falls below the wetwell airspace pressure, air from the wetwell redistributes between the drywell and wetwell through the wetwell-to-drywell vacuum relief system. Eventually enough air returns to equalize the drywell and wetwell pressures; however, during this drywell depressurization transient, there is a period of negative pressure on the drywell structure. A negative load condition of -13.8 kPa (-2 psig) is, therefore, specified for drywell design. The staff's evaluation of this drywell-to-wetwell negative differential pressure is given in Section 6.2.1.5.1 of this report.

6.2.1.2 Containment Analysis

The staff's review of the containment design included the temperature and pressure responses of the drywell and wetwell to a spectrum of LOCAs, the capability of the containment to withstand the effects of steam bypass from the drywell directly to the air region of the suppression pool, the capability of the drywell and wetwell to withstand external pressure, and the negative drywell-to-wetwell differential pressure. In addition, the staff considered GE's proposed design bases and criteria for the containment, the analyses and test data in support of the criteria and bases, and the loads resulting from pool dynamic phenomena.

<u>Containment Analytical Model</u>

In its calculation of the short-term and long-term containment pressure-temperature response to postulated high-energy line breaks, GE used the same analytical models and conservative assumptions that were presented and reviewed for the Mark III containment in General Electric Standard Safety Analysis Report (GESSAR) II (NUREG-0979). The staff found these to be acceptable, using independent confirmatory analyses with the CONTEMPT-LT28 computer code. These models and assumptions are discussed in the ABWR SSAR and NEDO-20533 and its Supplement 1, "The G.E. Mark III Pressure Suppression Containment Analytical Model." In response to the staff's request for additional information (RAI), GE stated that the analytical models described in NEDO-20533 are appropriate to calculate the ABWR containment responses to postulated accidents. Although originally written for predicting of Mark III transients, these models, which simulate the transient conditions in the containment, can be adapted for the ABWR containment configuration. These models simulate the drywell, vent systems, and wetwell (suppression pool and airspace). They are, therefore, adaptable to other containment configurations having the same basic components.

As indicated earlier, the ABWR containment design uses combined features of Mark II and Mark III designs, except for the unique feature of two drywell volumes (upper and lower). The vent system is a combination of vertical (Mark II design) and horizontal (Mark III design) drywell-to-wetwell vent systems, and the wetwell (suppression pool and airspace) is similar to a Mark II.

The staff was unable to conclude in the DSER (SECY-91-355) that the assumptions and models used to predict the containment pressure and temperature transients following a LOCA in the ABWR containment were acceptable. In the DSER (SECY-91-355), the staff identified the issue of containment pressure and temperature analysis as Open Items 3 and 10. Specific concerns are discussed below.

As discussed in the DSER (SECY-91-355), GE had not provided a detailed discussion to describe how the two ABWR drywell volumes (Open Item 3) and the combination vertical and horizontal vent system are modeled in the computer code to represent the physical geometry of the containment, and how the air carryover from the two drywell volumes to the wetwell is treated in the computer code. The impact of any difference in the hydrodynamic force, caused by venting, between the Mark III design (vent annulus) and ABWR design (pipe vents) was also unclear. In addition, the staff required tests or analyses to verify that following a LOCA:

- the combination vertical and horizontal drywell-to-wetwell vent system will perform (to demonstrate venting clearing, condensation, and chugging) as predicted

- the containment will perform (air carryover and containment pressure and temperature responses) as predicted by the analytical model

In a meeting held on May 6, 1992, GE clarified its model assumed for drywell volumes and the vent system. The same as for the Mark III containment model, GE assumed that the UD and LD volumes act as a single node and the drywell-to-wetwell vent system as a horizontal vent for its ABWR analysis. The contents of the LD start transferring to the wetwell as soon as the pressure starts decreasing. GE assumed 50 percent of the LD contents is transferred into the wetwell. The staff raised questions about the validity of these assumptions because they do not reflect the actual ABWR configuration. By a letter dated May 22, 1992, GE provided additional information to justify the conservatism of those assumptions. GE calculated the drywell pressure using a more realistic two-node drywell model and compared it with the drywell pressure described in the SSAR using the single-node drywell model. The results show that the single-node model results in a higher peak drywell pressure. Therefore, the single-node model is acceptable in that it is more conservative. This resolved Open Item 3, which was identified in the DSER (SECY-91-355). Furthermore, because the peak containment pressure occurs substantially later, after the vent-clearing process is complete, the slight geometrical differences are expected to result in negligible effects on the containment peak pressure and overall vent-clearing process. GE's use of NEDO-20533 for the ABWR is acceptable.

Regarding the test programs, GE stated that the ABWR test program is described in SSAR Appendix 3B. The test program includes 24 tests of LOCA condensation oscillation and chugging loads. These loads for the ABWR are identified to be potentially different from those for the MARK III design. GE expects other test data that were developed for the Mark III to be applicable for the ABWR because the total area ratio of the horizontal vents and vertical vents in the ABWR will be comparable with that in the Mark III. This was identified as Open Item 3 in the DSER (SECY-91-355), which is resolved. The hydrodynamic loads described in Appendix 3B are evaluated in Section 6.2.1.6 of this report. DSER Open Item 10 is dealt with in that section. The staff concludes that GE's assumptions and models for ABWR containment response analysis are acceptable.

6.2.1.3 Short-Term Pressure Response

The maximum drywell-to-wetwell differential pressure occurs during the blowdown phase (short term) of a LOCA. GE has performed analyses of various postulated primary system breaks, including a double-ended rupture of the main feedwater line, a double-ended rupture of the main steamline, and small-break accidents. The analyses show that the main feedwater line break (FWLB) yields the limiting drywell-to-wetwell differential pressure and peak drywell and wetwell pressure and is, therefore, the design-basis accident for the drywell and wetwell. The main steamline break (MSLB) yields the limiting drywell temperature. GE has provided comparative plots of drywell and wetwell short-term pressure and temperature responses to design-basis, 0.0465 m^2 (0.5 ft^2), 0.093 m^2 (0.1 ft^2), and 0.0093 m^2 (0.01 ft^2) breaks in both the main feedwater and main steamline piping inside the drywell. These figures substantiate the large guillotine breaks resulting in the highest drywell and wetwell pressure and temperature. However, in the DSER (SECY-91-355), the staff stated that these figures, comparing different size pipe breaks, did not indicate the same value of peak drywell and wetwell pressure as reported in SSAR Table 6.2-1. The staff asked GE to clarify these discrepancies.

In a letter dated May 22, 1992, GE compared the reported values in Table 6.2-1 and the corresponding calculated values in SSAR Figures 6.2-6, 6.2-8, 6.2-13, and 6.2-17. It demonstrated the consistency of the reported values between the table and figures in the SSAR. This is acceptable and this item is resolved. It was not tracked in the DSER or DFSER as an open item.

SRP Section 6.2.1.1C states that, for Mark III plants at the construction permit stage, the containment design pressure should provide at least a 15-percent margin above the peak calculated containment pressure and the design differential pressure between the drywell and the containment should provide at least a 30-percent margin above the peak calculated differential pressure. GE's calculated drywell peak pressure for the FWLB is 268 kPa (39 psig) and maximum calculated temperature is 170 °C (338 °F) resulting from the MSLB. The design pressure for the drywell is 310 kPa (45 psig), which provides a margin of 15 percent above the peak calculated pressure in the drywell and is equal to the margin recommended in the SRP. Therefore, this design margin for containment pressure is acceptable.

The calculated wetwell peak pressure and maximum temperature are 179 kPa (26 psig) and 97.2 °C (207 °F), which is 6.7 °C (12 °F) below the design temperature of 103.9 °C (219 °F) resulting from the FWLB. The design pressure for the wetwell is 310 kPa (45 psig), which provides a margin of 42 percent above the peak calculated pressure in the wetwell.

The calculated drywell-to-wetwell peak differential pressure is 110.3 kPa (16 psig) and the design drywell-to-wetwell differential pressure is 172.3 kPa (25 psig), which provides a design margin of 56 percent.

The staff concludes that the containment pressure and temperature transients following a LOCA in the ABWR containment are consistent with SRP Section 6.2.1 and, therefore, are acceptable.

6.2.1.4 Long-Term Response

Following the short-term blowdown phase of the accident, the suppression pool temperature and containment pressure continuously increase because of the input of decay heat and sensible energy into the containment. During this period, the ECCS pumps, which take suction from the suppression pool, reflood the RPV up to the level of the main steam nozzles. Subsequently, ECCS water flows out of the break and fills the drywell, establishing a recirculation flow path for the ECCS. The relatively cold ECCS water condenses the steam in the drywell and rapidly brings the drywell pressure down. After approximately 10 minutes, the residual heat removal (RHR) heat exchangers are automatically activated to remove energy from the containment by using the RHR service water system for recirculation cooling of the suppression pool. This is a conservative assumption because the RHR design permits automatic initiation of containment cooling well before 10 minutes. The containment spray also is conservatively assumed not to be used.

In the long-term analysis, GE accounted for potential postaccident energy sources, including decay heat, pump heat rate, sensible heat, and metal-water reaction energy. GE's long-term model also assumed that the containment atmosphere would be saturated and equal to the suppression pool temperature at any time. Therefore, the containment pressure is equal to the sum of the partial pressure of air and the saturation pressure of water corresponding to the pool temperature.

GE calculated a peak suppression pool temperature of 96.9 °C (206.46 °F). The calculated long-term secondary peak containment drywell and wetwell pressures are well below the calculated short-term peak pressures. GE's analysis for long-term response following a LOCA in the ABWR containment is acceptable in accordance with SRP Section 6.2.1.

6.2.1.5 Reverse Containment Pressurization

Certain events in the primary containment cause depressurization transients that can create negative drywell-to-wetwell, drywell-to-reactor building, or wetwell-to-reactor building pressure differentials. Therefore, vacuum relief provisions may be necessary to limit these negative pressure differentials within design values. The events that cause containment depressurization are

- inadvertent drywell/wetwell spray actuation during normal operation
- post-LOCA drywell depressurization as a result of condensation of the steam by the spilled ECCS subcooled water
- wetwell spray actuation following a stuck-open relief valve

6.2.1.5.1 Drywell Depressurization

Drywell depressurization, which creates a negative drywell-to-wetwell pressure differential and/or a negative drywell-to-reactor building pressure differential, is caused by two major events:

- post-LOCA drywell depressurization as a result of condensation of the steam by the spilled ECCS subcooled water
- inadvertent drywell spray actuation during normal operation

GE states that drywell depressurization following an FWLB results in the most severe negative pressure transient in the drywell. Without vacuum relief, this negative pressure transient may create a drywell-to-wetwell negative pressure differential of -275.8 kPa (-40 psig). This pressure differential is much greater than the design negative drywell-to-wetwell pressure differential of -13.8 kPa (-2 psig). Therefore, this transient is used to determine the size and the number of wetwell-to-drywell vacuum breakers.

On the basis of its analysis, GE further states that with a typical vacuum breaker diameter of 50.8 cm (20 in.), a loss coefficient, K, of 3, and one single failure, eight wetwell-to-drywell vacuum breakers are required to maintain the drywell-to-wetwell and drywell-to-reactor building negative pressure differentials below the design negative pressure differentials of -13.8 kPa (-2 psig).

In the DSER (SECY-91-355), the staff identified Open Item 4 regarding the adequacy of the vacuum breaker design. Specifically, GE needed to clarify the arrangement of vacuum breakers (e.g., LD or UD, two valves in series for bypass single-failure protection) and the test program that would demonstrate the performance of the vacuum breaker system.

In a telephone conversation on August 9, 1991, GE stated that analyses had been performed using first-principle analytical models. These analyses were similar to those performed for other BWRs and assumed that the spray efficiency was 100 percent. Subsequently, in a letter dated May 22, 1992, GE explained that there are eight vacuum breaker valves, including one valve to meet the single-failure criterion.

Each penetration opening into the LD has one valve. These 50.8-cm (20-in) swing check valves open passively on negative differential pressure and require no external power to actuate them. They will be installed horizontally in the wetwell airspace. Position locations of these valves are shown in Figures 1.2-3C and 1.2-13K in SSAR Amendment 6. Also, vacuum breakers are equipped with position switches facilitating the monitoring of valve position inside the control room. The design of the wetwell-drywell vacuum breakers is acceptable.

Concerning the testing aspect of the DSER (SECY-91-355) open item, the ABWR technical specifications (TS) must require the periodic inspection and testing of vacuum breakers during outages to ensure their operability. This was DFSER TS Item 6.2.1.5.1-1. Such testing is included in the BWR Owners Group Standard Technical Specifications, which GE is adopting for the ABWR. Since GE has included an acceptable testing requirement, this item is resolved.

DSER Open Item 4 is resolved because the design of the wetwell-drywell vacuum breakers has been found to be acceptable in accordance with SRP Section 6.2.1, and because the system will be tested as specified in the TS for the ABWR.

In a letter dated May 26, 1994, GE provided the results of an analysis that evaluated the effect of drywell spray actuation following a LOCA. The results of this analysis are bounded by the post-LOCA drywell depressurization case resulting from steam condensation by the spilled ECCS subcooled water.

6.2.1.5.2 Wetwell Depressurization

Wetwell depressurization, which creates a wetwell-to-reactor building negative pressure differential, can be caused by the following events:

- inadvertent drywell and/or wetwell spray actuation during normal operation
- wetwell spray actuation following a stuck-open relief valve
- drywell and wetwell spray actuation following a LOCA

GE states that the limiting negative pressure transient in the wetwell corresponds to wetwell spray actuation following a stuck-open relief valve. The effect of relief valve discharge into the suppression pool also heats the wetwell airspace, thus increasing its pressure. When the pressure in the wetwell becomes greater than the drywell pressure of 1.7 kPa (+0.25 psig), the wetwell-to-drywell vacuum relief system allows the flow of air from the wetwell to the drywell, thereby pressurizing both drywell volumes. Wetwell pressure and temperature peak when the reactor decay heat decreases below the heat removal capability from continued pool cooling and wetwell spray. Wetwell temperature and pressure decrease, but the drywell pressure remains at its peak value. When the pressure difference between the two volumes becomes greater than the hydrostatic head of water above the top vent, air flows back into the wetwell airspace, slowing down wetwell depressurization. The pressure differential between the drywell and the wetwell is maintained constant at the hydrostatic head above the top row of horizontal vents. The final pressure in the wetwell is lower than the drywell pressure because more air is transferred to the drywell during wetwell pressurization than is received during wetwell depressurization.

Inadvertent drywell or wetwell spray actuation during normal operation can cause depressurization of the sprayed volume because of the resultant condensation of vapor present in the airspace. However, the magnitude of this depressurization is less than the post-LOCA or stuck-open relief valve cases because of the relatively smaller mass of condensable vapor present during normal operation.

Calculation of the peak wetwell-to-reactor building negative differential pressure is based on an energy balance of the containment atmosphere before and after spray activation, assuming that the final air-vapor mixture is at 100-percent relative humidity (RH) and that there are no reactor building-to- wetwell vacuum breakers. Using these assumptions, the peak calculated wetwell-to-reactor building negative differential pressure was determined by GE analysis, assuming worse-accident conditions, to be -12.2 kPa (-1.77 psig). This is 10 percent less than the ABWR design value of -13.8 kPa (-2.0 psig). In a meeting with GE on May 6, 1992, the staff questioned the appropriateness of the assumption of 20-percent initial drywell RH in the analysis. In response to the staff's concern, GE performed a sensitivity study assuming a conservative drywell RH of 60 percent and a minimum suppression pool temperature of 23.9 °C (75 °F) for the postulated drywell and wetwell spray actuation during normal operation. The resulting maximum negative differential pressure shown in a letter dated June 10, 1992, is -10.2 kPa (-1.48 psi), which is less limiting than the case reported in the SSAR. The limiting case of wetwell spray actuation following a stuck-open relief valve is independent of the drywell humidity.

From its analysis given in the SSAR, GE concluded that vacuum relief from the reactor building to the wetwell was not required. The staff asked GE in what way was the design of the ABWR different from the designs of Mark II and III plants, which do require vacuum relief. GE stated that the wetwell spray system flow will be limited by design to a maximum flow of 1,893 L/min (500 gpm), which has the effect of limiting the negative pressure transient response to -12.2 kPa (-1.77 psig). Therefore, the staff concludes that the limiting case reported in the SSAR is acceptable. The staff also finds that the initial conditions, assumptions, and methodology used in the GE analysis are acceptable, as discussed above, and that vacuum relief from the reactor building to the wetwell is not needed.

6.2.1.6 Suppression Pool Dynamic Loads

GE submitted proprietary SSAR Appendix 3B to address the issue of suppression pool dynamic loads for the ABWR. The scope of Appendix 3B encompasses SRV actuation and LOCA phenomena, SRV discharge loads, LOCA loads, submerged structure loads, and load combinations. Although similar to the Mark III containment design, the ABWR has the following distinctive features that affect suppression pool dynamic loads: wetwell airspace pressurization, an LD volume, a smaller number of horizontal vents (30 in the ABWR compared with 120 in the Mark III), horizontal vent extension into the pool, vent submergence, and suppression pool width.

SRV actuation and LOCAs are the events that can impose dynamic loads on the suppression pool. SRVs discharge steam from the RPV through discharge piping that is routed into the suppression pool and fitted at the suppression pool end with a quencher to enhance heat transfer between the hotter SRV discharge fluid (steam and air) and the cooler suppression pool water.

SRV discharge into the suppression pool consists of the following three phases, which are listed in the order in which they occur: water clearing, air clearing, and steam flow. The discharge pipe standing column of water first is pushed out, or cleared, into the pool by blowdown steam pressure. Water clearing creates SRV pipe pressure and thermal loads, pipe reaction forces, drag loads on structures submerged in the pool, and pool boundary loads. After water clearing, air clearing occurs as air above the water column in the pipe is forced out the pipe and into the pool. The air-clearing phase generates expanding bubbles in the pool that cause transient drag loads on submerged structures as a result of both the velocity and acceleration fields and oscillating pressure loads on the pool boundary. Finally, the steam-flow phase creates pipe reaction forces, quencher thrust forces, structure thermal loads, and oscillating pool boundary loads as a result of steam jet condensation at the quencher.

For the ABWR, the FWLB and MSLB cause dynamic loads in the suppression pool. As with the SRV discharge, these events can be characterized by several phenomena that occur in the following order: vent clearing, pool swell, high steam flow, and chugging. After an FWLB or MSLB, with sufficient pressurization of the drywell, water in the vents is forced out into the pool. This vent water clearing causes submerged jet-induced loads on nearby structures and the pool basemat. After vent clearing, an air and steam bubble flows out the vents. The air component, originating from the drywell air, expands in the pool causing a rise in pool surface level called pool swell. Pool swell imposes loads on submerged structures and pool boundaries. After pool swell, a period of high steam flow occurs and steam is condensed in the pool vent exit area, imposing no significant loads on the pool system. Later, as vent steam flow decreases, the steam condensation process causes the steam bubble, which has been growing, to suddenly collapse, creating oscillatory loads. This process is called chugging and imposes significant vent and suppression pool boundary loads.

The ABWR SRV discharge is directed to the suppression pool through X-quenchers that GE has stated are identical to the quenchers used for the Mark II and Mark III designs. Therefore, GE concluded that the hydrodynamic load methodology, developed for the Mark II and III designs, was applicable for both the ABWR suppression pool geometry and the X-quencher configuration. However, in the DSER (SECY-91-355), the staff questioned how GE had addressed the SRV loads that would result from a second opening of the SRV while the SRV tailpipe is still hot from the initial SRV discharge (commonly referred to by the staff as "subsequent actuation" or "consecutive actuation" in NUREG-0802, "Safety/Relief Valve Quencher Loads: Evaluation, for BWR Mark II and III Containments") (Open Item 5). A subsequent SRV valve actuation becomes a concern because following the first actuation of the SRV, the second actuation could generate higher loads on the structure.

The water reentering the tailpipe (reflood) after the initial actuation of the SRV has been found in experiments to be transitory because the water column within the tailpipe does not reach equilibrium quickly. In addition, the tailpipe wall has not cooled to its initial temperature before the second actuation of the SRV. If the reactor system pressure should rise again to greater than the SRV set point, the SRV would discharge. However, the noncondensable gas from the drywell atmosphere that reentered the tailpipe through the vacuum breaker would have been heated by the tailpipe wall. This discharge (commonly called SRV air-clearing loads) could produce hydrodynamic wall pressures in the pool that might be significantly different from the initial air-clearing loads because of the higher noncondensable gas temperature in the tailpipe. As a result, the staff concluded that loads from both the initial and second actuation should be considered in the design of the system. GE stated in its submittal of June 1, 1992, that both the first and possible second actuation were considered and the structures will be analyzed to accommodate these second actuation loads. GE stated that these analyses are scheduled to be completed before final design approval. This item became DFSER Open Item 6.2.1.6-1.

GE stated in SSAR Section 3.9.3.3.1 that loading of the main steam and discharge piping resulting from SRV discharge was calculated on the basis of simultaneous actuation of all SRVs followed by a second actuation of all SRVs. The methodology used by GE to calculate hydrodynamic loading on SRV discharge piping resulting from initial and subsequent SRV actuations is consistent with the methodology used for earlier BWR (Mark II and III) designs. The effect of subsequent valve actuation is considered by assuming the SRV discharge pipe has been heated by the initial SRV actuation. GE assumed an initial SRV pipe temperature of 149 °C (300 °F) for the drywell piping and 93 °C (199 °F) for the wetwell piping. These temperature values are based on measured data from in-plant SRV blowdown tests. This methodology, which the staff finds acceptable, is consistent with that used for earlier BWRs and resolved DFSER Open Item 6.2.1.6-1.

The staff also questioned, in the DSER (SECY-91-355), the acceptability of the complete elimination of suppression pool temperature limits. GE had stated in its earlier submittal that suppression pool temperature limits for steady-state steam condensation were no longer needed. In the basis for this conclusion, GE referenced an analysis submitted to the staff by the BWR Owners Group. However, the staff had not yet completed its review of the BWR Owners Group request for the elimination of suppression pool temperature limits. This was Open Item 6 of the DSER (SECY-91-355). In addition, the date for completing the staff's review was shown to be very close to the date necessary for final closure of all issues for the ABWR. In light of these uncertainties, GE documented in its submittal of June 1, 1992, that the same suppression pool temperature limits will be used for the ABWR as those used in current Mark I, II, and III plants to ensure steady condensation of the SRV discharge. These criteria are documented in NUREG-0783, "Suppression Pool Temperature Limits for BWR Containments."

As currently implemented in the Mark I, II, and III designs, the suppression pool temperature limits involve a three-tier approach. The lowest temperature threshold requires the operator to take such actions as activating pool cooling to reduce the suppression pool temperature. The plant, however, can continue to operate at power during this time. The intent of this threshold is to ensure that the operator takes action to reduce pool temperature. This temperature is typically 35 °C (95 °F). Operation can continue until the suppression pool reaches 43 °C (110 °F). At this temperature, an automatic scram on high suppression pool temperature occurs. Finally, if the pool reaches 49 °C (120 °F), the TS require depressurization of the reactor coolant system and initiation of cold shutdown conditions. This was DFSER TS Item 6.2.1.6-1. GE included this item in its TS, which the staff has found are acceptable in Chapter 16 of this report, thus resolving this TS item.

This process ensures that suppression pool temperature limits for reactor scram and reactor depressurization, as defined in NUREG-0783, will not be reached. The staff finds acceptable GE's commitment to maintain current suppression pool temperature limits. This part of Open Item 6 in the DSER (SECY-91-355) is resolved.

Another part of Open Item 6 from the DSER (SECY-91-355) was associated with the justification used to establish the pool dynamic loads for the ABWR. The ABWR suppression pool contains structures above and below the normal level of the pool (such as SRV tailpipes, access tunnels to the LD, and a walkway in the suppression pool) that would be subject to pool drag and pool swell impact loads during the initial vent clearing and pool swell phenomena. The staff requested that GE provide the specific tests used to support the methodology to establish the hydrodynamic loads for the ABWR. GE committed to provide these tests in the SSAR. GE had identified the general database that will be used and the methodology to develop the hydrodynamic loads, but had not submitted data on the actual primary or secondary loads (wall pressure, thermal, drag, impact, etc.). This methodology for defining hydrodynamic loads was considered acceptable in the DFSER. This resolved both parts of Open Item 6 identified in the DSER (SECY-91-355). This resulted in DFSER Confirmatory Item 6.2.1.6-1 - GE would provide the specific load definition design information in a future SSAR amendment.

GE provided a typical pressure time history at the bottom of the pool boundary in SSAR Figure 3B-29. The staff finds this plot to be an acceptable hydrodynamic load definition for design certification. This resolved DFSER Confirmatory Item 6.2.1.6-1.

The ABWR configuration is similar to the Mark II containment design in that the suppression pool volume of the ABWR wetwell is approximately equal to the Mark II and III suppression pool volumes. Also, the sizes of the drywell volumes are quite similar. GE stated in its submittal of June 1, 1992, that the ABWR wetwell airspace volume is 5,947 m^3 (210,000 ft^3) as compared with a Mark II wetwell volume of 4,672 m^3 (165,000 ft^3). Therefore, it considered the methodology used to evaluate the bulk backpressure due to suppression pool gas space compression from pool swell for the Mark II design to be equally applicable for the ABWR.

The PICSM computer code, which models the transient behavior of suppression pool swell surface elevation, pressure in the wetwell airspace, and pool surface velocity, is described in GE technical report NEDE-21544 (proprietary). GE validated test data generated for the Mark II design; however, the code was not reviewed and approved by the staff.

The staff identified Open Item 10 in the DSER (SECY-91-355), related to modeling assumptions used by GE. For ABWR applications, the PICSM code was used to compare Mark III suppression pool swell test data from the pressure suppression test facility (PSTF) with analytical predictions from PICSM. The parameters that were compared were gas backpressure, water slug velocity, and water slug swell height. GE concluded that the PICSM code was technically adequate to model the suppression pool gas space pressure; however, the calculated water slug surface elevation required some modeling adjustments because of the horizontal vent configuration of the Mark III. A horizontal vent configuration introduces an air bubble into the suppression pool that does not spread uniformly across the entire suppression pool; therefore, it produces a swell that is not radially uniform. The pool swell has a higher rise on the inside radius of the pool.

The Mark III vent design is quite similar to that of the ABWR. Because of this similarity, GE felt that comparison with the Mark III data would be sufficient to validate the program for use on the ABWR design. This type of comparison also was necessary to determine if the difference in vent design (such as submergence of the vent and number of vents) would significantly affect the calculated results using the PICSM code.

Because of the horizontal vents for both the ABWR and the Mark III, the air bubble does not penetrate the entire width of the pool as was described above. As a result, the suppression pool water slug is not a constant thickness. The PICSM code does not implicitly model this nonuniform pool swell elevation. This nonuniform pool swell is different from the Mark II design in which the air bubble is uniformly distributed over the entire suppression pool. The vertical Mark II downcomers then distribute the air bubble throughout the entire pool cross-section. As a result, the air is injected into the pool in an almost uniform manner yielding an almost constant water slug thickness.

Because of the difference in vent configuration between the Mark II (which the PICSM code would model accurately) and the Mark III, GE found it necessary to develop a correlation for the PICSM code to account for the uneven pool slug rise observed in the Mark III PSTF tests. GE stated that by modeling 80 percent of the suppression pool horizontal surface area, it achieved agreement between PICSM calculated pool swell velocity and elevation and the PSTF test results. GE included this comparison in its submittal of June 1, 1992.

GE showed that the PICSM code without the correction factor can correctly predict the suppression pool airspace pressure time histories resulting from a reactor system blowdown in the drywell. By using a reduction factor of 80 percent for pool surface area, GE correctly predicted the swell elevation. On the basis of these comparisons, GE believes that the PICSM code has been validated for use on the ABWR.

GE also noted that the pool swell phenomenon is dependent on a reactor system blowdown into the drywell and that the largest postulated pipe break in the ABWR is a feedwater line break (FWLB). The Mark II and III designs must accommodate a recirculation line break that GE has eliminated in the design of the ABWR. A postulated FWLB into the ABWR drywell is calculated to be a lower pressurization event, causing less blowdown flow, because of the smaller pipe break size and energy input than was considered for Mark II and III designs. Because of the lower mass and energy input into the drywell from an FWLB, the suppression pool response is expected to be less severe.

GE also stated that the pool-to-vent area ratio is larger for the ABWR than for the Mark II and III designs. For the ABWR, the pool-to-vent area ratio is 38.5, for Mark II the ratio is typically 20.0, and for Mark III it is typically 12.0. GE believes that the larger pool relative to the vent area will cause the pool hydrodynamic loads to be reduced. This position is supported by NUREG-0808, "Mark II LOCA-Related Hydrodynamic Load Definition."

The staff documented its evaluation of definition of the Mark II design containment hydrodynamic load in NUREG-0808. In the evaluation of the pool swell phenomena (discussed in Section 2.1 of the NUREG report), the staff relied on comparisons with a substantial amount of data from tests conducted by both GE and Japan Atomic Energy Research Institute. These tests were directly applicable to the Mark II design. The computer code used for these comparisons was a GE-developed program called PSAM (GE topical report NEDO-21061, Revision 0, November 1975). It was used as part of the Mark II hydrodynamic load evaluation program. The staff has reviewed the Mark II program and approved the methodology and PSAM in NUREG-0808. However, it did not find GE's methodology within PSAM acceptable. Rather, the staff based its acceptance on the favorable comparisons with the database. As a result, the use of the program for configurations other than those encompassed by the test data would not be accepted without further comparisons with applicable test data.

The PICSM code, which GE referenced in its submittal of June 1, 1992, had not been reviewed by the staff. In addition, the use of the 80-percent area reduction factor may significantly alter the results of the program. GE's justification for use of an 80-percent reduction on pool surface area to achieve correct swell heights is based on the comparison with PSTF test heights for a Mark III-type suppression pool. However, GE did not discuss other such critical parameters of pool swell as water slug velocity and the way velocity is affected by the 80-percent reduction factor. In addition, the 80-percent reduction factor also may affect such factors as the effect of gas backpressure on the water slug because of the reduced water slug surface area exposed to the gas backpressure and inertia of the water slug may be affected by the reduced surface area. GE did not discuss these effects or the PICSM code's treatment of them in its submittal. PICSM is a newer version of the PSAM computer code; however, GE has not indicated the specific differences between the codes.

The staff agrees that the ABWR suppression pool design is similar to the Mark III design in terms of the water slug shape and that PSTF test data for Mark III may have limited applicability to the ABWR. This closes out Open Item 18 identified in the DSER (SECY-91-355). However, since the database discussed in the submittal of June 1, 1992, did not contain specific test results for an ABWR geometry (actual vent size and configuration were not used for ABWR-specific tests), the staff evaluated the uncertainties of the test results and whether these uncertainties can be tolerated in the ABWR design.

As a result, the staff evaluated the ABWR design and the PICSM load predictions to determine the degree of margin associated with each safety-related component. At a height of 7 m (23 ft) above the normal pool surface level, there are eight drywell-to-wetwell vacuum breakers that are mounted on the suppression pool wall and potentially subject to pool swell loads. The design indicates 7 m (23 ft) from the normal liquid level in the suppression pool to the bottom of the valve. The vents are equally spaced around the circumference of the suppression pool. Underneath the vacuum breakers, a continuous catwalk surrounds the pedestal wall within the suppression pool at the 6-m (20-ft) elevation, which is approximately 1 m (3 ft) below the vacuum breakers. The flooring is grating except for the area immediately below each vacuum breaker. GE states that a solid steel plate in this area will protect the valves from any direct effect of the pool swell water slug.

PICSM predicts liquid impact up to 6 m (21 ft) and froth impact beyond 6 m (21 ft). As a result, the design is based on the solid decking of the catwalk absorbing 100 percent of any possible liquid pool swell load. Additionally, the valves are designed to withstand froth loads assuming no load reduction as a result of the steel plate. This approach seems to be bounding in nature, and the loadings assessed for the valves are acceptable. However, GE has not specified the method for calculating the load to which the steel plate is designed or the specific load to which the plate will be subjected. This was DSER (SECY-91-355) Open Item 6.2.1.6-2, which was not tracked in the DFSER. The staff also stated in the DFSER that Tier 1 design information and inspections, tests, analyses, and acceptance criteria (ITAAC) were required for pool swell impact loads. (The ABWR ITAAC are discussed in Chapter 14 of this report.) The staff has determined that the design of the plate and supporting analysis are not needed for design certification but can be provided in the COL application, because of GE's commitment to have the analysis performed in accordance with the Mark II and III design methodology.

As specified in SSAR Section 6.2.7.4, the COL applicant will review the issue of providing appropriate structural features for protecting these vacuum breakers from pool swell loads and propose to the NRC staff an appropriate design for ensuring that these valves are adequately protected. The structural shielding features for pool swell loads will be determined on the basis of the methodology approved for the Mark II and III designs. For design of structural shielding features, pool swell loads will be defined to the maximum practical extent possible. This commitment to a COL action item is acceptable, and resolved DSER Open Item 6.2.1.6-2.

In the DSER (SECY-91-355), the staff raised an issue concerning the modeling assumptions used for calculating suppression pool boundary loads during the pool swell period occurring as a result of a postulated hydrodynamic event caused by a LOCA. The question was raised concerning the modeling assumptions used for conducting the tests to determine the resultant pool loads.

GE stated in its submittal of June 1, 1992, that it conducted a test program for suppression pool hydrodynamic loads specific to the ABWR because of anticipated differences in condensation oscillation (CO) and chugging (CH) loads in the ABWR design when compared to the Mark III design. GE anticipated differences in loads because the airspace (5,947 m^3 (210,000 ft^3)) in the ABWR suppression pool is much smaller than that in the Mark III design (4,672 m^3 (1,140,000 ft^3)). This smaller wetwell airspace volume increases the pool backpressure and may dampen pool response. Also, the LD in the ABWR may have an effect in that the total drywell volume of the ABWR is 7,334 m^3 (259,000 ft^3) compared to 7,787 m^3 (275,000 ft^3) for Mark III. However, the LD is a physically separate volume from the UD and is connected to the UD through the vent duct system compared with Mark III, where the drywell is one large volume. The ABWR also has a reduced number of horizontal vents (30) at a deeper submergence of 3.5 m (11.48 ft) from the top vent relative to the Mark III, which has 120 vents at a submergence of 2 m (7 ft). The suppression pool for the ABWR is also wider (7.5 m (24.6 ft)) than that in the Mark III (6.2 m (20.5 ft)).

The staff identified Open Item 9 in the DSER (SECY-91-355) concerning the scaling laws used by GE for developing the ABWR load definition. GE conducted 24 simulated blowdown tests on a single-cell model (a 36-degree sector of the ABWR suppression pool) that represented a single vent pipe system. The test facility used was scaled for two sizes of blowdown experiments. The subscale (SS) tests represented the ABWR scaled by a reduction factor of 2.5 and a scaled single vertical and single horizontal vent pipe system. The partial full-scale (FS) tests were conducted with the same scale factor for pool dimensions but a full-scale vertical and horizontal vent system and two horizontal vent pipes into the pool. The ABWR has three horizontal vent pipes into the suppression pool.

Using the SS and FS configuration in its test facility, GE ran 24 blowdown tests to obtain data on CO and CH loads to produce wall pressure data to be used to generate the load definition. Thirteen tests were conducted for CO loads on the SS test configuration and eleven were conducted for CH loads on the FS test configuration. For the SS tests, GE stated that linear dimensions were scaled, but thermodynamic properties associated with steam condensation such as pressure, temperature, and enthalpy were maintained at full-scale values.

The staff evaluated the above approach and concludes that these additional tests have added to the overall database relative to pool condensation loads and are acceptable. This resolved Open Items 7, 8, and 9, as identified in the DSER (SECY-91-355). However, GE had not addressed the use in the ABWR design of Mark III data from the PSTF blowdown tests that were conducted to verify the GE methodology used for the Mark III load definition studies. These tests were conducted and demonstrated the use of scaled tests, as reported in NUREG-0978, which showed close agreement between 1/3 and 1/9 scaled tests. The PSTF tests for Mark III on all scaled tests reported in NUREG-0978 were conducted with full-scale vent lengths and all three horizontal vents.

GE also had not demonstrated that the past condensation tests can or should be neglected in the development of the load definition for the ABWR. In addition, the staff was concerned about the adequacy of using only the SS and FS tests for which one and two horizontal vent pipes into the pool were used instead of the three vent pipes used in the ABWR and the PSTF tests for Mark III.

For previous tests conducted for Mark III at the PSTF, three horizontal vent pipes that were full scale were used. For the ABWR SS tests, the vent system was scaled, which the staff believes will affect the measured frequency spectrum. The use of scaled frequency responses for CO has not been demonstrated, nor has there been any attempt to justify the approach on the ABWR docket. The Mark III tests for unstable CO were conducted on a full-size vent system, which would not interfere with the frequency content of the measured wall pressures in the test facility.

Therefore, to resolve DSER Open Item 10, the staff concluded that GE should address the differences in load definition using the Mark III and II test databases and determine the effect on structural response of possible differences from frequency signatures for both CO and CH loads. The use of the ABWR SS and FS tests without correlation to the Mark III test database had not been demonstrated to be sufficient for modeling without the full-scale vent pipe configuration and was unacceptable. This was DFSER Open Item 6.2.1.6-3.

SSAR Figure 3B-22 shows a pressure time history representative of CO loads associated with the ABWR using the source load approach. SSAR Figure 3B-23 shows a pressure time history representative of Mark III CO loads. This pressure time history is based on the Mark III CO correlation as described in the GESSAR II (NUREG-0979). This comparison showed higher pressure amplitudes in the ABWR that were attributed to (1) a greater submergence depth (3.5 m (11.5 ft) compared with 2.3 m (7.5 ft)), (2) all three horizontal vents remaining open in the ABWR tests whereas the bottom two vents were closed in the Mark III tests at the onset of CO conditions, and (3) the contribution from increasing wetwell overpressure during the CO period in the ABWR tests. ABWR CO loads were not compared to Mark II test data because Mark II has a vertical vent while the ABWR has a horizontal vent. GE stated that a typical large chug from the full-scale database as shown in Figure 3B-24 exhibited characteristic features similar to chugging data from Mark II and Mark III testing. For the ABWR, GE defined the chugging load by using the key-chug approach, which was the same approach as that used for Mark II. GE also stated that the data supported the understanding (observed from prior tests) that chugging has some dependence on system parameters, such as mass flux and pool temperature, along with a substantial degree of randomness.

The staff believes that GE has adequately considered the differences between the Mark II, Mark III, and ABWR databases to determine that the suppression pool wall pressures for ABWR do not exhibit any unusual characteristics when compared to the Mark III wall pressures. The SS and FS tests appear to be adequate representations of the ABWR downcomer vents for predicting the suppression pool hydrodynamic response for unstable CO and CH loads. The staff finds that the proposed load definition methodology for unstable CO and CH loads acceptable. This resolved DFSER Open Item 6.2.1.6-3.

6.2.1.7 Subcompartment Pressure Analysis

Internal structures in the drywell, wetwell, and secondary containment form subcompartments or restricted volumes that are subjected to differential pressure after postulated pipe ruptures. In the drywell there are two such volumes: (1) the RPV annulus, which is the annular region formed by the RPV and the reactor shield wall, and (2) the drywell head, which is a cavity surrounding the RPV head.

The design of the containment subcompartments was based on the postulated worst-case design-basis accident (DBA) occurring in each subcompartment. For each containment subcompartment in which high-energy lines will be routed, mass and energy release data corresponding to a postulated line break were calculated. All breaks were considered to be full double-ended breaks.

In response to RAI Question (Q)430.17 regarding subcompartment pressurization from high-energy line breaks, GE submitted SSAR Tables 6.2-3 and 6.2-4 and Figures 6.2-37a through 6.2-37e. These tables and figures present subcompartment node and vent path initial conditions, break conditions, and physical characteristics as well as a flow chart showing the volume and junction connections between each subcompartment. GE modeled a total of 23 subcompartments connected with 35 separate flow path vents for the subcompartment analysis. Most of the vents are blowout panels that have a characteristic opening pressure and time. The subcompartments enclose some compartments of the RHR, ECCS, reactor water cleanup (RWCU), main steam, and main turbine systems. GE presented the calculated peak differential pressure for each subcompartment in SSAR Table 6.2-3.

The staff evaluated the aforementioned information in accordance with the requirements and guidance in RG 1.70 (Rev. 3) and SRP Section 6.2.1.2. The staff requested in Open Item 11 in the DSER (SECY-91-355) the following additional information:

- mass and energy release rates assumed for the subcompartment analyses
- methodology (i.e., computer codes), if any, used in calculating subcompartment pressurization
- nodalization sensitivity studies for the individual subcompartments to justify the final model
- basis for selecting subcompartment initial thermodynamic conditions
- individual subcompartment <u>design</u> pressure differential

Within the limitations of the available information, the staff made the following additional observations in Open Item 12 in the DSER (SECY-91-355).

- The selected subcompartment initial humidity specified in SRP Section 6.2.1.2, Item II.B.1, is 0 percent. Because of the capability of water vapor to absorb more energy than dry air, a humidity level of 0 percent results in a maximum peak differential pressure during a high-energy line break in a subcompartment. In its analysis, GE used a higher value for initial humidity.
- On the basis of subcompartment volume and relief vent properties, the trend of calculated peak differential pressure for rooms with the same pipe break was analyzed. A number of calculated subcompartment peak pressures did not follow the basic trend that was expected (that is, for the same pipe break, peak pressure should increase with smaller room volume and/or smaller vent area). The subcompartments with questionable peak pressures were SA7, SA4, SR5, SR4, and SR9.
- Using the COMPARE MODE 1A computer code, subcompartment and vent properties from SSAR Tables 6.2-3 and 6.2-4 and main steamline break mass and energy release data from SSAR Figures 6.2-24 and 6.2-25, the staff performed a review calculation for the pressurization of rooms SS1 and ST1 (steam tunnel and turbine building). This analysis showed pressures for the steam tunnel and for the turbine building that were significantly different from those reported in the SSAR.

These observations showed inconsistencies in subcompartment peak pressure trends, subcompartment pressures, and analytical assumptions. These differences, when considered collectively, may result in a less conservative structural design of containment subcompartments. The staff identified the above concerns as Open Items 11 and 12 in the DSER (SECY-91-355).

In a letter dated May 22, 1992, GE stated that its proprietary computer code, SubCompartment Analysis Method (SCAM), was used for the subcompartment analyses. GE stated that the SCAM code was used for the Mark III standard plant and was benchmarked against 13 NRC-specified subcompartment standard problems. GE has not submitted SCAM to the NRC for computer code review and approval. SCAM calculates the transient thermal-hydraulic response of connected subcompartments to high-energy line breaks. Each volume is modeled as a homogenous mixture with connecting flow paths using a self-choking compressible flow model and the volume pressure and temperature calculated for each time step. The vent flow model was extensively verified by experimental comparison in GE report NEDO-20533. SCAM treats air as an ideal gas and initializes the contents of each volume as a homogeneous mixture of air and water vapor. SCAM does not include gravitational potential energy in its flow energy equation and does not include the effects of heat transfer between materials in the volumes and the flowing fluid.

By a letter dated May 11, 1992, GE provided additional information on mass and energy release rates assumed for its subcompartment analyses. GE calculated the mass and energy release rate for main steam and RWCU line breaks by including friction losses in the pipe segment between the RPV and the postulated break location as well as flow choking (i.e., critical flow) and inventory depletion effects. Isolation valve closure determines the time for cessation of break flow. Friction losses included pipe length, fittings, elbows, valves, and other equipment with a hydraulic resistance in the pipe segment. This methodology resulted in different mass and energy release rates being calculated for the same pipe type and size break in different subcompartments. GE indicated in a telephone conversation on May 19, 1992, that the inconsistency identified in Open Item 12 of the DSER (SECY-91-153), may be explained by the use of these mass and energy release data. The pipe break mass and energy release data used by GE are based on estimated pipe lengths, fittings, elbows, valves, and other hydraulic resistance components in the pipeline. This assumed hydraulic resistance (i.e., fl/D) constitutes part of the design basis for the ABWR and must be confirmed and adhered to in the actual as-built plant, since deviations in such parameters as pipe routing and numbers and types of valves could affect the results of the subcompartment analysis. The staff indicated in the DFSER that this information should be included in the ITAAC for the ABWR containment. This was identified as a part of DFSER Open Item 6.2.1.7-1. On further evaluation, the staff revised its previous position and determined that the information on fl/D should be included in the SSAR rather than the ITAAC. In Amendment 30, GE provided this information (fl/D) in Table 6.2-4A and mass and energy release data in Table 6.2.4B of the SSAR. This is acceptable. This part of DFSER Open Item 6.2.1.7-1 is resolved. In addition, the staff and GE agreed that subcompartment analyses using as-built fl/D data and effects on the mass and energy releases should be included in the ITAAC. This is discussed below.

The GE SCAM subcompartment model for the ABWR consisted of a total of 23 volumes and 35 flow junctions. Most of the flow junctions are blowout panels that have a blowout pressure of 3.45 kPa (0.5 psig). Three different types of double-ended pipe breaks were analyzed in the subcompartment analysis: 71.1 cm (28 in.) main steam, 15.2 cm (6 in.) main steam, and 20.3 cm (8 in.) RWCU. All blowdowns and corresponding peak subcompartment pressures occurred over a period of less than 1 minute. All subcompartments were initialized at maximum expected air temperature and pressure and a humidity of 10 percent. Junctions were modeled with their flow area, forward and reverse loss coefficient and associated inertia based on flow length and area. The SCAM analyses did not include any nodalization sensitivity studies and relied on internal time step selection routines.

As part of the review of the ABWR subcompartment analyses, the staff performed independent audit calculations with the COMPARE MOD1A computer code, which is a subcompartment analysis code developed by Los Alamos National Laboratory for the NRC and approved for this application in SRP Section 6.2.1.2 (Item III). The staff developed a model of the ABWR subcompartment volumes, junction vents, and mass and energy release data based on the information provided by GE. Five of the pipe breaks were analyzed using COMPARE. These five pipe breaks resulted in peak pressures for 8 of the 23 subcompartments in the ABWR. The results of the COMPARE calculations of peak differential pressure were similar in magnitude and trend to those reported by GE. In all cases, the GE-calculated peak differential pressure was larger than that calculated by COMPARE. This is attributed to differences in modeling blowout panels and vent flow correlations between SCAM and COMPARE. However, the SCAM results were all bounding. Based on the COMPARE audit calculations, the staff confirmed in the DFSER that the results of the GE subcompartment pressure analyses were acceptable.

Review of the ABWR subcompartment analysis shows that the analysis was performed in accordance with SRP Section 6.2.1.2. Independent check calculations with the COMPARE MOD1A computer code confirmed both the trends and magnitude of the peak subcompartment differential pressures reported by GE in the SSAR.

Although SRP Section 6.2.1.2 states that zero initial humidity should be assumed for volumes, GE assumed a 10-percent humidity. GE stated that the subcompartment initial thermodynamic conditions were those corresponding to plant normal operating conditions, as defined in SSAR Appendix 3I. The staff's COMPARE analyses confirm that the effect of the 10-percent initial humidity is insignificant.

Item II.B.2 of SRP Section 6.2.1.2 requires that an appropriate nodalization scheme be chosen and confirmed by sensitivity studies so that there is no substantial pressure gradient within a node. In a meeting of May 6, 1992, GE stated that on the basis of its judgment of adequacy, no specific nodalization sensitivity studies were performed.

Furthermore, in a conference call on May 19, 1992, and in a letter dated May 22, 1992, GE committed to revise the SSAR to include the values of design differential pressure for each subcompartment in order to confirm that the margin between calculated and design peak differential pressure of 40 percent required by SRP Section 6.2.1.2 (Item II.B.5) is achieved for the ABWR. The staff found this 40-percent design margin acceptable subject to the SSAR revision to reflect this commitment. This was identified as Confirmatory Item 6.2.1.7-1 in the DFSER. In Amendment 30, GE provided this information in SSAR Table 6.2-3. There are a few exceptions in which the 40-percent margin cannot be satisfied. However, there still exist substantial margins such as 28 percent, 34 percent, or 38 percent shown in Table 6.2-3. The staff has determined that these few exceptions are acceptable provided the subcompartment analyses using as-built data are included in the ITAAC. Therefore, the staff concludes that DFSER Confirmatory Item 6.2.1.7-1 is resolved. In addition, DSER Open Items 11 and 12 (SECY-91-153) were resolved in the DFSER.

Following the issuance of the DSER (SECY-91-355), GE modified the reactor shield wall, extending it to a height of 0.1 m (0.33 ft) below the containment top slab. This results in a smaller vent area for a pipe break in the shield annulus. The staff requested GE to confirm the adequacy of the revised shield wall design. In Amendment 30, GE states that the results of its subcompartment pressurization re-analysis, accounting for the reduced vent area and redefined DBA break in the annulus, demonstrate the adequacy of the shield wall structure and the RPV and its internal structures. Therefore, this new issue of annulus pressurization resulting from the modification of the shield wall is resolved.

Therefore, on the basis of the above discussion, the staff concludes that GE's subcompartment analysis is acceptable.

GE submitted the design description and ITAAC for the ABWR containment. The ITAAC were being reviewed by the staff when the DFSER was issued. This was identified as DFSER Open Item 6.2.1.7-1. GE subsequently submitted a revised design description and ITAAC. In ITAAC Section 2.14.1, "Primary Containment System," GE has included containment pressure and temperature analyses of the DBA using as-built data. In addition, the verification of the subcompartment analysis of the reactor building is included in ITAAC Section 2.15.10, "Reactor Building," as part of the structural analysis reconciling as-built data, which is described in Section 3H.5.3 of the SSAR. The staff finds the above commitments regarding subcompartment analyses acceptable. Therefore, DFSER Open Item 6.2.1.7-1 is resolved. The adequacy and acceptability of the design description and the ITAAC are evaluated in Section 14.3 of this report.

6.2.1.8 Steam Bypass of the Suppression Pool

The staff evaluated GE's analysis of steam bypass of the suppression pool provided in SSAR Section 6.2.1.1.5 in accordance with SRP Section 6.2.1.1.C. Containment pressurization from steam bypass of the suppression pool is strongly affected by the containment size and wetwell-to-drywell volume ratio. Of the Mark I, II, and III containments, the ABWR containment design is most similar to the Mark II in size and wetwell-to-drywell volume ratio. Therefore, the staff concludes that the acceptance criteria in SRP Section 6.2.1.1.C for Mark II containments are the most applicable. These acceptance criteria specify (1) an effective steam bypass capability for small breaks ($A/K^{1/2}$) of 46.5 cm^2 (0.05 ft^2) (2) automatic initiation of wetwell sprays, (3) preoperational high-pressure leak test, (4) periodic low-pressure leak tests, (5) periodic visual inspection of the vacuum relief system, (6) redundant position indicators and alarms on all vacuum breakers, and (7) monthly operability test of vacuum breakers.

In a pressure suppression-type containment, such as the ABWR, steam released from the primary system following a postulated LOCA is collected in the drywell volume and directed through the drywell connecting vents to the suppression pool where it is condensed. Therefore, no steam is supposed to enter the wetwell air volume. However, the potential exists for steam to bypass the suppression pool by leakage through the vacuum breakers or directly from leak paths in the drywell-to-wetwell boundary. Such bypass could lead to undesirable pressurization of the containment, as the pressure suppression function has been lost. BWRs have the capability to accommodate limited amounts of suppression pool bypass. An allowable amount of suppression pool bypass can be defined as the amount that will not result in the containment pressure exceeding the containment design pressure.

6.2.1.8.1 Analyses

GE performed the analyses given in SSAR Sections 6.2.1.1.5.3 and 6.2.1.1.5.4 to investigate the sensitivity of the containment design to suppression pool bypass. In general, slow reactor depressurization events lead to more limiting suppression pool bypass cases. This is the case because the horizontal vents in the suppression pool remain covered and all of the steam released from the reactor passes through the bypass pathway into the wetwell airspace.

For the first analysis, GE assumed a controlled reactor cooldown rate of 55.6 °C (100 °F) per hour for approximately 6 hours. The assumptions were conservative, and credit was not taken for structural heat sinks or actuation of the containment sprays. The analysis indicates a maximum allowable effective leakage path ($A/K^{1/2}$) of 5 cm^2 (0.0054 ft^2).

For the second analysis, GE also assumed a controlled reactor cooldown rate of 55.6 °C (100 °F) per hour for approximately 6 hours and credited operation of the wetwell portion of containment sprays with a flow rate of 114 m^3/hr (4,026 ft^3/hr) and heat transfer to the containment boundary surfaces. A spectrum of postulated reactor coolant system pipe breaks were evaluated to determine the most limiting case. The containment sprays were assumed to initiate 30 minutes after the containment reached a pressure of 103.0 kPa G (14.9 psig). This pressure is above the emergency operating procedure spray actuation pressure as specified in SSAR Section 18A. This analysis is similar to that performed for the Mark II containments. The analysis indicates a maximum allowable effective leakage path ($A/K^{1/2}$) of 50 cm^2 (0.054 ft^2).

The staff concludes that sufficient capability exists in the containment to handle limited amounts of steam bypass of the suppression pool.

6.2.1.8.2 Wetwell Sprays for Mitigation of Steam Bypass

As discussed in Section 6.2.1.8.1 above, GE assumed a wetwell spray flow of 114 m^3/hr (4,026 ft^3/hr) to mitigate the consequences of steam bypass. Wetwell spray cannot be operated in isolation. Combined wetwell and drywell sprays are the normal mode of operation. With operator intervention, the wetwell sprays could be combined with either the suppression pool cooling mode or low pressure vessel injection mode. An orifice is included in the wetwell spray line to limit the flow to the stated amount. Wetwell spray flow is balanced by the orifice to provide (1) enough flow to mitigate pressurization of the wetwell due to steam bypass and (2) not too much flow to exceed the negative design pressure of the wetwell as discussed in SSAR Section 6.2.1.1.4.

GE deviates from the SRP in Section 6.2.1.1.5.6.1 of the SSAR by not requiring automatic actuation of the wetwell sprays 10 minutes after a LOCA signal. GE's analyses in Section 6.2.1.1.5.3 of the SSAR show that approximately 30 minutes is available for initiation of the containment sprays after the wetwell pressure has passed the lower limit of the containment spray actuation levels.

The staff concludes that GE has demonstrated that sufficient time is available to initiate containment sprays for the analyzed bypass scenario. Therefore, the staff agrees that automatic initiation of containment spray is not necessary.

6.2.1.8.3 Periodic Testing and Inspection

The acceptance criteria in SRP Section 6.2.1.1.C indicate the need for a preoperational high-pressure leak test, periodic low-pressure leak tests, and periodic visual inspection of the vacuum relief system. In Section 6.2.1.1.5.7 of the SSAR, GE committed to perform these tests and inspection. The acceptance criterion for both the high- and low-pressure leak tests is a measured bypass leakage area that is less than 10 percent of the effective suppression pool bypass capability ($A/K^{1/2}$) of 50 cm^2 (0.054 ft^2).

The preoperational high-pressure leak test is discussed in SSAR Sections 6.2.1.1.5.7.1 and 14.2.12.1.41. The low-pressure leak test is discussed in SSAR Section 6.2.1.1.5.7.2 and is included in Surveillance Requirement 3.6.1.1.3 of TS 3.6.1.1, "Primary Containment." The periodic visual inspection of the vacuum relief system is discussed in SSAR Section 6.2.1.1.5.7.4 and will be performed each refueling outage in accordance with TS 3.6.1.6, "Wetwell-to-Drywell Vacuum Breakers."

The staff concludes that these testing and inspection requirements will ensure that the suppression pool steam bypass capability of the containment is not exceeded.

6.2.1.8.4 Vacuum Breaker Position Indicators and Alarms

In Section 6.2.1.1.5.8.1 of the SSAR, GE committed to provide redundant position indicators on all vacuum breakers with redundant indication and alarm in the control room. The sensitivity of the indicator system is adequate to detect a total valve opening, for all valves, that is less than the design bypass capability for a small break. The detectable valve opening is based on the assumption that it is evenly divided among all vacuum breakers.

In its discussions with GE, the staff stated that the redundant indicator and alarm system should be tested and calibrated periodically and included in the TS. TS 3.6.1.6 requires a channel calibration of the vacuum breaker position indicator system every 18 months, thus ensuring the system's sensitivity. The staff finds this acceptable.

The staff concludes that the vacuum breaker position indicator and alarm system will ensure that the contribution of steam bypass of the suppression pool through the vacuum breaker system is within the containment design basis.

6.2.1.8.5 Monthly Vacuum Breaker Operability Test

GE deviates from the SRP in Section 6.2.1.1.5.6.2 of the SSAR by not requiring the monthly operability test of the vacuum breakers. This test was designed to ensure that the vacuum breakers adequately perform the design function of opening and then returning to a fully closed position.

There are eight vacuum breaker lines in the ABWR containment with one swing check valve per line, as opposed to most operating BWRs, which have two valves per penetration. In the DFSER, the staff concluded that the wetwell-drywell vacuum breaker system was acceptable. Failure of one of the vacuum breakers to open is within the design basis of the plant, as only seven are required to relieve pressure differentials.

Given the acceptability of a single barrier in the vacuum breaker lines, monthly operability testing could result in one of the valves failing to reseat because of the binding of the mechanical operator. Failure of one of the vacuum breakers to close could result in the containment pressure exceeding its design basis.

With only one barrier per vacuum breaker line, the staff concludes that monthly operability testing, which could result in failure of the valves to reseat or in binding of the mechanical operators, may not be necessary because the valves are designed for single failure to open. However, the staff believes that the single barrier per line places additional importance on the vacuum breaker position indicator and alarm system and wetwell spray system as discussed above. These systems will ensure that the design-basis steam bypass of the suppression pool is not exceeded through the vacuum breakers. In addition, TS 3.6.1.6 requires a plant shutdown within 12 hours when one vacuum breaker is not closed or cannot be demonstrated to be closed.

6.2.1.8.6 Conclusion

The staff concludes that GE's analysis, provisions for periodic testing and inspection, and vacuum breaker position indicator and alarms are acceptable and meet SRP Section 6.2.1.1.C (Rev. 6), "Pressure-Suppression Type BWR Containments," relative to steam bypass of the suppression pool. In addition, GE has provided adequate justification for not providing automatic actuation of wetwell sprays and not requiring monthly operability (stroke) tests of the vacuum breakers. Since failure of one of the vacuum breakers to close could result in the containment pressure exceeding its design basis, this acceptability relies greatly on the instrumentation provided to detect opening of the vacuum breaker lines and places added importance on the wetwell spray system.

6.2.1.9 Containment Debris Protection for ECCS Strainers

The emergency core cooling system (ECCS) suction strainers are located in the suppression pool, and their function is to ensure that debris in the suppression pool does not lead to clogging of ECCS pumps, heat exchangers, valves, and spray nozzles. To accomplish this function, debris in the suppression pool will be filtered out on the surface of the suction strainers. An excessive accumulation of debris on the strainer surface could lead to inadequate net positive suction head (NPSH) and failure of the ECCS pumps.

In 1985, the NRC issued RG 1.82, Revision 1, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident," which contains guidance on the sizing criteria for ECCS strainers. Recent events at operating reactors involving the clogging of ECCS strainers have led the staff to conclude that the guidance in RG 1.82, Revision 1, may not be conservative enough to eliminate this concern. To address this issue for operating reactors, the NRC issued Information Notice (IN) 92-71, "Partial Plugging of Suppression Pool Strainers at a Foreign BWR"; IN 93-34, "Potential for Loss of Emergency Cooling Function Due to a Combination of Operational and Post-LOCA Debris in Containment"; Supplement 1 to IN 93-34; and Bulletin 93-02, "Debris Plugging of Emergency Core Cooling Suction Strainers." The staff is still working on resolving this issue for operating reactors.

To address this issue for the ABWR, GE committed to the following: (1) the guidance for strainer sizing in RG 1.82, Revision 1, will be met; (2) for surfaces in contact with the suppression pool, stainless steel will be used; (3) the upper drywell connecting vent openings will be protected with horizontal plates and vertical trash racks; (4) the strainers will be in a "T" configuration; (5) a suppression pool cleanup system will be used; and (6) the COL applicant will develop a program for maintaining suppression pool cleanliness.

In the advance SER, the staff stated that it believed that the actions specified by GE were appropriate; however, GE did not address the possible lack of conservatism in RG 1.82, Revision 1, due to the deleterious effect of finely fragmented insulation. Reducing the total amount of insulation in the containment would not resolve this problem because the sizing criterion is based on correlations in the regulatory guide. Therefore, less insulation would lead to smaller strainers. The staff believed an acceptable resolution to this issue was to size the strainers in accordance with RG 1.82, Revision 1, but provide a factor of 3 sizing margin to account for uncertainty in the synergetic effects of strainer clogging from insulation, corrosion products, and other debris. In the advance SER, the staff identified this as Open Item F6.2.1.9-1.

In Amendment 35, GE committed to size the RHR suction strainers 3 times the area that results from RG 1.82 for all breaks. The HPCF and RCIC system suction strainers are sized according to RG 1.82, but with conservatism in the mass of debris assumed to be deposited on the strainers from a design basis aspect. GE also committed to provide a 10 percent margin in the net positive suction head available from the static head of the suppression pool for conservatism. Based on the commitments in Amendment 35, the staff finds this acceptable. This resolved Open Item F6.2.1.9-1.

6.2.2 Containment Heat Removal System

The containment heat removal system, which is an integral part of the RHR system, consists of three redundant loops. Each loop is designed so that a failure in one loop cannot cause a failure in another. In addition, each of the loops and associated equipment are located in a separate protected area of the reactor building to minimize the potential for single failure, including the loss of onsite or offsite power causing the loss of function of the entire system. The system equipment, piping, and support structures are designed to seismic Category I criteria.

The containment heat removal system encompasses the following RHR operating modes:

- Low-Pressure Flooder (LPFL) Mode

  After a LOCA, containment cooling starts as soon as the LPFL injection flow begins. During this mode, water from the suppression pool is pumped through the RHR heat exchangers and injected into the reactor vessel. The LPFL mode is automatically initiated by a low water level in the reactor vessel or high pressure in the drywell. In addition, each loop in the RHR system can be placed in operation by means of a manual initiation push-button switch.

- Suppression Pool Cooling Mode

  After a LOCA, the suppression pool cooling subsystem provides a means to remove heat released into the suppression pool. During this mode of operation, water is pumped from the suppression pool through the RHR heat exchangers and back to the suppression pool. This mode is automatically initiated, as needed, by closing the LPFL injection valves and opening the suppression pool return valves. In response to a request for additional information (RAI), GE stated that the heat removal function will be initiated within 10 minutes after a LOCA. The staff found this to be sufficiently conservative and adequate to achieve the necessary containment cooling function.

- Containment (Wetwell and Drywell) Spray Cooling Mode

  Two of the RHR loops include containment spray cooling subsystems. Each subsystem provides both wetwell and drywell spray cooling. This subsystem provides steam condensation and primary containment atmospheric cooling after a LOCA by pumping water from the suppression pool, through the RHR heat exchangers, and into the wetwell and drywell spray spargers in the primary containment. The normal mode of containment spray operation is combined wetwell and drywell sprays. However, the wetwell sprays can be operated in conjunction with either the suppression pool cooling mode or low pressure flooder mode. The drywell sprays can be operated in isolation through a series of operator actions; however, this is not intended to be used in isolation following an accident.

Provisions have been made to permit inservice inspection (ISI) of the RHR system components and functional testing of active components.

The location of suction and return lines in the suppression pool will facilitate mixing of the return water with the total pool inventory before the return water becomes available to the suction lines.

RG 1.1, "Net Positive Suction Head (NPSH) for Emergency Core Cooling and Containment Heat Removal System Sumps," prohibits design reliance on pressure or temperature transients expected during a LOCA for ensuring NPSH. The ABWR NPSH design assumes 0-kPa (O-psig) containment pressure and the maximum expected fluid temperatures resulting from a LOCA and, therefore, is acceptable. The suppression pool makeup system will provide additional water from the condensate storage tank (CST) through the suppression pool cleanup (SPCU) system to the suppression pool by gravity flow during normal conditions. Following a LOCA, the ECCS will take suction from the suppression pool. The quantity of water is sufficient to account for all conceivable postaccident entrapment volumes (i.e., places where water can be stored while maintaining long-term of the drywell vents with water).

The staff concludes that the containment heat removal systems satisfy SRP Section 6.2.2 and RG 1.1 and are acceptable.

6.2.3 Secondary Containment Functional Design

The ABWR secondary containment boundary completely surrounds the primary containment and is designed to remove fission products released from the primary containment during a DBA to limit the whole-body and thyroid doses to 10 CFR Part 100 requirements at the site boundary and the GDC 19 limit for the control room operator, as discussed in Chapters 12, 15, and Section 6.4 of this report. The staff reviewed the secondary containment functional design in accordance with SRP Section 6.2.3. The design is considered acceptable if the relevant requirements of GDC 4, 16, and 45 and Appendix J to 10 CFR Part 50 are complied with. The relevant requirements are as follows. GDC 4 as it relates to structures, systems, and components important to safety being designed to accommodate the effects of normal operation, maintenance, testing and postulated accidents, and being protected against dynamic effects (e.g., the effects of missiles, pipe whipping, and discharging fluids) that may result from equipment failures. GDC 16 as it relates to reactor containment and associated systems being provided to establish an essentially leak-tight barriers against the uncontrolled release of radioactivity to the environment. GDC 43 as it relates to atmosphere cleanup systems having the design capability to permit periodic functional testing to assure system integrity, the operability of active components, and the operability of the system as a whole and the performance of the operational sequence that brings the system into operation. 10 CFR Part 50, Appendix J as it relates to the secondary containment being designed to permit preoperational and periodic leakage rate testing so that bypass leakage paths are identified.

The components of the secondary containment are designed to withstand missiles, pipe whip, postaccident environments, seismic events, a single active failure, and a loss of offsite power coincident with an accident, in accordance with GDC 4. The two systems that fulfill this function are the secondary containment heating, ventilation and air conditioning (HVAC) system and the standby gas treatment system (SGTS). They are both located inside secondary containment.

The HVAC system will maintain a slightly negative pressure in the secondary containment during normal operation to prevent any radioactivity from escaping to the environment. The SGTS will provide postaccident filtration and removal of airborne halogens and particulates from the secondary containment. The SGTS is designed to maintain a negative pressure of -0.25 in. water gauge (secondary containment to environment) after any postulated accident. GE indicates that testing and inspection of the integrity of the secondary containment is part of the testing of the SGTS. The staff's evaluation of the SGTS is given in Section 6.5.3 of this report.

SRP Section 6.2.3 states that all openings, such as personnel doors and equipment hatches, should be under administrative control. These openings should be provided with position indicators and alarms having readout and alarm capability in the main control room (MCR). The effect of open doors or hatches on the functional capability of the depressurization and filtration systems should be evaluated. The staff requested (in RAI Q430.34) that GE provide a list of the secondary containment openings and the instrumentation used to ensure that each one is closed during a postulated DBA. In response to RAI Q430.34, GE submitted SSAR Table 6.2-9, which lists all secondary containment penetrations along with their elevation and diameter; however, GE did not sufficiently address the staff concerns. The staff identified this issue as Open Item 14 in the DSER (SECY-91-355).

To address the staff's concern, GE revised SSAR Table 6.2-9 and provided notes to the table, marking the applicable penetrations with an asterisk. The notes state that (1) the HVAC openings have safety-related isolation valves with both local monitoring and remote (in control room) monitoring and (2) the doors are monitored in the control room in accordance with SSAR Section 13.6.3.4. The staff finds that GE's response is acceptable because the other penetrations, not covered by the notes, are fluid piping systems that are designed under containment isolation provisions. Therefore, DSER Open Item 14 is resolved.

Furthermore, in RAI Q430.32 and Q430.34, the staff asked for information on the capability of the SGTS to draw a negative pressure following an accident assuming that all lines that do not receive an isolation signal are open and assuming the worst-case single failure of a secondary containment isolation valve (CIV) to close. In addition, the staff asked GE to identify the instrumentation used to ensure that the CIVs shut. GE's initial response did not specify the alarm capability for these indicators and did not address alarms available in the control room.

In response to RAI Q430.32, GE revised SSAR Section 6.5.1.3.1 to state that a secondary containment draw-down analysis will be performed to demonstrate the capability of the SGTS to maintain the designed negative pressure following a LOCA, including inleakage from the open, nonisolated penetration lines identified during construction engineering and in the event of the worst single failure of a secondary CIV to close. GE also added SSAR Section 6.5.5.1 to state that the COL applicant will perform an SGTS dose, functional damage, and draw-down analysis in accordance with SSAR Sections 6.5.1.2.3.7 and 6.5.1.3.1(5). The staff found GE's response acceptable. This was identified as COL Action Item 6.2.3-1 in the DFSER.

To further address this issue, GE submitted a list of secondary containment penetrations in SSAR Table 6.2-9. All piping and cable-tray penetrations will be sealed with a sealing compound for leakage and fire protection and are all provided with containment isolation features. To ensure that the SGTS is capable of maintaining the design negative pressure, GE stated that the draw-down analysis would assume 50-percent containment inleakage regardless of valve positions. During construction engineering, the COL applicant will identify the piping that could cause inleakage. SSAR Section 6.5.1.3.2 states that each SGTS fan is sized to individually establish a continuously negative differential pressure (considering the effect of wind) within 10 minutes after SGTS initiation. The dose analysis assumes direct leakage from the containment to the environs for twice the required draw-down period. The staff finds that the SGTS design includes some conservatism when considering any malfunction of secondary containment isolation features. The staff concludes that it does not need to evaluate the results of the draw-down analysis for its FSER safety finding. Therefore, COL Action Item 6.2.3-1 is acceptable as included in the SSAR.

The staff has reviewed the secondary containment functional design and the capability of the SGTS to maintain a negative pressure in the secondary containment following a LOCA. The SGTS has two parallel and redundant filter trains, both of which will be automatically actuated (one train will be placed in the standby mode) by a high drywell pressure signal or a low reactor water level signal, or when high radioactivity is detected in the secondary containment or refueling floor ventilation exhaust. The system will be on standby during normal plant operation and may be manually initiated from the control room for primary containment de-inerting prior to plant shutdown. Normal operation of the SGTS at power is limited to 90 hours per year for both trains combined. There are alarms and indicators for all LOCA and high-radiation mitigation systems.

The staff concludes that the secondary design and the SGTS meet the requirements of GDC 16 as it relates to establishing an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment, and are adequately designed to ensure that failure of any active component, assuming loss of offsite power, cannot impair the capability of the system to perform its safety function and, therefore, is acceptable.

Secondary Containment Bypass Leakage

Although the primary containment is enclosed by the secondary containment, there are systems that penetrate both the primary and secondary containment boundaries, creating potential paths through which radioactivity in the primary containment could bypass the leakage collection and filtration systems associated with the secondary containment. A number of the system lines contain physical barriers or design provisions that can effectively eliminate bypass leakage. These include water seals, containment isolation provisions, and vent return lines to the controlled regions. The acceptance criteria by which potential bypass leakage paths are reviewed are given in Branch Technical Position (BTP) CSB 6-3, "Determination of Bypass Leakage Paths in Dual Containment Plants," of the SRP.

BTP CSB 6-3 states that the evaluation of bypass leakage involves both the identification of bypass leakage paths and the determination of leakage rates. Potential bypass leakage paths are formed by penetrations which pass through both the primary and secondary containment boundaries. Penetrations that pass through both the primary and secondary containment may include a number of barriers to leakage (e.g., isolation valves, seals, gaskets, and welded joints). While each of these barriers aid in the reduction of leakage, they do not necessarily eliminate leakage. Therefore, in identifying potential leakage paths, each of these penetrations should be considered, together with the capability to test them for leakage in a manner similar to the containment leakage tests required by Appendix J to 10 CFR Part 50.

In RAI Q430.33, the staff requested GE to specify and justify the maximum allowable fraction of primary containment leakage that may bypass the secondary containment boundary. In its response, GE stated that only valve leakage through process piping can bypass the secondary containment and that leakage will be monitored via the containment local leakage test (Type C test) on the outboard CIV valves. In response to the related RAI

Q430.52 (b and c), GE stated that information on potential bypass leakage paths is given in SSAR Section 6.2.3 and SSAR Table 6.2-10.

GE has evaluated each penetration in the secondary containment in accordance with the guidance of BTP CSB 6-3. SSAR Table 6.2-10 identifies where each path terminates, the leakage barriers for the path, and whether the path is considered a potential bypass leakage path. The potential bypass leakage paths identified in SSAR Table 6.2-10 are the drywell and wetwell purge systems and the main steam system. Each of these paths has redundant containment isolation valves which are Type C tested in accordance with Appendix J. Additionally, the drywell and wetwell purge systems terminate inside the secondary containment where the SGTS collects and treats any leakage. GE has accounted for 4.2 m^3/hr (140 standard cubic feet per hour (SCFH)) leakage through the MSIVs as the secondary containment bypass leakage rate to the environment, and this leakage has been treated separately in the offsite dose analysis. The testing and inspection of the integrity of the secondary containment will be part of the testing of the SGTS as described in SSAR Section 6.5.1. Therefore, the staff finds that the ABWR meets the requirements of GDC 43 and 10 CFR Part 50, Appendix J regarding the inspection and testing of the secondary containment system, and is acceptable.

Tier 1 design information and ITAAC are required for the functional design of the secondary containment. GE submitted the design description and ITAAC for the secondary containment. However, the results of the staff's review was not complete when the DFSER was issued. This was identified as Open Item 6.2.3.1-1 in the DFSER. GE subsequently submitted a revised design description and ITAAC. The adequacy and acceptability of the design description and ITAAC are evaluated in Section 14.3 of this report. Therefore, DFSER Open Item 6.2.3.1-1 is resolved.

The staff concludes that the secondary containment functional design and the SGTS are in compliance with GDC 4, 16, and 43, and Appendix J to 10 CFR Part 50, and with the guidance in SRP Section 6.2.3 and BTP CSB 6-3 and, therefore, are acceptable.

6.2.4 Containment Isolation System

The containment isolation system includes containment isolation valves (CIVs) and associated piping and penetrations necessary to allow normal or emergency passage of fluids through the primary containment while preserving the capability to prevent or limit the escape of fission products from the containment boundary in the event of a LOCA. The staff's review of this system includes the number and location of isolation valves, valve actuation signals and valve control features, the positions of valves under various plant conditions, the protection afforded isolation valves from missiles and pipe whip, and the environmental design conditions specified in the design of components. The containment isolation system design bases and containment isolation provisions should conform to GDC 1, 2, 4, 16, 54, 55, 56, and 57, as appropriate. Justification should be provided if deviations from these requirements exist.

GDC 1, 2, and 4 as they relate to systems important to safety being designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed; systems being designed to withstand the effects of natural phenomena (e.g., earthquakes) without loss of capability to perform their safety functions; and systems being designed to accommodate postulated environmental conditions and protected against dynamic effects (e.g., missiles, pipe whip, and jet impingement), respectively. GDC 16 as it relates to a system, in concert with the reactor containment, being provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment. GDC 54 as it relates to piping systems penetrating the containment being provided with leak detection, isolation, and containment capabilities having redundant and reliable performance capabilities, and as it relates to design provision incorporated to permit periodic operability testing of the containment isolation system, and leak rate testing of isolation valves. GDC 55 and 56 as it relates to lines that penetrate the primary containment boundary and either are part of the reactor coolant pressure boundary or connect directly to the containment atmosphere being provided with isolation valves as follows:

(a) One locked closed isolation valve inside and one locked closed isolation valve outside containment; or

(b) One automatic isolation valve inside and one locked closed isolation valve outside containment; or

(c) One locked closed isolation valve inside and one automatic isolation valve outside containment; or

(d) One automatic isolation valve inside and one automatic isolation valve outside containment.

GDC 57 as it relates to lines that penetrate the primary containment boundary and are neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere being provided with at least one locked closed, remote-manual, or automatic isolation valve outside containment.

The containment isolation provisions of the ABWR for system lines that penetrate the containment can be classified into three areas:

(1) system lines that meet the explicit requirements of GDC 54, 55, 56, and 57 regarding leak detection, isolation, and valve location and testing

(2) system lines that differ from the explicit requirements of GDC 54, 55, 56, and 57, but that difference has been justified under the specific guidelines given in SRP Section 6.2.4, which constitute acceptable alternate containment isolation provisions

(3) other lines that must be reviewed on a case-by-case basis to determine if an acceptable alternative basis exists for allowing a deviation from the explicit GDC on grounds not previously articulated in SRP Section 6.2.4.

During its review, the staff requested GE to provide additional information regarding the containment isolation provisions. RAI Q430.31, Q430.32, Q430.34, Q430.35, Q430.36, Q430.37, Q430.39, Q430.40, Q430.41, Q430.43, and Q430.44, all involve issues affecting the containment isolation system design. GE responded to all of these questions, and its responses are acceptable with the exception of those to Q430.34 and Q430.36.

Portions of GE's responses to Q430.32 and Q430.34, regarding secondary containment isolation related to the SGTS draw-down analysis, are evaluated in Section 6.2.3 of this report. In the DSER (SECY-91-355), the staff stated that the SGTS draw-down analyses should be included in the SGTS ITAAC. This was incorrectly identified in the DFSER as Open Item 6.2.4.1-1. It should have been DFSER Open Item 6.2.4-1. GE submitted a revised set of design descriptions and ITAAC. The adequacy and acceptability of the design description and ITAAC are evaluated in Section 14.3 of this report. On the basis of this evaluation, DFSER Open Item 6.2.4-1 is resolved.

In its submittal of March 11, 1992, GE further responded to RAI Q430.34 by stating that instrumentation requirements for the secondary containment openings were contained in SSAR Section 6.2.3.5. Subsequently, GE added a statement in SSAR Section 6.2.3.2 that all piping and cable-tray penetrations will be sealed with a sealing compound for leakage and fire protection. All doors are vestibule type with card reader access security systems that are monitored. The HVAC penetrations are designed to close in the event of a design-basis accident. The required testing procedures and frequency are included in the TS.

In Q430.36, the staff asked GE to identify the systems and the relevant interface requirements for CIVs not within the ABWR scope and discuss essential and nonessential systems in accordance with RG 1.141, "Containment Isolation Provisions for Fluid Systems," Revision 0. GE stated that all isolation valves are within the scope of the ABWR standard plant. However, it did not provide containment isolation provisions for the essential and nonessential systems. This was identified as Open Item 15 in the DSER (SECY-91-355).

In response to DSER Open Item 15, GE stated that containment isolation provisions for the isolation valves were specified in SSAR Table 6.2-7, which gives information on CIVs on a system-by-system basis. However, the table did not identify the system lines as essential or nonessential to address the Three Mile Island (TMI) action plan requirements (NUREG-0737, Item II.E.4.2). RG 1.141 contains guidance on the classification of essential versus nonessential systems. Each nonessential penetration (except instrument lines) is required to meet GDC 54, 55, 56, and 57 and will be isolated automatically by the containment isolation signal. Essential systems, such as systems related to engineered safety features (ESF) or systems needed for safe shutdown of the plant, may include remote-manual CIVs.

The staff finds that SSAR Table 6.2-7 contains the valve information on a system-by-system basis with containment isolation provisions for both ESF and non-ESF systems. It concludes that the valve actuation for the ESF system reflects the function of the essential system, and is acceptable. Therefore, DSER Open Item 15 is resolved.

In the DSER (SECY-91-355), the staff stated that although GE had specifically committed to meet GDC 54, 55, 56, and 57, there was no commitment to meet GDC 1, 2, 4, and 16 in accordance with SRP Section 6.2.4. GDC 1 requires that structures, systems, and components (SSCs) important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. GDC 2 requires that SSCs important to safety be protected from the effects of natural phenomena. GDC 4 requires that these SSCs be protected against dynamic effects. GDC 16 requires that the reactor containment and associated systems be provided to establish an essentially leaktight barrier against the uncontrolled release of radioactivity to the environment. This lack of commitment to meet GDC was identified as DSER Open Item 16.

In its response to this open item, GE stated that the ABWR commitments to GDC 1, 2, 4, and 16 are contained in SSAR Sections 3.1.2.1.1, 3.1.2.1.2, 3.1.2.1.4, and 3.1.2.2.7, respectively. The staff reviewed SSAR Section 3.1 and found that GE had met the above-cited GDC in the ABWR design. However, SSAR Section 3.1.1 states that the GDC are intended to guide the design of all water-cooled nuclear power plants and the GDC are subject to a variety of interpretations. In reviewing these criteria in SSAR Section 3.1.2, the staff noted that some of the wording pertaining to the GDC did not match the wording in 10 CFR Part 50. The staff pointed out these discrepancies and asked GE to change the wording. GE made the changes in SSAR Section 3.1.2. The staff finds this acceptable. Therefore, DSER Open Item 16 is resolved.

In addition, in response to Q430.41, GE stated that instead of meeting the requirements of GDC 56 for the high-pressure core flooder (HPCF) and residual heat removal (RHR) test and pump miniflow bypass lines, reactor core isolation cooling (RCIC) turbine exhaust and pump miniflow bypass lines, and suppression pool cleanup (SPCU) suction and discharge lines, it will use GE Safety Standard 20, Nos. 8 and 9. The staff stated that these standards did not meet the explicit requirements of GDC 56 and GE must provide justification for this deviation. This was identified as Open Item 17 in the DSER (SECY-91-355) and as Open Item 6.2.4-3 in the DFSER.

GE stated that GE Standard 20, Nos. 8 and 9, are GE's criteria to meet GDC 56 on another defined basis. Specifically, No. 8 is the criterion applicable to a line that (1) penetrates the containment, (2) communicates with the containment interior, (3) is not an instrument line, and (4) is not a suppression pool effluent line. No. 9 is similar to No. 8, but is applicable to the effluent line that communicates with the suppression pool. The term "communicates" is intended to mean that these lines are not closed loops, they may leak, or open to the containment interior or the suppression pool.

GE Safety Standard 20, No. 8, requires that each of these lines be provided with two isolation valves. At least one valve should be located outside the containment; the other valve may be located either inside or outside the containment. Alternative, one isolation valve outside the containment, which will be normally closed (or a blind flange) and will not receive a signal to open after an accident, may be used. On influent lines having two valves, one may be a check valve, and the valve outside the containment must be capable of automatic or remote-manual closure, or should be normally locked closed. On effluent lines or where a second valve is not provided on an influent line, these valves should be capable of automatic and remote-manual closure or should be normally locked closed. The valves should be located as close as practicable to the containment. GE Safety Standard 20, No. 9, requires that the suppression pool effluent lines be provided with one remote-manual valve outside the containment that should be located as close to the containment as practicable. GE also provided additional information in SSAR Table 6.2-7 (Amendment 34) by adding notes h, i, j, k, l, q, and r to address the containment isolation provisions for these system lines.

The staff has reviewed the valve arrangement in SSAR Table 20.3.2-3 and SSAR Section 6.2.4.3.2.2 (Amendment 34) and finds that these system lines should not be automatically isolated following a containment isolation signal because of the specific safety function they will perform. Specifically, the RCIC system, in conjunction with the HPCF and RHR system, will provide core cooling to prevent excessive fuel temperature during a LOCA. To protect the RCIC pump from overheating, the RCIC system contains a pump miniflow bypass line that will discharge into the suppression pool when the line to the reactor vessel is isolated. When flow through the pump to the vessel is sufficient, the isolation valve in the miniflow line will close automatically, thus directing all flow to the reactor. The HPCF and RHR test and pump miniflow bypass lines have isolation capabilities commensurate with the importance to safety of isolating these lines. The RCIC pump miniflow bypass line will be isolated by a normally closed, remote manually actuated valve outside the containment. The RCIC turbine exhaust line, which penetrates the containment and will discharge to the suppression pool, is equipped with a normally open, motor-operated, remote-manual gate valve. The RHR and HPCF test and miniflow bypass lines each has a motor-operated valve outside the containment. The SPCU suction and discharge lines each has two isolation valves outside the containment because the penetration is under water. All these lines terminate below the suppression pool water level and will be sealed from the containment atmosphere following a LOCA.

Because of the specific safety function that these system lines will perform, their automatic isolation following a containment isolation signal is not appropriate. The staff finds that the containment isolation provisions do not degrade containment isolation capability and conform with the system's function. SRP Section 6.2.4 contains guidance for satisfying the GDC on another defined basis in that remote-manual actuation is provided for accident protection. Therefore, the staff concludes that the containment isolation design for the system lines meets GDC 56 on another defined basis (GE Safety Standard 20, Nos. 8 and 9) and is acceptable. This resolved DFSER Open Item 6.2.4-3.

During its review of SSAR Table 6.2-7, which delineates CIV information in response to Q430.35, the staff identified some isolation valve design features that did not conform to SRP Section 6.2.4. In the table, the normal valve positions are either "open" or "closed." There is no way to determine if closed is the same as locked closed as it is stipulated in SRP Section 6.2.4. In addition, GE did not justify the isolation valve closure times of less than 30 seconds for some relatively large-diameter valves for the drywell atmosphere systems, such as atmospheric control system (ACS) 55 cm (22 in.)-diameter valve T31-F004 and flammability control system (FCS) 15 cm (6 in.)-diameter valve T49-F006A. The selection of the drywell atmosphere systems closure times, as specified in SRP Section 6.2.4, should be based on the analysis of radiological consequences for the DBA. In the DSER (SECY-91-355), the staff asked GE to identify which valves were "open," "closed," or "locked closed" and provide the technical basis for drywell atmosphere closure times. This was identified as part of Open Item 17 in the DSER (SECY-91-355).

GE stated that valve condition is given on the system piping and instrumentation diagram (P&ID) as locked open/closed, normally open/closed, and normally energized/deenergized. For clarity, the staff prefers that these valve positions are also specified in SSAR Table 6.2-7. In discussions with the staff, GE committed to make the change and this became Confirmatory Item 6.2.4-1 in the DFSER. In a subsequent amendment, GE added a note to SSAR Table 6.2.7 which states that the P&ID's identify which CIV's are locked closed. This is acceptable. This resolved DFSER Confirmatory Item 6.2.4-1.

In its submittal of March 11, 1992, GE stated that ACS isolation valves (T31-F001, -F002, -F004, -F006, and -F009) are 55 cm (22 in.)-diameter butterfly valves with an air operator that can travel 90° to close. These valves will be normally closed because their main function is to support containment purging and nitrogen inerting when the reactor is at less than 15-percent power. The smaller valves (T31-F005, -F039, -F040, and -F041) are 5 cm (2 in.)-diameter, air-operated globe valves that will close within 15 seconds. These air-operated valves may be opened for short periods during reactor operation to lower primary containment pressure or to keep the primary containment pressurized (by adding nitrogen) for preventing any air in-leakage. T31-F008 is a 25 cm (10 in.)-diameter, normally closed outboard air-operated butterfly isolation valve that will be connected to the SGTS and opened in series with T31-F005. T31-F025 is a 41 cm (16 in.)-diameter, normally closed outboard butterfly isolation valve that will be opened only during initial inerting of the primary containment when the reactor is below 15-percent power. Open/close times of 20 seconds or less are planned for all except the 5 cm (2 in.)-diameter valves.

FCS isolation valves (T49-F001, -F002, -F006, and -F007) are normally closed 15 cm (6 in.)-diameter gate valves. These valves may be actuated individually for testing during reactor operation. Two of these valves will be air or nitrogen operated and two valves will be motor operated. This system will not be activated for days following a LOCA and a closure time of less than 30 seconds is planned.

In the DSER (SECY-91-355), the staff stated that the closure times for these valves, including instrumentation delay, should be determined by demonstrating compliance with 10 CFR Part 100 regarding offsite radiological consequences. The closure time should be justified on the basis of an analysis of the radiological consequences of a LOCA. However, GE had not explained how the valve open/close times were determined, or that the radiological consequences following a LOCA meet the requirements of 10 CFR Part 100. This was identified as Open Item 6.2.4-4 in the DFSER, which was part of Open Item 17 in the DSER (SECY-91-355).

GE stated that the FCS is a closed-loop system that will be capable of controlling combustible gas concentrations in the containment atmosphere for the design-basis LOCA without relying on purging and without releasing radioactive material to the environment. The time limit for valve closure is not important, since offsite radiological consequences need not be considered for a closed-loop system. Therefore, the staff finds that the closure time of less than 30 seconds for the FCS is acceptable.

SRP Section 6.2.4 states that the closure times of the containment purge and vent isolation valves should be established on the basis of minimizing the release of containment atmosphere to the environs and that isolation valve closure times of about 5 seconds or less may be necessary. SSAR Section 6.2.4.3.2.2.2.3 (Amendment 34) states that the ACS CIV closure time is equal to or less than 20 seconds. The ABWR TS require the 55 cm (22 in.)-diameter ACS CIVs to be normally sealed closed and to be opened only when reactor power is less than 15 percent during the 24-hour period following entry into Mode 3 for inerting the containment (startup) and the 24-hour period before entering Mode 4 for de-inerting the containment (shutdown). If open, they will automatically close on drywell high pressure or high radiation detected in the exhaust flow. The exhaust radiation detectors are designed to be very sensitive and will be set at a lower setpoint compared to the ones inside the containment to ensure early detection. GE stated that the difference between 5 and 20 seconds is considered to be insignificant and that on the basis of its analysis, the closure times of these valves are significantly shorter than necessary to prevent any radiological impact. Additionally, GE stated that valves with moderate speed will be more reliable and, therefore, ensure containment integrity.

The staff finds that the closure time of 20 seconds or less for the ACS isolation valves is acceptable based on GE's accident analysis and justification that slower acting valves are more reliable. The accident analysis in Chapter 15 of the SSAR assumes, among other things, that all containment isolation valves isolate within 60 seconds following a postulated LOCA for calculating the offsite doses. The 20-second or less closure time of the ACS CIVs is small compared to this assumption. Additionally, the analysis assumes primary containment releases through the penetrations to be 0.5-percent by weight per day of the containment free volume. The contribution to the offsite doses due to the leakage through the ACS CIVs for the period from 5- to 20-seconds following a LOCA is insignificant when compared to the offsite doses due to the assumed leakage through the penetrations, and is acceptable. This resolved Open Item 6.2.4-4.

Containment Purge System

The staff requested GE (in RAI Q430.42) to address the containment isolation provisions for the purge lines that show conformance with BTP CSB 6-4, "Containment Purging During Normal Plant Operations," of the SRP. In its initial response, GE incorporated a new proprietary SSAR Section 9.4.5.6, "Primary Containment Supply/Exhaust System," which gives design information on the primary containment purge supply/exhaust system. The system consists of the supply fan, a high-efficiency particulate air (HEPA) filter, a purge fan, duct work, and controls. GE also provided additional information in SSAR Table 6.2-7 and Figure 6.2-39a (the atmospheric control system P&ID).

In response to RAI Q430.42, GE stated that the containment purge supply and exhaust lines, connected to both the drywell and wetwell, consist of one supply and one exhaust penetration each for the drywell and wetwell. Both the purge supply and exhaust lines, each of which is connected to both the drywell and wetwell, have two parallel isolation valves that will be located as close as possible to the outside of the primary containment. One of these valves (55 cm (22 in.)-diameter) is intended for use for (high-volume) inerting and purging. The other valve (5 cm (2 in.)-diameter) will be used for any necessary nitrogen makeup during power operation. The two 5 cm-diameter exhaust valves will be used for any necessary venting (e.g., for pressure control) during operation. All these isolation valves are air operated and will automatically be closed by high drywell pressure or Level III low reactor vessel water level signals. They also will fail in the closed position if actuating power is lost. The large-diameter valves are butterfly-type valves with a closure time of less than 20 seconds. The small-diameter valves are globe-type valves and have closure times of less than 15 seconds. The staff finds that the containment isolation provisions for the purge valves are in conformance with BTP CSB 6-4. However, this valve configuration does not comply with GDC 56, which requires one isolation valve inside and one isolation valve outside the containment for each penetration. This was identified as Open Item 6.2.4.1-1 in the DFSER and as Open Item 18 in the DSER (SECY-91-355).

GE stated that purge and vent lines do not extend into the containment and have both inboard and outboard containment isolation valves (CIVs) located outside the primary containment so that they are not exposed to the harsh environment of the wetwell and drywell and are accessible for inspection and testing during reactor operation. SSAR Section 6.2.4.3 (Amendment 34) states, in part, that the CIVs for the atmospheric control system located outside the containment will be located as close to the containment as practical. The piping from the containment up to and including both valves is an extension of the primary containment boundary and is designed in accordance with ASME Code, Section III, Class 2 requirements. The CIVs are protected from the effects of flood and dynamic effects of pipe breaks in accordance with SSAR Sections 3.4 and 3.6. The arrangement of the isolation valves and connecting piping is such that a single failure of an inboard valve, or a single active or passive failure in the connecting piping or an outboard valve, cannot prevent isolation of the ACS. The valves are air operated with a pilot dc solenoid valve that will fail closed on loss of air or loss of electrical power. The power for the dc solenoids will be supplied from independent electrical divisions.

The staff's position on this issue is that locating both isolation valves outside the containment is acceptable if piping and valve design criteria are sufficiently conservative to preclude a breach of integrity. In general, the isolation barriers should be designed to ESF criteria and protected against floods, missiles, pipe whip, and jet impingement. GDC 56 permits containment isolation provisions for lines penetrating the primary containment boundary that differ from GDC 56, provided the basis for acceptability is defined. The staff concludes that this valve arrangement precludes a breach of piping integrity, meets the single-failure criterion, and is protected against the effects of flooding, missiles, pipe whip, and jet impingement and meets GDC 56 on another defined basis, and is acceptable. Therefore, DFSER Open Item 6.2.4.1-1 is resolved.

In the DSER (SECY-91-355), the staff identified a number of criteria that are delineated in BTP CSB 6-4, but that GE did not address, including

- radiological consequence analysis for a LOCA with the purge system initially open (BTP CSB 6-4, B.5.a)
- system structural integrity design under LOCA thermal-hydraulic conditions (BTP CSB 6-4, B.5.b)
- design provisions to ensure that isolation valve closure is not prevented by debris entrained in escaping air and steam (BTP CSB 6-4, B.1.g)
- during emergency core cooling system (ECCS) backpressure containment pressure reduction analysis for a LOCA which the purge system is initially open (BTP CSB 6-4, B.5.c)
- evaluation of case-by-case purge isolation valve maximum allowable leak rate (BTP CSB 6-4, B.5.d)
- technical justification for a purge system isolation valve closure time of more than 5 seconds (BTP CSB 6-4, B.1.f)
- design provisions to ensure that simultaneous venting of the wetwell and drywell will not occur

In its submittal of March 11, 1992, GE addressed these criteria as follows:

- Radiological consequence analysis for a LOCA with the purge system initially open relates to SSAR Section 16.3.6.3.2, regarding the technical specification requirements for the oxygen concentration in the primary containment. The atmospheric control system (ACS) 55 cm (22 in.)-diameter purge isolation valves normally will be sealed closed during plant operation and will be opened only during the inerting (startup) and de-inerting (shutdown) process when reactor power is less than 15 percent and within 24 hours of reactor shutdown or startup. Within such a limited time, if a LOCA does occur, these valves will have closed before the onset of fuel failure. In the event of a radioactivity leak during inerting/de-inerting, the radiation detectors at the purge and vent exhaust line will detect the condition and isolate the ACS CIVs. GE stated that the exhaust radiation detectors are designed to be very sensitive and will be set at a lower setpoint compared to the ones inside the containment to ensure early detection. Therefore, the potential for a LOCA to occur while the ACS 55 cm (22 in.)-diameter purge isolation valves are opened is small because the valves are allowed to be opened for limited times and the radiological consequence analysis shows that should a LOCA occur while they are open, site radiological limits will not exceed the limits of 10 CFR Part 100.
- Penetrations, piping, isolation valves, and rupture discs will maintain their structural integrity for all accident conditions. Periodic Type C tests of the isolation valves will be conducted at the containment peak pressure. The design meets seismic Category I and safety Class 2 requirements. Isolation valves will close automatically following LOCA signals and will fail closed on loss of instrument air. Purge system isolation valves (AO-F001, AO-F002, AO-F003, AO-F004, AO-F006, and AO-F025) will be locked closed whenever the reactor is above 15-percent power. Purge exhaust isolation valve AO-F005 normally will be closed and will open only for 90 hours per year along with operation of the standby gas treatment systems (SGTS) for containment pressure control. Nitrogen will be added to the primary containment during reactor operation by opening isolation valves AO-F0039 and AO-F0040 for the drywell and AO-F0039 and AO-F0041 for the wetwell.
- Drywell and wetwell purge penetrations will have seismic Category I debris screens.
- ECCS systems with suction from the suppression pool are designed with the primary containment at atmospheric pressure and without crediting net positive suction head for containment backpressure.
- Case-by-case maximum allowable leakage rates for isolation valve are based on valve size, type, and containment peak pressure for Type C tests as required by the TS. Test connections are provided. This was DFSER TS Item 6.2.4.1-1. Since GE included the testing requirements in the ABWR TS, the TS item is resolved.
- Purge system isolation valves AO-F005, AO-F040, and AO-F041 will be opened for short periods during reactor operation for containment pressure control. These valves are all 5 cm (2 in.)-diameter and are capable of full closure within 5 seconds.

GE provided adequate information to address these criteria with the exception of simultaneous venting of the drywell and wetwell. BTP CSB 6-4 prohibits simultaneous venting of the drywell and wetwell. In the DFSER, the staff stated that GE should show how this will be ensured. This was identified as Open Item 6.2.4.1-2.

In its submittal of January 22, 1993, GE proposed to add Section 6.2.4.3.5 to the SSAR. This section (Amendment 34) states, in part, that the large (55 cm (22 in.)) purge and vent lines for the ACS will not be used for purging or venting during normal reactor operation. The isolation valves in these lines are normally closed and they are not needed for pressure control of the containment during normal operations. Pressure control of the containment during operation will be maintained by a single, small (5 cm (2 in.)) nitrogen line and a single, small (5 cm (2 in.)) vent line. The small vent line will be attached to the 55 cm (22 in.) drywell purge exhaust line and bypass the closed 55 cm (F004) valve. There is no equivalent vent line from the wetwell. Therefore, the drywell and wetwell will not be vented simultaneously during operation, and the system has only one supply and one exhaust line as required by BTP CSB 6-4. The staff finds that GE's justification is acceptable. DFSER Open Item 6.2.4.1-2 is resolved.

The staff also requested (in RAI Q430.254) that GE explain how the drywell and wetwell purge supply and exhaust subsystems can meet BTP CSB 6-4. GE committed to provide such information. This was identified as Confirmatory Item 6.2.5-1 in the DFSER and as part of Open Item 21 in the DSER (SECY-91-355).

In response, GE stated that drywell and wetwell purging is not required during normal operation. The staff finds that BTP CSB 6-4 does not apply to the plant that does not have an online purge system. Therefore, DFSER Confirmatory Item 6.2.5-1 is resolved.

GDC 54 requires "containment capability having redundancy" for piping that penetrates the containment. However, SSAR Figure 6.2-39 shows common CIVs for the containment purge supply (T31-F001 from the reactor building HVAC system, T31-F025 from the 40 cm (16 in.) nitrogen purge line, and T31-F039 from the 5 cm (2 in.) nitrogen supply line) and exhaust (T31-F009 to the reactor building HVAC system and T31-F008 to the SGTS). The staff was concerned that this valve arrangement would leave the system vulnerable to common-mode failures. Each purge and exhaust line should have redundant and independent CIVs to comply with GDC 54. This was identified as Open Item 6.2.4.1-3 in the DFSER and as part of Open Item 18 in the DSER (SECY-91-355).

In its submittal of January 22, 1993, GE stated, in part, that the containment purge system has redundant CIVs, each powered from an independent electrical division, and the CIVs are arranged so that any single active failure will not compromise the integrity of the containment. GE also stated that this arrangement has adequate redundancy and independence and is not unduly vulnerable to common mode failures.

The staff reviewed the valve arrangement in SSAR Figure 6.2-39 and finds that the common CIVs mentioned above are the outboard CIVs, which are redundant. There are independent inboard CIVs for containment purge supply (T31-F002, -F003), nitrogen supply (T31-F040, -F041) and exhaust (T31-F004, -F005, -F006, -F007). All of these inboard CIVs will be closed automatically on receipt of a containment isolation signal and are designed to fail in the closed position on loss of air or loss of power to the pilot solenoid valves. Therefore, the staff concludes that the valve arrangement does not degrade containment integrity, meets the single-failure criterion, and is acceptable. DFSER Open Item 6.2.4.1-3 is resolved.

10 CFR 52.47(a)(1)(ii) requires that an application for a standard design certification to include a demonstration of compliance with any technically relevant portions of the TMI-related requirements identified in 10 CFR 50.34(f). In accordance with 10 CFR 50.34(f)(2)(xiv), GE has incorporated the following containment isolation provisions in the ABWR design:

- The design ensures all nonessential systems will be automatically isolated by the containment isolation system on receipt of a containment isolation signal.

- Each nonessential system line (except instrument lines) is designed with two isolation barriers in series.

- The design allows resetting the isolation signal without automatically reopening the CIVs. The ABWR system design ensures that resetting the isolation signal will not result in the automatic reopening of the CIVs. The reopening of any CIV is on a valve-by-valve basis once the isolation signal has cleared and following a subsequent logic reset.

- The design utilizes a containment pressure setpoint for initiating containment isolation that is as low as compatible with normal operation. Specifically, GE has committed to a high drywell setpoint pressure of 0.14 kg/cm^{2}g (2 psig) to isolate nonessential penetrations.

- The design includes automatic closing on a high radiation signal for all systems that provide a path to the environs. Specifically, the containment purge and vent isolation valves will close on high radiation levels in the secondary containment HVAC air exhaust or in the fuel handling area HVAC air exhaust.

The staff finds that the containment isolation provisions for the ABWR design meet the requirements of 10 CFR 50.34(f)(2)(xiv).

GE submitted the design description and the ITAAC relating to the containment isolation system. The staff's review of this material was not complete when the DFSER was issued. This was identified in the DFSER as Open Item 6.2.4.1-4. GE subsequently provided a revised design description 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, DFSER Open Item 6.2.4.1-4 is resolved.

The staff has reviewed the information in the SSAR and information submitted in response to staff questions concerning the containment isolation system to ensure conformance to SRP Section 6.2.4 and BTP CSB 6-4. It concludes, as described in the preceding section, that the containment isolation system meets GDC 1, 2, 4, 16, 54, 55, 56, and 57.

6.2.5 Combustible Gas Control in Containment

Following a LOCA, hydrogen may accumulate within the containment as a result of the following phenomena: (1) metal-water reaction between the zirconium fuel cladding and the reactor coolant, (2) radiolytic decomposition of the water in the reactor core and the containment, and (3) corrosion of metals by ECCS spray solutions. If a sufficient amount of hydrogen is generated, it may react with the oxygen present or generated in the containment following an accident. To monitor and control the buildup of hydrogen and oxygen within the containment, GE has incorporated the following systems and capabilities in the ABWR design:

(1) Atmospheric Control System

The atmospheric control system (ACS) is designed to maintain the primary containment oxygen concentration below the maximum permissible limit (3.5 percent) in accordance with RG 1.7, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident," Revision 2, during normal, abnormal, and accident conditions to ensure an inert atmosphere. The containment atmosphere will be inerted using adequately sized nitrogen storage tanks that are provided with makeup capability. The ACS is designed to withstand missiles, pipe whip, flooding, tornados, a safe shutdown earthquake, LOCA environment, and a single active failure. However, GE states that the ACS is non-safety grade, whereas the SRP Section 6.2.5 acceptance criteria to satisfy GDC 41 state that the combustible gas control system should be safety grade because this system is relied on to ensure that containment integrity is maintained following an accident. The staff identified this as Open Item 19 in the DSER (SECY-91-355).

GE stated that the ACS consists of a nitrogen supply, injection, and exhaust lines and the containment overpressure protection system. The safety-related functions normally associated with the ACS will be performed by the safety-related high-pressure nitrogen system (HPIN). The only portions of the ACS that are safety related are the containment penetrations and isolation valves that isolate the ACS from the HPIN. This is acceptable. In addition, the staff's review of the HPIN design is contained in Section 9.3.1 of this report, and the staff finds this HPIN design is acceptable. This resolved DSER Open Item 19.

(2) Containment Atmosphere Monitoring System

The containment atmosphere monitoring system (CAMS), as addressed in SSAR Section 7.6.1.6, is a safety-grade, seismic Category I system designed to meet GDC 41 by monitoring the drywell and suppression chamber for high levels of hydrogen, oxygen, and radiation during accident conditions. The system will allow operators to confirm that the containment is inerted so that containment nitrogen purging can be terminated and also to verify that de-inerting is complete to ensure safe personnel entry into the primary containment. The system has a measurement range of 0 to 25 percent (by volume) at 100-percent relative humidity.

The CAMS consists of two independent but redundant Class 1E divisions, which are electrically and physically independent to remain operable as a result of a single active failure coincident with a loss of offsite power. The system has two subsystems: radiation monitoring subsystem and hydrogen/oxygen monitoring subsystem. The radiation monitoring subsystem contains two channels per division; one for monitoring the drywell and the other for monitoring the suppression chamber. Each monitoring channel consists of an ionization chamber, a log rate meter, and a recorder. Each channel has a range of 1 to 10E+7/R hr and will initiate an alarm in the control room on high radiation level or on system failure. This subsystem can be initiated automatically on a LOCA signal or manually actuated from the control room. The radiation monitors do not provide any bypass or interlock capability.

The hydrogen/oxygen monitoring subsystem consists of two divisions that will take samples of the drywell and wetwell and feed them to monitors for measurement, recording, and control room alarm. The piping is stainless steel and continuously heat traced to remove moisture from the system during measurements. Gas calibration racks are provided to perform equipment calibrations during operating conditions. A thermal delay bypass feature will allow system testing during power operation. There are no automatic isolation functions associated with the system.

Each CAM subsystem will be powered from a divisional 120-V ac Class 1E instrument bus. This power source also will supply the heat-tracing blanket used for the sampling lines.

The staff concludes that the CAMS has adequate capability for monitoring the containment atmosphere for hydrogen/oxygen control, thus satisfying GDC 41 regarding containment atmosphere cleanup.

(3) Capability of Post-LOCA Purging of the Containment

Post-LOCA primary containment backup purging capability is required in accordance with RG 1.7 as an aid for containment atmosphere cleanup following a LOCA. During normal plant operation, the bleed line will function, in conjunction with the nitrogen purge line, to maintain primary containment pressure at about 5.2 kPag (0.75 psig) and oxygen concentration below 3.5 percent by volume. This will be done by making up the required quantity of nitrogen in the primary containment through the makeup line or relieving pressure through the bleed line. Flow through the bleed line will be directed through either the SGTS or the reactor building secondary containment HVAC system and will be monitored for radiation. However, GE provided neither the purge rate that would be required to maintain the oxygen concentration below 3.5 percent by volume nor the radioactive consequence analysis for the staff to review. This was identified as Open Item 6.2.5-1 in the DFSER and as Open Item 20 in the DSER (SECY-91-355).

GE stated that postaccident containment backup purging capability is not needed to maintain the oxygen concentration below 3.5 percent because the containment is inerted. Pressure in the containment during normal operation will be maintained by a 5 cm (2 in.) nitrogen supply line and a 5 cm (2 in.) vent line. No continuous purging is required during normal operation. The staff finds this acceptable. This resolved DFSER Open Item 6.2.5-1.

With respect to postaccident hydrogen generation analysis, GE stated that the analytical model described in GE report NEDO-22155, "Generation and Mitigation of Combustible Gas Mixtures in Inerted BWR Mark I Containment" (nonproprietary) was used to compute hydrogen and oxygen generation from radiolysis. NEDO-22155 was reviewed by the staff for the EPRI requirements document certification and found unacceptable. As a result, GE performed a new hydrogen generation analysis using the RG 1.7 methodology. The results of the analysis are provided in Appendix E of SSAR Chapter 19. On the basis of its review of GE's analysis, the staff concludes that the ABWR will be able to withstand a 100-percent fuel-clad metal-water reaction.

(4) Flammability Control System (Hydrogen Recombiner)

SSAR Section 6.2.5.2.7 states that there will be two permanently installed recombiners in the secondary containment. However, GE did not provide information on dedicated redundant containment penetrations to demonstrate that the recombiners could perform their safety function assuming a single active failure. Also, GE did not indicate whether the recombiners are safety grade. This was identified as Open Item 6.2.5-2 in the DFSER and as Open Item 21 in the DSER (SECY-91-355).

In Amendment 34, GE revised SSAR Section 6.2.5.2.7 which states, in part, that the flammability control system (FCS) consists of two permanently installed safety-related thermal hydrogen recombiners located in separate rooms in the secondary containment and controlled from the main control room. Independent drywell and suppression chamber penetrations are provided for the two recombiners. Each penetration has two normally closed isolation valves. The staff has reviewed GE's FCS design and finds that the recombiner configuration meets the single-failure criterion and is acceptable. Therefore, DFSER Open Item 6.2.5-2 is resolved.

10 CFR 52.47(a)(1)(ii) requires that an application for a standard design certification include a demonstration of compliance with any technically relevant portions of the TMI requirements set forth in 10 CFR 50.34(f). The staff's evaluation of the TMI-related requirements on hydrogen control is provided as follows:

(a) 10 CFR 50.34(f)(1)(xii), "Evaluation of Alternative Hydrogen Control Systems," requires the applicant to perform an evaluation of alternative hydrogen control systems that would satisfy the requirements of 10 CFR 50.34(f)(2)(ix). At a minimum, consideration should be given to a hydrogen ignition and postaccident inerting system and the evaluation should include

- a comparison of costs and benefits of the alternative systems considered
- analyses and test data to verify compliance with the requirements of 10 CFR 50.34(f)(2)(ix) for the selected system
- preliminary design descriptions of equipment, function, and layout for the selected systems

GE has provided information for the last two items as identified above but has not provided cost and benefit information for alternative systems. If the alternative systems are used in the design, this information must be provided by the COL applicant. This was identified as COL Action Item 6.2.5-1 in the DFSER.

In its submittal of March 5, 1993, GE added Section 6.2.7 to the SSAR. SSAR Section 6.2.7.1 (Amendment 32), regarding alternative hydrogen control, states that the COL applicant will provide a comparison of costs and benefits for alternative hydrogen control in accordance with SSAR Section 6.2.5. Therefore, COL Action Item 6.2.5-1 has been clarified and found to be acceptable. The staff concludes that the ABWR design meets the requirements of 10 CFR 50.34(f)(1)(xii).

(b) 10 CFR 50.34(f)(2)(ix), "Hydrogen Control System Preliminary Design," requires the applicant to provide a system for hydrogen control that can safely accommodate hydrogen generated by the equivalent of a 100-percent fuel-clad metal-water reaction. This system and any associated systems should 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 amount of hydrogen that would be equivalent to that generated from a 100-percent fuel-clad metal-water reaction, or that the post-accident atmosphere will not support hydrogen combustion
- 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
- equipment necessary for achieving and maintaining safe shutdown of the plant and maintaining containment integrity will perform its safety function during and after its exposure 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

In SSAR Section 19B.2.18, GE states that the ABWR containment will have an inert atmosphere and will be able to withstand the pressure and energy addition from a 100-percent fuel-clad metal-water reaction. The staff concludes that the inerted containment and the provision for permanently installed hydrogen recombiners are acceptable as hydrogen control measures and adequately address 10 CFR 50.34(f)(2)(ix).

(c) 10 CFR 50.34(f)(3)(iv) requires the applicant to provide one or more dedicated containment penetrations, equivalent in size to a single 0.9 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 a followup of one of the requirements identified under TMI action plan, Item II.B.8 of NUREG-0660.

The staff's evaluation of the size of the penetration that can be used for venting the containment is provided in Section 20.5.44 of this report.

(d) 10 CFR 50.34(f)(3)(vi), "Dedicated Hydrogen Penetrations (Hydrogen Recombiners)," requires the applicant to provide 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 Figure 6.2-40 shows redundant dedicated hydrogen recombiner penetrations. This satisfies 10 CFR 50.34(f)(3)(vi). However, in the DFSER, the staff stated that GE had not provided design information for the recombiners in the SSAR, but had committed to include this information in a future SSAR amendment. This was identified as Confirmatory Item 6.2.5-2 in the DFSER.

In Amendment 32, GE provided this information in SSAR Section 6.2.5.2.7. The staff reviewed this section and P&ID Figure 6.2-40 (Amendment 34) and finds that GE has addressed the FCS design requirements in the SSAR. This is acceptable. Therefore, DFSER Confirmatory Item 6.2.5-2 is resolved.

(e) 10 CFR 50.34(f)(2)(xv), "Containment Purging/Venting," requires the applicant to provide a capability for containment purging/venting designed to minimize purging time consistent with ALARA (as low as is reasonably achievable) principles for occupational exposure. It also requires assurance that the purge system will reliably isolate under accident conditions.

In SSAR Section 19A.2.27, GE states that during normal operation, all large valves in containment ventilation lines will be closed and only small, 5 cm (2 in.), nitrogen makeup valves will be opened. These are air-operated valves with rapid closure times that prevent substantial releases from the containment in the event of a transient requiring containment isolation. GE also states that the 5 cm (2 in.) nitrogen bleed lines will be sufficient to maintain normal containment pressure during normal operation when used in conjunction with containment spray and the drywell cooling system. However, SSAR Figure 6.2-39 shows that T31-F007 and T31-F010 (36 cm (14 in.) valves) also will be open during normal operation. In the DFSER, the staff requested GE to correct this discrepancy. This was identified as Confirmatory Item 6.2.5-3.

In its submittal of March 5, 1993, GE revised SSAR Section 19A.2.27 to state, in part, that all large valves in containment ventilation lines will be closed during normal operation with the exception of two large valves in the containment overpressure protection system (COPS) where flow will be prevented by rupture discs in the piping. Furthermore, in a submittal dated May 7, 1993, GE revised SSAR Table 3.9-8 to specify that valves T31-F007 and T31-F010 will be leakage rate tested every 2 years during refueling outages in accordance with the requirements of the of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code). The staff finds that this discrepancy is clarified, and is acceptable. Therefore, DFSER Confirmatory Item 6.2.5-3 is resolved.

In SSAR Section 19A.3.3, GE states that a testing program will be provided to ensure that the large ventilation valves will close within the limits ensured in the radiological design bases. In the DFSER, the staff stated that the test program should include valves T31-F007 and T31-F010 and that the COL applicant should submit details of these tests. This was identified as COL Action Item 6.2.5-1 in the DFSER.

In response, GE stated that valves T31-F007 and -F010 will normally be open and will be closed every 2 years for leakage rate testing in accordance with the requirements as addressed in SSAR Table 3.9-8, "In-service Testing of Safety Related Pumps and Valves." The staff finds that these valves are passive pressure control valves, not the primary containment isolation valves. Containment integrity is preserved by the rupture discs downstream of the valves. These valves are subject to leak rate tests in accordance with ASME/American National Standards Institute (ANSI) OM-1987, Part 10, to verify their normally open position and their capability to close. Therefore, these valves should not be included in the ventilation valve testing program. On the basis of this review, COL Item 6.2.5-1 is deleted and need not be included in the SSAR.

In response to a request for additional information (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 (55-cm (22-in.)) purge lines can be open. GE has made this modification. This was DFSER TS Item 6.2.5-1 and is resolved.

The staff finds that the methods used to maintain containment pressure without backup purging capability meet the requirements of 10 CFR 50.34(f)(2)(xv).

Three RAI questions (Q430.45, Q430.46, and Q430.47) on the combustible gas control system were transmitted to GE. These questions dealt with the subjects of scope and interface, compliance with RG 1.7, and BTP ASB 9-2 of the SRP regarding hydrogen and oxygen production and accumulation. GE has responded to all of these questions and amended SSAR Section 6.2.5. GE states in that section that the combustible gas control systems, consisting of the FCS and atmospheric control system, are completely within the scope covered by the ABWR SSAR and that there are no interfaces with equipment or systems outside the scope of this submittal. GE also states that the analysis of hydrogen and oxygen production is based on the parameters listed in RG 1.7. The fission product decay energy model used is that presented in SRP Section 9.2.5, BTP ASB 9-2, and inputs to the analysis are provided in SSAR Section 6.2.5.3. The staff finds that GE's response and the SSAR as amended adequately address the issues identified in the staff's RAIs.

Generic Letter (GL) 89-16, "Installation of Hardened Wetwell Vent," addressed the need for modifications of BWR containment designs to reduce their vulnerability to severe accident challenges. The staff finds that the ABWR design has included the containment overpressure protection system which addresses this GL. The ABWR design for severe accident conditions is evaluated in Section 19.2 of this report.

The staff concludes that the design of the combustible gas control systems including the containment ACS and FCS are acceptable and meet the requirements of 10 CFR Part 50, Sections 50.44 and 50.34 and GDC 41, 42, and 43 based on the following:

(1) GE has resolved all the open issues on combustible gas control. The design meets the requirements of 10 CFR 50.44 with respect to means for controlling hydrogen and capability for measuring hydrogen concentration and inerting in the containment.

(2) GE has demonstrated compliance with all technically relevant portions of the TMI requirements set forth in 10 CFR 50.34(f).

(3) GE has met the requirements of GDC 41 with respect to systems being provided to control hydrogen and oxygen concentration in the containment following postulated accidents, GDC 42 with respect to periodic inspection of the systems, and GDC 43 with respect to periodic testing of the systems.

(4) The design of the combustible gas control system meets the acceptance criteria set forth in SRP Section 6.2.5 and the limitation of hydrogen and oxygen concentration specified in RG 1.7.

6.2.6 Containment Leakage Testing

The staff reviewed GE's containment leakage testing program described in the SSAR for compliance with the containment leakage testing requirement in Appendix J to 10 CFR Part 50 (Appendix J). Such compliance provides adequate assurance that the containment leaktight integrity can be verified throughout the service lifetime and that the leakage rates will be checked periodically during service on a timely basis to maintain such leakage within the specified limits. Maintaining containment leakage within limits provides reasonable assurance that, if any radioactivity is released within the containment, the loss of the containment atmosphere through potential leak paths is not in excess of the limits specified for the site.

The staff reviewed the containment leakage testing program to ensure that the containment penetrations and system isolation valve arrangements are designed to satisfy the containment integrated and local leakage rate testing requirements of Appendix J.

Type A Test

The preoperational containment integrated leakage rate (Type A) test is intended to measure the primary containment overall integrated leakage rate after the containment has been completed and is ready for operation and the local leak rate tests (LLRTs) of all mechanical, fluid, electrical, and instrumentation systems penetrating the containment pressure boundary have been performed.

The objectives of the initial integrated leakage rate test (ILRT) are to

- verify that the containment integrated leakage rate does not exceed the containment design-basis accident DBA leakage rate, L_a, which is 0.5 percent by weight of the containment atmosphere in 24 hours, at peak containment accident pressure, P_a related to DBA
- establish a minimum allowable leakage rate, L_t, at reduced pressure, P_t, which is used during subsequent ILRTs
- obtain data that may be used to develop the leakage rate characteristics and history of the containment system

- demonstrate by a verification test the accuracy of the integrated leakage rate instrumentation to satisfactorily determine the containment integrated leakage rate

The preoperational test will be performed at both the reduced pressure, P_t, and the peak containment accident pressure, P_a. P_t will be chosen so that it is greater than 0.5 P_a. After the initial ILRT, a set of three Type A tests will be performed at approximately equal intervals during each 10-year service period with the third test of each set coinciding with the end of each 10-year major inservice inspection shutdown. The total measured containment leakage rate, L_{tm}, at reduced pressure, P_t, should not exceed 0.75 L_t as established by the initial ILRT. The accuracy of the leakage rate tests will be verified by using a supplemental method of leakage measurement.

In conducting a Type A test, certain systems that are normally filled with water and operating under post-LOCA conditions need not be vented to the containment atmosphere. In addition, systems required to function during the Type A test should be operable in their normal mode and need not be vented, but the test results for such systems should be added to the Type A test total. GE has confirmed that the leakage test values of such system lines will be added to the Type A test results. All other system lines will be vented or drained before the Type A test. However, during its initial review, the staff found that although GE's Type A test program was acceptable, GE had not identified the systems or the reasons why these systems would not be vented or drained during the ILRT. This issue was identified as a part of Open Item 27 in the DSER (SECY-91-153).

In response to this concern, GE stated that SSAR Section 6.2.6.1.3 provides additional criteria for the integrated leak rate test, thus addressing this issue. The staff reviewed SSAR Section 6.2.6.1.3 and finds that the criteria for system lines not to be vented or drained during a Type A test are acceptable. This part of DSER Open Item 27 is resolved.

In reviewing the certified design material for the primary containment system, the staff found that the design description and ITAAC stated that the main steam isolation valve (MSIV) leakage was not included in the primary containment allowable leakage of 0.5 percent. GE stated that the analytical predictions of radiological consequences in the SSAR had been based on the assumption of a containment leakage of 0.5 percent per day plus a separate MSIV leakage. The next revision of the SSAR was to clarify that MSIV leakage is to be excluded from the 0.5 percent per day. The staff reviewed SSAR Section 6.2.1.1.2 and finds that this issue has been adequately clarified.

<u>Type B Tests</u>

Containment penetrations whose designs incorporate resilient seals, gaskets, or sealant compounds; piping penetrations fitted with expansion bellows serving as the containment boundary; airlock door seals; equipment and access doors with resilient seals; and other testable penetrations are to be Type B tested during preoperational testing and thereafter at periodic intervals during the lifetime of the unit in accordance with Appendix J. The Type B tests are necessary to ensure the continuing integrity of the penetrations.

To facilitate LLRT, GE has proposed a permanently installed system consisting of a pressurized gas source (nitrogen or air) and the manifolding and valving necessary to subdivide the testable penetrations into groups of two to five. Each group will then be pressurized, and if any leakage is detected (by pressure decay or flow meter), individual penetrations can be isolated and tested until the source and nature of the leak is determined.

GE states, in SSAR Section 6.2.6, that the combined local leakage rate of all components subject to Type B and Type C tests (described in subsequent paragraphs) will not exceed 60 percent of L_a. Type B tests will be performed at peak containment accident pressure during each reactor shutdown for major fuel reloading, but in no case at intervals greater than 2 years. Airlocks will be tested at initial fuel loading and at least once every 6 months thereafter. Additionally, whenever they are opened during periods when containment integrity is required, they will be tested within 3 days of opening in accordance with Appendix J. These tests are required for the airlocks since they contain inflatable seals.

In the DSER, the staff stated that because the intent of the Appendix J testing program has never been to require a forced reactor shutdown just to conduct these tests within preset test intervals, GE would either have to clarify whether provisions for conducting all Type B tests at power exist in the ABWR design or request an exemption from the requirement for conducting Type B tests at 2-year intervals and justify the request. Also, GE had not provided (1) the acceptance criteria for testing the airlocks, (2) a list of all containment penetrations that are subject to Type B tests, and (3) a list of all penetrations that are excluded from Type B tests (if any) and the rationale for such exclusions. In the DSER, the staff stated that it could not conclude that the proposed Type B testing program for the ABWR was acceptable. These issues were identified as part of Open Item 27 in the DSER (SECY-91-153).

GE subsequently provided information on Type B test criteria and listed all containment penetrations in SSAR

Table 6.2-8. The staff finds that the Type B test criteria specified in SSAR Section 6.2.6 comply with Appendix J and SRP Section 6.2.6. SSAR Section 6.2.6.2 specifies that containment penetrations whose designs incorporate resilient seals, bellows, gaskets, or sealant compounds, airlocks and lock door seals, equipment and access hatch seals, and electric canisters, and other such penetrations are to be leak tested during preoperational testing and at periodic intervals thereafter in conformance with Type B leakage rate tests defined in Appendix J. SSAR Section 6.2.6.4 states that Type B and C tests may be conducted at any time during normal plant operations or during shutdown periods, as long as the time interval between tests for any individual Type B or C test does not exceed 2 years. The staff reviewed SSAR Section 6.2.6 and SSAR Table 6.2-8 and finds that the Type B test requirements are acceptable. This part of DSER Open Item 27 is resolved.

Type C Tests

All primary containment isolation valves whose seats are exposed to the containment atmosphere after a LOCA will be Type C tested pneumatically with air or nitrogen at P_a. Valves that are sealed by water will be leak tested with water as the test medium. The test pressure will be applied in the same direction as when the valve is required to perform its safety function, unless it can be shown that results from tests with pressure applied in a different direction are equivalent or more conservative. Type C testing will be performed by local pressurization, using either the pressure decay method (for pneumatic testing) or the flowmeter method (for both pneumatic and hydrostatic testing).

SSAR Section 6.2.6.3.1 states that Type C tests will be performed on all containment isolation valves required by Appendix J. All testing will be performed pneumatically, except that hydraulic testing might be performed on isolation valves using water as a sealant provided the valves were demonstrated to exhibit leakage rates that did not exceed those in the ABWR technical specifications. However, the SRP states that Type C testing with water is permissible only if the system line for the valve is not a potential containment atmosphere leak path. This was identified as Open Item 6.2.6-1 in the DFSER.

In response to the staff's concern, GE revised SSAR Section 6.2.6.3.1 (Amendment 21) to state that all testing will be performed pneumatically, except that hydraulic testing using water as a sealant may be performed during isolation valve Type C tests provided the system line for the valve is not a potential containment atmosphere leak path. The staff finds the statement acceptable and DFSER Open Item 6.2.6-1 is resolved.

During its initial review, the staff found that GE had not adequately responded to the staff's RAI of July 7, 1988, on a number of issues. These issues included (1) Type C test interval, (2) test pressure for main steam isolation valves, (3) test methodology for ECCS isolation valves, (4) test procedures for valves not covered by Appendix J procedures and a list of such valves, (5) a list of all primary containment isolation valves that will be Type C leak tested, (6) a list of all valves that will be hydrostatically tested and the test pressure, and (7) a list of all valves that will be tested in the reverse direction and the justification for such testing. Additionally, it was not clear whether lines that contain valves that do not have 30-day water-leg seals will be drained and the valves then pneumatically tested as required. These were identified as part of Open Item 27 in the DSER (SECY-91-153).

GE's response and the staff's evaluation of these issues follows:

(1) SSAR Section 6.2.6.4 states that Type B and C tests may be conducted at any time during normal plant operations or during shutdown periods, as long as the time interval between tests for any individual Type B or C test does not exceed 2 years. GE has specified a Type C test interval. The staff finds this acceptable, and this part of DSER Open Item 27 is resolved.

(2) SSAR Section 6.2.6.3.1 specifies that MSIVs and isolation valves isolated from a sealing system are to use a pressure of at least P_a. The staff finds that the test pressure meets Appendix J and is acceptable. This part of DSER Open Item 27 is resolved.

(3) GE did not address test methodology for ECCS isolation valves. SSAR Section 6.2.6.3 does not mention this issue. This was identified as Open Item 6.2.6-2 in the DFSER.

    GE stated that the test information is provided in revised SSAR Table 6.2-7. The staff reviewed SSAR Table 6.2-7 and finds that the test requirements for the ECCS isolation valves are addressed in its associated notes g, h, i, j, k, and n. This is acceptable. DFSER Open Item 6.2.6-2 is resolved.

(4) GE did not provide test procedures for valves not covered by Appendix J and a list of such valves. This was identified as Open Item 6.2.6-3 in the DFSER.

GE stated that the test requirements for isolation valves not covered by Appendix J are addressed in SSAR Section 3.9.6.2 (inservice testing of safety-related valves), SSAR Section 3.9.7.3 (pump and valve inservice testing), and SSAR Table 3.9-8. These valves are subject to ASME Code, Section XI leak rate tests. The staff's evaluation of these valves is provided in Chapter 3 of this report. In response to RAI Q430.50h, GE stated that those valves not specifically Type C tested will be tested as part of the Type A test. GE has addressed the test requirements for the valves not covered by Appendix J. The staff finds this acceptable. Therefore, DFSER Open Item 6.2.6-3 is resolved.

(5) GE listed all containment isolation valves that require Type C testing in SSAR Table 6.2-7. The staff finds that the test provisions for the values listed in the table meet Appendix J and are acceptable. This part of DSER Open Item 27 is resolved.

(6) GE did not list all the valves subject to hydrostatic test as addressed in SSAR Section 6.2.6.3.1. This was identified as Open Item 6.2.6-4 in the DFSER, which was part of Open Item 27 in the DSER (SECY-91-153).

GE stated that the test requirements are provided in revised SSAR Table 6.2-7. The staff reviewed SSAR Table 6.2-7 (Amendment 34) and finds that GE has identified, in notes d, g, h, j, k, n, and q, those isolation valves that will be filled with water following a LOCA. SSAR Section 6.2.6.3.1 (Amendment 34) states that these valves are in lines designed to be, or remain, filled with a liquid for at least 30 days following a LOCA and will be leakage rate tested with that liquid. The liquid leakage measured will not be converted to equivalent air leakage or added to the Type B and C test total. The staff finds that the hydrostatic test criterion meets Appendix J and is acceptable. DFSER Open Item 6.2.6-4 is resolved.

(7) GE did not justify and list the valves to be tested in the reverse direction. This was identified as a part of Open Item 27 in the DSER (SECY-91-153).

GE stated that SSAR Table 6.2-7, note e, provides the criterion for valves to be tested in the reverse direction and the valves listed in the table with that note will be tested in the reverse direction. The staff finds that the test requirement complies with Criterion III.C.1 of Appendix J and is acceptable. This part of DSER Open Item 27 is resolved.

(8) GE did not specify whether lines that contain valves that do not have 30-day water-leg seals will be drained before the valves are pneumatically tested. This was identified as Open Item 6.2.6-5 in the DFSER and as part of Open Item 27 in the DSER (SECY-91-153).

GE stated that the valves that do not have 30-day water-leg seal are indicated in SSAR Table 6.2-7 by note (u) in the Type C test requirements entry. The staff verified that the isolation valves to be pneumatically tested are indicated by note (u) in the test requirements entry of the table, and finds that they are acceptable. DFSER Open Item 6.2.6-5 is resolved.

During its initial review of SSAR Section 6.2.6 regarding containment leakage testing, the staff also identified the following open items in the DSER (SECY-91-153).

(1) GE did not indicate whether the test, vent, and drain connections used to facilitate ILRTs and LLRTs will be kept closed and under administrative control during normal plant operations and whether they would be subject to periodic surveillance testing to ensure their integrity and to verify the effectiveness of administrative controls. In the DFSER the staff stated that SSAR Section 6.2.6.3.1 did not address its concern. This issue was identified as Open Item 6.2.6-6 in the DFSER.

In its submittal of March 5, 1993, GE added Section 6.2.7.2 (Amendment 32) to the SSAR to state that the COL applicant will maintain the primary containment boundary by administrative controls in accordance with SSAR Section 6.2.6.3.1. SSAR Section 6.2.6.3.1 (Amendment 34) states, in part, that all test connections, vent lines, or drain lines consisting of double barriers (e.g., two valves in series, one valve and a cap, or one valve and a flange), which are connected between isolation valves and form a part of the primary containment boundary, need not be Type C tested because of their infrequent use and multiple barriers as long as the barrier configurations are maintained using an administrative control program. These lines will be surveillance inspected at cold shutdown and at 31-days intervals (internal and external to the primary containment, respectively) as required by the TS. The staff finds that these test connections are designed with multiple barriers, are under an administrative control program, and will be surveillance inspected so that any leakage can be detected. In addition, Appendix J does not specify test requirements for these lines. This is acceptable. DFSER Open Item 6.2.6-6 is resolved.

(2) GE relies on closed loops outside the containment as containment isolation barriers for some engineered safety feature (ESF) system containment penetrations. The staff's position is that a closed loop outside the containment that meets the criteria of Section 3.6 of ANSI/American Nuclear Society 56.2-1976 can be considered a second containment isolation barrier, thereby eliminating the need for a second containment isolation valve at the penetration. However, each barrier (i.e., the single isolation valve at each penetration and closed piping loop outside the containment) is subject to leak rate testing. Inclusion of this position in the SSAR was identified as Open Item 6.2.6-7 in the DFSER.

In its submittal of April 26, 1993, GE stated that there are two closed-loop systems in the ABWR design that provide an extension of the containment. These are the containment atmosphere monitoring system and the fuel pool cooling system. These systems will be leak tested by opening their lines to the containment atmosphere during the containment ILRT. The staff finds that their test requirements have been specified in SSAR Table 6.2-7 (Amendment 32), and are acceptable. DFSER Open Item 6.2.6-7 is resolved.

(3) In response to the staff's RAI Q430.52b of July 7, 1988, GE submitted information regarding secondary containment inleakage and potential bypass leakage paths. However, it did not indicate whether the bypass leakage paths will be leak tested as specified in BTP CSB 6-3, "Determination of Bypass Leakage Paths in Dual Containment Plants," of the SRP. The staff found that SSAR Table 6.2-10, Table 6.5-2, and Section 6.5.1.3.2 did not address this issue adequately. Clarification of this issue was identified as Open Item 6.2.6-8 in the DFSER.

In response to this issue, GE identified all the potential leakage paths that could bypass the secondary containment in revised SSAR Table 6.2-10. The staff reviewed SSAR Table 6.2-10 (Amendment 34) and finds that the potential bypass paths are the main steamlines, feedwater lines, drywell purge suction and exhaust lines, and wetwell purge suction and exhaust lines. The staff noted that the leakage rate through the main steam isolation valves (MSIVs) has been considered as the secondary containment bypass leakage rate and has been treated separately in the offsite dose analysis. Leakage through feedwater isolation valves (FWIVs) will be monitored by Type C test. Leakage from the purge lines also will be monitored by Type C test and can be detected by the radiation detectors at the purge and vent exhaust lines. In addition, SSAR Table 6.5-2, regarding source terms used for the standby gas treatment system (SGTS) charcoal adsorber design, specifies that leakage rates assumed for calculation are 0.5 percent per day for the primary containment and 50 percent per day for the secondary containment.

Additionally, primary containment leakage could circumvent the secondary containment and bypass the leakage collection and filtration systems. This leakage is generally not quantifiable through LLRTs, but must be considered in the radiological consequence analysis of a DBA. SSAR Section 6.5.1.3.2 (Amendment 34) states that, for the ABWR dose analysis, direct transport of containment leakage to the environment was assumed for the first 20 minutes after initiation of a LOCA (in addition to the leakage through the MSIVs). Each SGTS fan is sized to establish a continuously negative pressure within 10 minutes after SGTS initiation.

The staff finds that GE has provided all the necessary measures to control bypass leakage in the ABWR design in conformance with BTP CSB 6-3 and provide acceptable margins in the offsite dose analysis. This is acceptable. DFSER Open Item 6.2.6-8 is resolved.

(4) GE did not provide procedures for factoring potential contributions from the hydrogen recombiners into the ILRT results. However, the staff considered this to be a COL action item. In the DFSER, the staff stated that it would ensure that COL applicants factored the potential contributions from the hydrogen recombiners into the ILRT results in accordance with SRP Section 6.2.6. This was identified as COL Action Item 6.2.6-1.

In response to this concern, GE added note (v) in SSAR Table 6.2-7 (Amendment 32) to state that the flammability control system (FCS) is a closed-loop, safety-grade system required to be functional following an accident. Whatever leakage (if any) will be returned to the primary containment. In addition, during an ILRT, these valves will be opened and the lines will be subjected to Type A tests. The staff finds that any leakage from the penetrations or the lines connected to the hydrogen recombiners will be included in the ILRT results. Therefore, COL Action Item 6.2.6-1 is deleted and will not be included in the SSAR.

In Bulletin 82-04, "Deficiencies in Primary Containment Electrical Penetration Assemblies," was identified as a candidate operating experience issue. This bulletin addresses potential generic safety concerns related to electrical penetration assemblies supplied by the Bunker Ramo Company. However, the assemblies are no longer manufactured by Bunker Ramo and are not used in the ABWR design. Therefore, this operating experience issue is not applicable to the ABWR design.

GE submitted the design description and the ITAAC relating to containment leakage testing. The staff's review of this material was not complete when the DFSER was issued. This was identified in the DFSER as Open Item 6.2.6-9. GE subsequently provided a revised design description 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, DFSER Open Item 6.2.6-9 is resolved.

On the basis of this review, the staff concludes that the ABWR containment leakage testing program complies with Appendix J to 10 CFR Part 50. Such compliance provides adequate assurance that leaktight integrity of the containment can be verified periodically throughout its service life to ensure that leakage rates are maintained within the limits of the TS. Maintaining containment leakage rates within such limits provides reasonable assurance that, in the event of any radioactivity releases within the containment, the loss of the containment atmosphere through the leak paths will not be in excess of the acceptable limits specified for the site. Compliance with Appendix J as described in this section constitutes an acceptable basis for satisfying the requirements of GDC 52 with respect to the capability for containment leakage rate testing, the requirements of GDC 53 with respect to provisions for containment testing and inspection, and the requirements of GDC 54 with respect to the capability for detecting piping leakages.

6.2.7 Fracture Prevention of Containment Pressure Boundary

The staff reviewed the ABWR measures involving fracture prevention of ferritic materials used in the containment pressure boundary in accordance with SRP Section 6.2.7. Containment pressure boundary ferritic materials are acceptable if they meet the requirements of GDC 51 as it relates to the reactor containment pressure boundary being designed with sufficient margin to assure that under operating, maintenance, testing, and postulated accident condition the ferritic materials will behave in a nonbrittle manner and the probability of rapidly propagating fracture is minimized.

The primary containment vessel of the ABWR is a reinforced-concrete structure with ferritic parts (the removable head, personnel locks, equipment hatches, and penetrations), which will be made of material that has a nilductility transition temperature, RT_{NDT}, of at least -17 °C (30 °F) below the minimum service temperature. This meets the requirements of GDC 51. GDC 51 is only applicable to parts of the containment that are to be made of ferritic materials.

In RAI Q251.12, the staff requested that GE clarify the applicability of GDC 51 because it appeared that GE intended that GDC 51 be applied to the concrete portion of the containment. GE responded that GDC 51 is applicable to the removable drywell head, personnel locks, equipment hatches, and penetrations, which will be made of ferritic materials. GE responded satisfactorily to the staff's request and revised SSAR Section 3.1.2.5.2.2 accordingly. Therefore, GE's commitment to GDC 51 for the items listed above in the containment design meets SRP Section 6.2.7 and is acceptable.

6.2.8 Severe Accident Considerations

GE addresses containment performance during severe accidents in SSAR Chapter 19, and the staff's review is documented in Chapter 19 of this report.

6.3 Emergency Core Cooling System

The staff reviewed the emergency core cooling system (ECCS) in accordance with SRP Section 6.3. The staff acceptance criteria are based on meeting the relevant requirements of the following regulations:

(1) GDC 2 as it relates to the seismic design of structures, systems, and components (SSC) whose failure could cause an unacceptable reduction in the capability of the ECCS to perform its safety function. Acceptability is based on meeting Position C2 of RG 1.29.

(2) GDC 4 as related to dynamic effects associated with flow instabilities and loads (e.g., water hammer).

(3) GDC 5 as it relates to SSC important to safety shall not be shared among nuclear power units unless it can be demonstrated that sharing will not impair their ability to perform their safety function.

(4) GDC 17 as it relates to the design of the ECCS having sufficient capacity and capability to assure that specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded and that the core is cooled during anticipated operational occurrences and accident conditions.

(5) GDC 27 requires that the reactivity control systems have a combined capability, in conjunction with poison addition by the emergency core cooling system, or reliably controlling reactivity changes under postulated accident conditions, with appropriate margin for stuck rods.

(6) GDC 35, 36, and 37 as they relate to the ECCS being designed to provide an abundance of core cooling to transfer heat from the core at a rate so that fuel and clad damage will not interfere with continued effective core cooling, to permit appropriate periodic inspection of important components, and to permit appropriate periodic pressure and functional testing.

(7) 10 CFR Part 5, Subsection 50.46, and Appendix K to 10 CFR Part 50 as it relates to the ECCS being designed so that its cooling performance is in accordance with an acceptable evaluation model.

The ECCS is designed to provide coolant inventory to the reactor coolant system in the event of a loss-of-coolant accident (LOCA) in the pressure boundary. The ECCS capability extends to failures as large as a double-ended rupture of the largest piping carrying water or steam, and spurious safety/relief valve operation.

6.3.1 System Description

The ECCS consists of the following:

- reactor core isolation cooling (RCIC) system
- high-pressure core flooder (HPCF) system
- automatic depressurization system (ADS)
- low-pressure flooder (LPFL) system

Unlike that in current BWR designs, the RCIC system in an ABWR design is a part of the ECCS. The initiation logic is diversified by adding a high drywell pressure input as well as maintaining the typical system initiation on reactor pressure vessel (RPV) level 2.

The RCIC system is a high-pressure reactor coolant makeup system that will start independently of the ac power supply. The system will provide sufficient water to the reactor vessel to cool the core and to maintain the reactor in a standby condition if the vessel becomes isolated from the main condenser and experiences a loss of feedwater flow. The system also is designed to maintain reactor water inventory, in the event of a loss of normal feedwater flow, while the vessel is depressurized to the point where the residual heat removal (RHR) system can function in the shutdown cooling mode at a reactor pressure of 1,034 kPa (150 psig).

The RCIC system consists of a steam-driven turbine-pump unit and associated valves and piping capable of delivering 3,028 L/min (800 gpm) of makeup water to the reactor vessel through the feedwater system. Fluid removed from the reactor vessel following a shutdown from power operation normally will be made up by the feedwater system and supplemented by inleakage from the control rod drive system. If the feedwater system becomes inoperable, the RCIC system will start automatically when the water level in the reactor vessel reaches the level 2 (L2) trip set point or drywell pressure reaches the high drywell pressure trip set point. The RCIC system also can be started by the operator from the control room. The system is capable of delivering rated flow within 29 seconds of initiation. Primary water supply for the RCIC system will be from the condensate storage tank (CST), and a secondary supply will be from the suppression pool.

A detailed evaluation of the RCIC system is given in Section 5.4.6 of this report.

The HPCF system will maintain the reactor vessel water level above the top of the active core in the event of a break of a 2.54 cm (1 in.)-diameter pipe or smaller and will provide cooling in the event of large-pipe breaks. Actuation of the HPCF system will not require the depressurization of the reactor vessel. The HPCF system consists of two loops. Each loop includes a single motor-driven centrifugal pump that will take suction from the CST or the primary containment suppression pool. An automatic switching feature is based on indication of low CST level. The HPCF flow rate is dependent on the reactor pressure. SSAR Table 6.3-1 states that the rated HPCF flow of 12,113 L/min (3,200 gpm) will be attained at a reactor pressure of approximately 689 kPa (100 psig), which is consistent with accident analysis assumptions. The HPCF system is designed to operate from normal offsite auxiliary ac power or from the emergency diesel generators. Each HPCF pump will be powered from a different diesel generator. The system will initiate automatically by either low water level 1.5 or high drywell pressure signals. The system also can be placed in operation manually from the main control room and the remote shutdown panel (loop B only).

If the RCIC and HPCF systems do not maintain the reactor water level, the ADS will reduce the reactor pressure so that flow from the RHR system, which will be operating in the LPFL mode, enters the reactor vessel in time to cool the core and limit fuel cladding temperature.

Of the 18 nuclear system pressure relief valves, eight ADS relief valves will relieve high-pressure steam to the suppression pool. The evaluation of the pressure relief valves is given in Section 5.2.2 of this report. The ADS will be actuated when the following conditions are satisfied: (1) drywell high pressure, (2) reactor low water level (level 1), and (3) a permissive signal of high RHR or HPCF pump discharge pressure. A time delay of 29 seconds will be used to confirm that the low water level 1 signal is present and is consistent with the ECCS pump startup time. The instrumentation and controls for the ADS are discussed in Chapter 7 of this report.

The LPFL system will replace reactor vessel water inventory following large-pipe breaks. The system is part of the RHR system, which consists of three independent loops (A, B, and C). Each loop has a motor-driven pump with a capacity of 15,900 L/min (4,200 gpm) that will take suction from the suppression pool and supply water to the reactor vessel. All three RHR loops include heat exchangers that will be cooled by the RHR reactor building cooling water system and transfer the decay heat from the reactor core to the ultimate heat sink. The three LPFL (RHR) pumps will be powered from ac power buses that have standby backup sources of power. RHR pumps A, B, and C will receive emergency power from the three separate diesel generators. The RHR system valve logic will require LPFL system alignment in the event of a LOCA. The LOCA event takes precedence over other RHR system functional modes. The system will initiate automatically by either low water level 1 or high drywell pressure signals. The reactor must be depressurized below the reactor low-pressure permissive signal before LPFL injection to the reactor occurs. Each of the two high-pressure core flooding loops and two of the three low-pressure flooding loops will discharge water into the core through a separate overhead flooder sparger. Low-pressure flooding loop A will discharge into the RPV through the feedwater system. Internal vessel piping connects each sparger to the vessel nozzle. The ABWR flooder design and relative location will result in reduced personnel radiation exposure as compared to the current BWR core spray design because the peripheral location of the flooder minimizes the need for work over the fuel during inservice inspection. Spray distribution in the core is not critical because there will be no core uncovery during a LOCA.

6.3.2 Evaluation of Single Failures

The staff reviewed the SSAR system description and piping and instrument drawings to ensure that abundant core cooling will be provided during the injection phase with and without offsite power and assuming a limiting single failure as required by GDC 35. A low reactor vessel water level and/or containment high pressure signal is required to start pumps and open discharge valves.

GE provided in SSAR Section 6.3.3 an analysis to demonstrate that the most limiting break size, break location, and single failure had been considered for the ABWR. The most limiting combinations are given in Table 6.1 of this report. The staff finds that the SSAR information supports the finding that the ECCS systems meet the single-failure criterion.

6.3.3 Qualification of Emergency Core Cooling System

The ECCS is designed to meet seismic Category I requirements in compliance with RG 1.29, "Seismic Design Classification," (Rev. 3), as discussed in Section 3.2 of this report. The ECCS is housed in structures designed to withstand seismic events, tornados, floods, and other phenomena, in accordance with the requirements of GDC 2, as discussed in Section 3.2.1 of this report. ECCS equipment is designed in compliance with RG 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," (Rev. 4), as discussed in Section 3.2 of this report.

Protecting the ECCS against pipe whip and against discharging fluids, in compliance with the requirements of GDC 4, is discussed in Section 3.6 of this report. Evaluation of the instrumentation and controls for the ECCS, including compliance with RG 1.47, "Bypass and Inoperable Status Indication for Nuclear Power Plant Safety Systems," in accordance with the applicable requirements of GDC 27, is discussed in Section 7.3 of this report. Compliance with the inservice inspection requirements of GDC 36 is discussed in Section 6.6 of this report.

Environmental qualification of the ECCS equipment for operation under normal and accident conditions, as required by GDC 4, is discussed in Section 3.11 of this report.

Table 6.1 Single failure evaluation

Assumed Failure	Systems Remaining
Emergency diesel generator A	All ADS, RCIC, 2 HPCF, 2 RHR/LPFL
Emergency diesel generator B or C	All ADS, RCIC, 1 HPCF, 2 RHR/LPFL
RCIC injection valve	All ADS, 2 HPCF, 3 RHR/LPFL
One ADS valve	All ADS minus one, RCIC, 2 HPCF, 3 RHR/LPFL

Note: ADS = automatic depressurization system
HPCF = high-pressure core flooder
LPFL = low-pressure flooder
RCIC = reactor core isolation cooling
RHR = residual heat removal

In accordance with the pertinent requirements of GDC 35, the available net positive suction head (NPSH) for the pumps in the ECCS, as reflected in calculations submitted by GE, is adequate to prevent cavitation and ensure pump operability in accordance with RG 1.1, "Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps (Safety Guide 1)," (Rev. 0). The pump head and NPSH requirements for ECCS pumps are included in the ITAAC for verification by the COL applicant.

Each of the low-pressure lines that will discharge into the reactor coolant system has a testable check valve inside the primary containment backed up by a normally closed motor-operated gate valve outside the containment. Relief valves in the low-pressure lines will protect against leakage from the reactor coolant system. An interlock on the motor-operated valves will prevent them from opening until the reactor coolant pressure is below the low-pressure ECCS design pressure.

Containment isolation in accordance with the requirements of GDC 55 is discussed in Section 6.2 of this report. The periodic testing and leak-rate criteria for those valves that isolate the reactor system from the ECCS are discussed in Section 3.9.6 of this report. Detection of leaks from those portions of the ECCS within the primary coolant pressure boundary is discussed in Section 5.2.5 of this report.

To protect the pumps from overheating, all the ECCS pumps have minimum flow bypass lines to permit a limited amount of flow if an isolation valve between the reactor coolant system and the ECCS is closed for any reason. When flow in the injection lines is sufficient for pump cooling, valves in the minimum flow bypass lines will close automatically, diverting all flow to the pressure vessel. Each LPFL pump suction line from the suppression pool has an open motor-operated valve outside the containment. The suction line of the HPCF from the suppression pool contains a closed motor-operated valve so the HPCF initially will draw water from the CST. When the CST water is exhausted, the suppression pool suction valve will open automatically.

Isolation of the suppression pool from the reactor building in accordance with GDC 56 is discussed in Section 6.2 of this report.

As a backup to the HPCF system, the ADS will be used to depressurize the reactor and allow the LPFL to function in the event of a small break. Nitrogen will be supplied to the valves of the ADS from seismically qualified accumulators.

One of the design requirements of the ECCS is that cooling water flow be provided rapidly following the initiation signal. By always keeping the ECCS pump discharge lines full, the lag time between the signal for pump start and the initiation of flow into the RPV can be minimized. In addition, full discharge lines reduce potentially damaging waterhammer occurrences on system startup. The RHR system has three jockey pumps, one in each loop (discharge line fill pump). Maintaining the filled status of the system is ensured by continuous indication of pump operation and pump discharge pressure. The makeup water system connections to the RCIC and HPCF system will maintain the pump discharge line in a filled condition up to the injection valve.

The capability of the ECCS pumps to operate for an extended period of time during the long-term recirculation phase following a LOCA is discussed in Section 3.9.6.1 of this report.

Safety/relief valve (SRV) operability will be demonstrated during the power ascension phase of the plant startup test program by manually actuating each SRV (including the ADS valves) one at a time to show that no blockage exists in the valve discharge line. After commercial turnover, all of the SRVs will be tested in accordance with Section XI, Article IWV, of the ASME Code. Linear variable differential transformers and thermocouples are to be installed in the discharge line of each SRV to monitor valve position and SRV leakage in accordance with TMI-2 Action Plan Item II.D.3 (NUREG-0737) and are discussed in Section 20.5.23 of this report.

Environmental qualification of the SRVs is discussed in Section 3.11 of this report.

The ADS time delay is set at 29 seconds rather than 120 seconds (as in present BWRs) to satisfy the design goal of no core uncovery during a LOCA. The ADS most likely will not be initiated because of the availability of three high-pressure core injection systems. The ABWR emergency procedures guidelines (EPGs) (which are provided in SSAR Chapter 18) states that the operator is allowed to prevent ADS actuation in two instances. The first is during an anticipated transient without scram (ATWS) event. Because the ABWR design incorporates an automatic ADS inhibit signal following an ATWS, operator action is to verify the automatic ADS inhibit signal as addressed in SSAR Chapter 18 (EPGs). The second case is when the operator believes that the reactor water level may go below the reactor low water level 1 setpoint but will remain above the top of the active fuel (TAF) without ADS actuation. Because no ADS timer is provided for LOCAs in the containment, where high drywell pressure occurs, and because the ADS time delay is short, operator intervention is unlikely. However, during LOCAs outside the containment or LOCAs that do not result in a high drywell pressure signal, an 8-minute timer will be initiated and the EPGs allow the operator to assess whether the level is maintained above TAF. If the high-pressure ECCS cannot control the reactor water level, it is prudent to allow the ADS to actuate and quickly depressurize the vessel to gain the additional reflooding capacity of the low-pressure ECCS. The EPGs provide sufficient guidance to ensure that operating procedures will reflect the importance of allowing ADS actuation when high-pressure makeup systems are not available.

For transient and accident events that do not directly produce a high drywell pressure signal (e.g., stuck-open relief valve or steamline break outside the containment) and that are further complicated by the loss of all high-pressure ECCS systems, manual activation of the ADS was originally required to provide adequate core cooling. However, TMI Action Plan Item II.K.3.18 (NUREG-0737), as incorporated in 10 CFR 50.34f(1)(vii), requires ADS logic modification to eliminate operator action. GE's proposed design modification is consistent with Option 4 of the BWR Owner's Group response to TMI Action Plan Item II.K.3.18. This option requires the addition of a timer that bypasses the high drywell pressure permissive if the reactor water level is low for a sustained period and the addition of a manual inhibit switch. An 8-minute high drywell pressure bypass timer has been added to the ABWR ADS initiation logic. (An analysis was performed to evaluate the adequacy of the 8-minute bypass timer.) This timer will initiate on a low water level 1 signal. When the timer runs out, it will bypass the need for a high drywell pressure signal to initiate the standard ADS initiation logic. The bypass timer will be tested periodically. This test is required by the TS. This was DFSER TS Item 6.3.3-1. (This item has been added to the ABWR TS and has been found to be acceptable.) This timer can be inhibited as indicated in the ABWR EPGs if the operator determines that water level can be maintained above TAF. The proposed GE modification to ADS logic is consistent with the staff-approved resolution of Item II.K.3.18 and is acceptable because it ensures the initiation of the ADS and low-pressure injection systems when high drywell pressure does not occur. In the DSER (SECY-91-153) the staff identified a concern about the ADS in Open Item 28. The above discussion on bypass timers shows that DSER Open Item 28 has been resolved. Assurance of the proposed modification to the ADS logic was DFSER Confirmatory Item 6.3.3-1. GE submitted the required changes in the SSAR in Section 6.3.3.4, which is acceptable to the staff. GE also included this action item in the SSAR. This Confirmatory Item 6.3.3-1 is resolved.

6.3.4 Testing

RG 1.68, "Initial Test Programs for Water-Cooled Nuclear Power Plants," (Rev. 2) and GDC 37 require that the ECCS system be designed to permit appropriate periodic pressure and functional testing to assure the integrity of components, and the operability and performance of the components and ECCS system. GE states that ECCS integrity, operability, and performance are demonstrated by preoperational and periodic testing, which is acceptable to the staff.

6.3.4.1 Preoperational Tests

Preoperational tests will ensure proper functioning of controls, instrumentation, pumps, piping, and valves. Pressure differentials and flow rates are measured for later use in determining acceptable performance in periodic tests. GE commits to conform to the guidelines of RG 1.68 for preoperational and initial startup testing of the ECCS, as noted in Section 14.2 of this report.

6.3.4.2 Periodic Component Tests

The staff stated in the DFSER that the TS should state that the ECCS subsystems (except for the ADS) will be tested periodically to show that specified flow rates are attained. This was DFSER TS Item 6.3.4.2-1. GE has included this statement in the TS. The staff also stated in the DFSER that the COL applicant should perform a test every refueling outage in which all subsystems are actuated through the emergency operating sequence. This was DFSER COL Action Item 6.3.4.2-1. GE provided this procedural COL action item in Amendment 31 of the SSAR.

6.3.5 Performance Evaluation

GE's ABWR-specific LOCA analysis demonstrates that the reference fuel design meets the requirements of 10 CFR 50.46 and is based on the initial core design.

The staff expected the exposure-dependent maximum average planar linear heat generation rate (MAPLHGR) to be provided in the TS. This was DFSER TS Item 6.3.5-1. GE submitted the MAPLHGR in TS 3.2.1; hence this item is resolved. In Section 4.2 of this report, the specific fuel and core design used for the analyses of SSAR Sections 6 and 15 is described. Section 4.2 of this report also describes fuel design criteria that specify requirements for any alternative fuel that may be used by the COL applicant or licensee in reload cycles.

The staff used the information in the SSAR, the results of the LOCA analyses, along with the results of its review of each ECCS system to verify that the proposed ECCS meets the performance criteria in 10 CFR 50.46. Compliance with the first three criteria below is demonstrated analytically. Coolable geometry is maintained if the first two criteria are met. Long-term cooling capability is verified by the composite review of the ECCS and the various support systems. The five acceptance criteria for the ECCS, as specified in 10 CFR 50.46, are the following:

(1) The calculated maximum peak cladding temperature (PCT) shall not exceed 1204 °C (2200 °F).

(2) The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation.

(3) The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding plenum volume, were to react.

(4) Calculated changes in core geometry shall be such that the core remains amenable to cooling.

(5) The calculated core temperature shall be maintained at an acceptably low value, and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core after any calculated successful initial operation of the ECCS.

GE has demonstrated compliance with the first three of these criteria as shown in Table 6.2 of this report.

Table 6.2 Demonstration of compliance with ECCS criteria

	Maximum	
Criterion	From Break Analyses	Allowable
Peak cladding temperature, °C (°F)	621 (1149)	1204 (2200)
Maximum cladding oxidation, %	0.03	17
Maximum total hydrogen generation, %	0.03	1

There is no core uncovery and hence the core geometry is maintained. Moreover, a coolable geometry was demonstrated by compliance with the criteria for PCT and maximum cladding oxidation as discussed in NEDO-20566, "Analytical Model for Loss-of-Coolant Analysis in Accordance with 10 CFR Part 50, Appendix K," proprietary, November 1975.

Because there are no jet pumps in the ABWR, the core flow pattern is similar to that in BWR/2 plants that do not use jet pumps (e.g., Oyster Creek, Nine Mile Point Unit 1). The GE thermal-hydraulic code SAFER, which was approved for BWR/2 plants without jet pumps (NEDE-30996P-A, "SAFER Model for Evaluation of Loss-of-Coolant Accidents for Jet Pump and Non-Jet Pump Plants," (Volumes I and II) October 1987), was used to perform LOCA analysis for the ABWR. The SAFER code was used to calculate the long-term thermal-hydraulic behavior of the coolant in the vessel during a LOCA. Some important system parameters calculated by SAFER are vessel pressure, vessel water level, and ECCS flow rates. The SAFER code also calculates PCT and local maximum oxidation. The staff's SER (dated February 19, 1987, "Review of NEDE-30996(P), 'SAFER Models for Evaluation of Loss-of-Coolant Accident for Jet Pump and Non-Jet Pump Plants,' Volumes I and II") on NEDE 30996A-P documents agreement between SAFER predictions and test results from the ABWR full integral simulation test (FIST) facility. GE made some modifications to the FIST facility for the ABWR and conducted additional tests. Jet pump elimination and reactor internal pump (RIP) flow were simulated. GE also modified ECCS capacities, break sizes, and locations. The SAFER code was compared to TRACG-P (proprietary) for the ABWR plant. (See staff SER of February 19, 1987.)

ABWR-specific data based on tests for the RIP coastdown time following a pump trip and flow area and pressure losses through the RIPs were used as input to the SAFER model.

GE computer codes LAMB, SCAT, and GESTR were used for the ABWR LOCA analysis.

The LAMB code is used to analyze the short-term thermodynamic and thermal-hydraulic behavior of the coolant in the vessel during a postulated LOCA. In particular, LAMB predicts the core flow, core inlet enthalpy, and core pressure during the initial phase of the LOCA event (i.e., the first 5 seconds). GE used the LAMB computer model (documented in NEDE 20566P-A, "Analytical Model for Loss-of-Coolant Accident Analysis in Accordance with 10 CFR Part 50, Appendix K," January 1976, and approved by the staff) to perform analyses presented in the ABWR SSAR. The ABWR input to the LAMB computer model is the same input as that used for operating plants except for recirculation system modeling. Because the ABWR has RIPs instead of jet pumps, the recirculation system is modeled assuming no induced flow from a jet pump. Thus, the drive flow from the recirculation pumps is set equal to the total recirculation flow.

The SCAT code is used to evaluate the short-term thermal-hydraulic response of the coolant in the core during a postulated LOCA. Using the LAMB results as input, SCAT analyzes the convective heat transfer process in the thermally limiting fuel bundle during the initial phase of the LOCA event. In particular, SCAT predicts the departure from nucleate boiling at any one of the 24 axial nodes in the fuel bundle. GE used the SCAT computer model (documented in NEDE 20566P and approved by the staff) to perform analyses presented in the ABWR SSAR. Specifically, GE used SCAT to determine the transient thermal-hydraulic conditions within a bundle and predict the time when the loss of nucleate boiling occurs. The SCAT program only models the hot bundle and sets the boundary conditions based on LAMB results. The ABWR core design is based on a GE standard fuel that is similar to BP 8 x 8R with two water rods. This type of fuel has already been analyzed using SCAT for operating BWRs. Therefore, the differences in the ABWR from previous BWRs do not represent any special application of the SCAT model.

The GESTR code is used to provide best-estimate predictions of the thermal performance of GE nuclear fuel rods experiencing variable power histories. For LOCA analysis, the GESTR code is used to initialize the fuel stored energy and fuel rod fission gas inventory at the onset of a postulated LOCA. GE used the GESTR computer model (documented in NEDE 30996P-A and approved by the staff) to perform LOCA analyses presented in the ABWR SSAR. GE used the results from GESTR to establish the initial conditions (i.e., stored energy and rod internal pressure) at the start of the LOCA within each fuel rod. As stated above, GE based its ABWR core design on a GE standard fuel, which also has been analyzed using GESTR for operating BWRs. The differences in the ABWR from previous BWRs do not affect the fuel thermal-mechanical performance predicted by the GESTR model. Therefore, the use of GESTR for the ABWR does not represent any special application of the GESTR model.

No model changes were made to any of the GE codes used in the LOCA analysis; only input data were changed for ABWR-specific design features.

The GE analyses included break sizes ranging from a bottom head drain line break 0.0020 m^2 (0.0218 ft^2) to the main steamline break outside the containment 0.39 m^2 (4.24 ft^2). (Since the bottom head drain line will tie into the RHR/cleanup system lines, the total break flow for the maximum bottom head drain line break includes flow from the vessel through the bottom head drain line penetration as well as through the RHR/cleanup system lines.) GE analyzed different break sizes in conjunction with ECCS failure combinations. The cases were evaluated to establish the trend of PCT curves (Appendix K to 10 CFR Part 50 and bounding values) versus break size. Eight break sizes are summarized in SSAR Table 6.3-4. The most limiting break is the main steamline break outside the containment, which results in a PCT of 621 °C (1149 °F). This is well below the 1204 °C (2200 °F) acceptance criterion. Because there will be no large pipe below the top of the core and no core uncovery for any size LOCA, the calculated PCT of 621 °C (1149 °F) is low compared to the PCT for current BWRs.

The staff confirmed that the LOCA analysis methodology used by GE for the ABWR was consistent with and bounded by the staff's generically approved LOCA analysis methodology. The staff concluded that the PCT, peak local oxidation, and core-wide metal-water reaction values were well below staff acceptance criteria. Thus, the analyses and results are in accordance with NRC requirements and GE has demonstrated conformance with the ECCS acceptance criteria of 10 CFR 50.46 and Appendix K to 10 CFR Part 50.

Long-term cooling is ensured by the use of redundant RHR systems that have adequate water sources available to remove the decay heat generated in the reactor core and transfer the heat to the ultimate heat sink. GE identified no single failure that would prevent the ECCS from meeting this criterion. The systems are designed to prevent any core uncovery.

The LPFL flow will be diverted manually to wetwell spray cooling for containment pressure control. The ABWR EPGs require cautions in the emergency operating procedures to deter the operator from premature flow diversion. These guidelines, which caution the operator against diversion unless adequate core cooling is ensured, have been accepted by the staff (NUREG-0737, TMI-2 Action Plan Item I.C.1, discussed in Chapter 20 of this report). LPFL diversion is identified in the procedure as secondary to core cooling requirements, except in those instances outside the design envelope involving multiple failures for which maintenance of containment integrity is required to minimize risk to the environment.

The ECCS meets GDC 17 in that its capacity and capability are sufficient to ensure that acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of anticipated operational occurrences; and the core will be cooled and vital functions will be maintained in the event of postulated accidents as given in SRP Section 6.3.

6.3.6 Certified Design Material

GE submitted the design description and the inspections, tests, analyses, and acceptance criteria (ITAAC) for the HPCF system. This was DFSER Open Item 6.3.6-1. GE subsequently provided a revised design description and ITAAC. The adequacy and the 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.

6.3.7 Conclusions

As discussed above, the staff finds the design of the ECCS acceptable. The ECCS conforms with the review guidelines and acceptance criteria of SRP Section 6.3 and its pertinent RGs. Therefore, the ECCS meets the performance criteria of 10 CFR 50.46 and the pertinent requirements of GDC 2, 4, 17, 35, 36 and 37 as set forth in the SRP, as summarized below:

(1) 10 CFR 50.46 and Appendix K performance requirements, as described in Section 6.3.5 of this report.

(2) GDC 2, as it relates to the seismic design of SSC whose failure could cause an unacceptable reduction in the capability of the ECCS to perform its safety functions, as discussed in Section 3.2.1 of this report.

(3) GDC 4, as related to dynamic effects associated with flow instabilities and loads, as discussed in Section 3.6 of this report.

(4) GDC 5 is not applicable since the ABWR is a single unit plant.

(5) GDC 17, as it relates to the ECCS design for sufficient capacity and capability to assure that specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded and that the core is cooled during anticipated operation occurrences and accident conditions, as discussed in Section 6.3.5 of this report.

(6) GDC 27, as it relates to the reactivity control systems to have a combined capability, in conjunction with poison addition by the emergency core cooling system, or reliably controlling reactivity changes under postulated accident conditions, with appropriate margin for stuck rods is discussed in Sections 4.2, 4.6 and 9.3.5 of this report.

(7) GDC 35, 36, and 37, as they relate to the ECCS design to provide an abundance of core cooling to transfer heat from the core at a rate so that fuel and clad damage will not interfere with continued effective core cooling, to permit appropriate periodic inspection of important components, and to permit appropriate periodic pressure and functional testing, as discussed in Sections 6.3.2, 6.3.4, and 6.6 of this report.

6.4 Control Room Habitability Systems

The staff reviewed the control room habitability systems in accordance with SRP Section 6.4. Conformance with the acceptance criteria formed the basis for the staff's evaluation of the control room habitability systems with respect to the applicable regulations of 10 CFR Part 50. Specifically, the SRP acceptance criteria require the design to meet GDC 4 as it relates to accommodating the effects of and being compatible with the environmental conditions associated with postulated accidents, including the effects of the release of toxic gases; GDC 19 as it relates to maintaining the control room in a safe, habitable condition under accident conditions by providing adequate protection against radiation and toxic gases; additional TMI requirement 10 CFR 50.34(f)(2)(xxviii) as it relates to the evaluation of potential pathways for radioactivity and radiation that may lead to control room habitability problems; and TMI Action Plan Item III.D.3.4 (NUREG-0737) requirements as they relate to providing protection against the effects of the release of toxic substances, either on or off the site. Since the ABWR design is applicable only for a single unit, GDC 5 is not applicable.

The control room habitability systems will provide (1) missile protection, (2) radiation shielding, (3) radiation monitoring, (4) air filtration and ventilation, (5) lighting, (6) personnel and administrative support, and (7) fire protection. The control building heating, ventilating, and air conditioning (HVAC) system and components are located in a seismic Category I structure that is protected from tornado, missile, pressure and flood damage. The HVAC ducting is ESF grade. HVAC hangers are designed to seismic Category I standards. The HVAC system will maintain the control room atmosphere temperature at a habitable level to permit prolonged personnel occupancy throughout a postulated design-basis accident (DBA). The system design provides for control room pressurization with respect to the surrounding spaces and filtered intake during accident situations and for purging of smoke and toxic gases. The system is capable of automatic transfer from its normal operating mode to its emergency or isolation modes on detection of adverse conditions (e.g., high radiation, smoke). The system has sufficient redundancy to ensure operation under emergency conditions, assuming the single failure of any one active component. Backup power sources are provided for the essential components of the HVAC system. Section 9.4.1 of this report provides more information on the control room HVAC system. The habitability systems are designed to detect and limit the introduction of radioactive material and smoke into the control room. The ABWR design relies on noncombustible construction and heat- and flame-resistant materials throughout the plant to minimize the likelihood of fire and consequent fouling of the atmosphere with smoke or noxious vapor. Further, the number of individual respirators (subject to periodic operational testing) is sufficient to protect against the intrusion of toxic gases into the control room. Non-seismic pipe, ductwork for kitchen and sanitary facilities, and other nonessential components in the control building are designed to ensure that their failure during a safe shutdown earthquake will not adversely affect essential components. Potential sources of danger such as pressure vessels and carbon dioxide firefighting containers will be located outside the control room and the compartments containing control building life support systems. There are no high-energy lines near the control room; therefore, the habitability systems are protected against the dynamic effects that may result from possible failures of such lines. Section 3.6.1 of this report addresses protection of safety-related equipment from the effects of pipe breaks. The staff concludes that the control room habitability systems satisfy GDC 4.

By letter dated July 7, 1988, the staff requested additional information on a number of issues (e.g., makeup air inlet for control room emergency zone, locations of control room ventilation inlets relative to major potential plant release points, radiation protection instrumentation, and minimum positive pressure during the pressurization mode) regarding the design features provided for compliance with GDC 19. This was identified as Open Item 29 in the DSER (SECY-91-153). GE has provided the requested information. The staff has reviewed and finds it acceptable as discussed below. Therefore, DSER Open Item 29 is resolved.

The ABWR main control room (MCR) is physically integrated with the reactor building and turbine building and is located between these two structures. During and following a LOCA, which is the controlling DBA for the radiological consequences to the control room operators, radiation exposure to the operators will consist of contributions from airborne fission products entrained in the control room ventilation system and direct gamma radiation from the surrounding buildings and process equipment.

GDC 19 requires that the control room be designed to permit personnel to occupy it under accident conditions. GE has proposed that this requirement be met by incorporating shielding and emergency ventilation systems into the control room design, and by two roof-mounted and automatically controlled room air intakes. GE has stated that the structure housing the control room is designed and will be constructed to meet seismic Category I criteria, as is the emergency ventilation system, which is also designed to meet the single-failure criterion.

GE has provided for the radiation protection of control room occupants by the use of shielding walls and by the installation of redundant safety grade emergency ventilation systems. The inhabited portions of the ABWR control room are located underground (4.3 m (14 ft) below grade). A distance of more than 11.3 m (37 ft) (including 2 m (6.5 ft) of concrete shielding and 0.46 m (1.5 ft) of floor) separates the control room and steamlines. GE stated that the expected radiation exposure rate in the control room resulting from direct gamma radiation during normal operation and after an accident is less than 0.006 mSv/hr (0.6 mR/hr). The staff accepted GE's estimate of direct gamma radiation of less than 0.006 mSv/hr (0.6 mR/hr) after an accident because of the decay of short-lived isotopes and the immediate closure of the main steam isolation valves that are upstream of the pipes above the MCR.

If a significant concentration of airborne radioactive materials is detected at the normal control room ventilation system air intake, the air intake will isolate automatically. Automatic control room pressurization will occur immediately, and filtered air will be taken in by either of two separate emergency ventilation systems. Each emergency ventilation system consists of, at a minimum, a 50 mm (2 in.)-thick charcoal adsorber for removal of iodines. The intakes for these systems are separated from each other and from the plant stack. It is not likely that both air intakes would simultaneously admit equivalent levels of radioactive contaminated air.

The viability of the dual-inlet concept depends on whether or not the placement of the inlets ensures that airborne radionuclide concentrations in one inlet will always be relatively low. The capability to ensure that this condition exists at an inlet depends, in part, on building wake effects, site-specific terrain, and wind stagnation or reversal. For the ABWR design, the inlets are located at the extreme edges of the control building. However, the staff finds that it is possible, under certain low-probability conditions, for both inlets to be drawing air from the same source of radioactive materials, and that the location of the air inlets is less than ideal, since they are not on opposite sides (180°) of potential radiation release points.

In Amendment 8, GE claimed a factor of 4 reduction in the estimate of the atmospheric dispersion parameter for the control room to account for dilution effects associated with the ABWR air inlet configurations and the ABWR-specific building arrangement. SRP Section 6.4 allows a factor or 10 for reduction of the parameters for a dual air intake design with automatic selection control features and with the inlets placed on plant structures on opposite sides. Therefore, the staff stated in the DFSER that GE's proposed reduction factor of 4 (instead of 10) was reasonable and acceptable because the lower reduction factor consecutively accounts for air inlets not being on opposite (180°) sides of potential radiation.

In Amendment 24, GE changed the control room HVAC system design by deleting automatic selection features of the most favorable (less radioactive) air intake from either of two separate emergency air intakes to the control room. In addition, GE revised (1) the filtered emergency air intake flow rate into the control room, (2) the control room air recirculation flow rate, and (3) the control room limiting atmospheric dispersion values provided in Table 15.6-14 of the SSAR. Therefore, the staff disallowed a factor of 4 reduction given in the DFSER for estimating the atmospheric dispersion parameters for the control room, since GE had deleted the automatic selection features of the most favorable air intake from the ABWR design. Even though SRP Section 6.4 allows a factor of 2 for reduction of the parameters for a dual air intake design without manual or automatic selection features, the staff has not provided, and GE did not request in Amendment 24, any reduction factors because the emergency control room air intakes are not placed on opposite sides of potential radiation release points.

The staff recalculated the control room operator doses using the design-basis LOCA described in Section 15.3.1 of this report and revised control room design parameters provided in Amendment 24. The revised staff assumptions and dose estimates are listed in Table 15.10 of this report. The staff concludes that the calculated doses still meet the radiological consequences values of GDC 19 and TMI Action Plan Item III.D.3.4, "Control Room Habitability." The staff further concludes that the ABWR control room design still 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 TMI requirement 10 CFR 50.34(f)(2) (xxviii).

GE identified an interface requirement described below for the applicants referencing the ABWR design to protect operators against the effects of the release of toxic substances. However, as a result of further review of all interface items, GE committed to revise the SSAR to reclassify this item as a COL action item.

By letter dated April 16, 1993, GE stated that the ABWR standard plant site design parameters (SSAR Table 2.0-1) did not include toxic gases in the site vicinity. A COL action item has been included in Section 6.4.7 of the SSAR requiring utility applicants referencing the ABWR design to demonstrate that control room operators are adequately protected against the effects of the release of toxic substances, either on or off the site, and that the plant can be safely operated or shut down under conditions created by a DBA. The need for site-specific toxic gas protection will be reviewed to ensure that the control room operators are protected against releases of hazardous material in accordance with TMI Action Plan Item III.3.D.4. The amounts and locations of any possible sources of toxic substances in each plant vicinity will be identified following the methods outlined in RGs 1.78 and 1.95. Specific detectors to permit automatic control room isolation will be provided, where necessary.

In the DFSER, the staff stated that the following were to be included in the ITAAC. The COL applicant will need to verify that the following are consistent with the licensing basis documentation: the asbuilt design; the operating, maintenance, and emergency procedures and training; the performance characteristics of the control room habitability system; and the TS and surveillance procedures. This was identified as Open Item 6.4-1 in the DFSER. GE provided a revised design description 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, DFSER Open Item 6.4-1 is resolved.

The staff reviewed the ITAAC for the control room habitability area (CRHA) HVAC system (Table 2.15.5a, Design Commitments 1 through 10) and the control building (Table 2.15.12, Design Commitments 1 through 13) as they relate to control room habitability. In the DFSER, the staff found them to be acceptable with some exceptions. These were identified as DFSER Open Item 6.4.2. GE provided a revised design description and ITAAC. The adequacy and acceptability of the design description and the ITAAC are evaluated in Chapter 14.3 of this report. On the basis of this evaluation, DFSER Open Item 6.4-2 is resolved.

The staff concludes that the control room habitability systems meet the acceptance criteria of SRP Section 6.4 and are, therefore, acceptable.

6.5 Fission Product Removal and Control Systems

6.5.1 Engineered Safety Features Atmosphere Clean up Systems

The staff reviewed the engineered safety feature (ESF) atmosphere cleanup systems in accordance with SRP Section 6.5.1. Conformance with the acceptance criteria formed the basis for the staff's evaluation of these systems with respect to the applicable regulations of 10 CFR Part 50.

The ABWR design has two ESF filter systems: the emergency air filtration system of the CRHA HVAC system and the standby gas treatment system (SGTS).

SRP Section 6.5.1 acceptance criteria applicable for the emergency air filtration system's high-radiation mode of the CRHA HVAC system are GDC 19 as it relates to the system being designed to ensure habitability of the control room under accident and LOCA conditions and GDC 61 as it relates to the design of the system for radioactivity control under normal and postulated accident conditions. As the staff concluded in Section 6.4 of this report, the emergency air filtration system, which is part of the CRHA HVAC system, complies with GDC 19 radiation exposure limits for the control room operators. The SRP states that the specified acceptance criteria are met by conforming with the guidelines of RG 1.52, "Design, Testing, and Maintenance Criteria for Post Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," Revision 2, and SRP Table 6.5.1-1 regarding the system instrumentation. Therefore, the staff has used system compliance with RG 1.52 and SRP Table 6.5.1-1 as the basis for concluding that the ESF filter system meets the applicable GDC and consequently the applicable SRP acceptance criteria. The above compliance evaluation is discussed in Section 9.4.1 of this report for the emergency air filtration system of the CRHA HVAC system and in this section for the SGTS.

SRP Section 6.5.1 acceptance criteria applicable for the SGTS are GDC 41 as it relates to the design of the system as it will be used for containment atmosphere cleanup following postulated accidents and for the control of releases to the environment, GDC 42 as it relates to the inspection and testing of the SGTS, GDC 61 (see the above paragraph) and GDC 64 as they relate to monitoring radioactive releases via the SGTS under normal, anticipated operational occurrences and postulated accident conditions. As the staff concludes in Section 11.5 of this report, the system complies with GDC 64.

The staff's evaluation of the two ESF filter systems is given below.

Emergency Air Filtration System: The function of this system is to supply filtered outside air to the main control area envelope (MCAE) and to pressurize the control room after a design-basis accident (DBA). This system is evaluated in Section 9.4.1 of this report. In Section 9.4.1 of the SSAR and the DFSER, the emergency air filtration system of the CRHA HVAC system was erroneously identified as the control building outdoor air cleanup system. GE has corrected this error in SSAR Amendment 20, and SSAR Amendment 32 identified separate and independent discharge and return paths to and from the MCAE for the emergency filtration system.

Standby Gas Treatment System: The primary function of the SGTS is to filter and thereby reduce offsite airborne releases of radioiodine and particulates following a DBA (e.g., LOCA, fuel-handling accident, fuel cask drop accident). This will be accomplished by automatic isolation of the secondary containment from its normal HVAC air paths and automatic actuation of the SGTS on receipt of a LOCA signal, detection of high radiation levels in the secondary containment normal HVAC exhaust or in the refueling floor exhaust or loss of secondary containment HVAC supply/exhaust fans signal. The system can also be manually initiated from the control room (e.g., for de-inerting during power operation) to reduce airborne radioactive iodine and particulate releases. The SGTS will maintain the secondary containment at a negative pressure of 6.4 mm (1/4 in.) of water gauge or greater relative to the surrounding spaces within 20 minutes from the time the secondary containment is isolated and process the effluent gases through a filter train to remove airborne iodines and particulates. The system consists of two identical, parallel, physically separated, 100-percent-capacity subsystems, 6,800 m^3/hr (4,000 cfm) each, with associated piping and ducts, valves, dampers, and controls. Each subsystem consists of a moisture separator, electric process heater, prefilter, pre-high-efficiency particulate air (HEPA) filter, charcoal adsorber (15-cm (6 in.) depth), post-HEPA filter, process fan (6,800 m^3/hr (4,000 cfm) at 1 atmosphere and 21 °C (70 °F)) and cooling fan (700 m^3/hr (412 cfm) at 1 atmosphere and 21 °C (70 °F)). IE Bulletin 80-03 has been met since GE has revised SSAR Section 6.5.1.3.3 to state that the charcoal tray and screen will be all welded construction in accordance with the bulletin. Therefore, the SGTS filter trains will preclude the possible loss of charcoal from adsorber cells. The system is designed to seismic Category I requirements and will be housed in a seismic Category I structure.

In the DSER (SECY-91-153), the staff stated that Appendix 6A to Chapter 6 of the SSAR demonstrated the system's compliance with each of the regulatory positions in RG 1.52, (Rev. 2), except one relating to the requirement for redundancy in filter trains. Originally, the SGTS design included only a single filter train. In the DSER (SECY-91-153), the staff identified the use of a single filter train in the SGTS as Open Item 30. Before the DFSER was issued, GE submitted in a letter dated February 13, 1992, piping and instrumentation diagrams (P&ID's) for a number of systems including the SGTS. The SGTS P&ID showed two filter trains. Therefore, in the DFSER, DSER Open Item 30 was resolved. However, as a followup to GE's proposed design change, the staff required GE to confirm the design change by (1) submitting layout drawings showing the locations of the two filter trains and explaining the adequacy of the spatial separation between the trains, and (2) revising SSAR Section 6.5.1, Table 6.5-1, and Appendices 6A and 6B. The staff identified the above requirements as Confirmatory Item 6.5.1-1 in the DFSER. GE provided the required information and SSAR section, table, and appendices in an amendment to the SSAR. The staff has reviewed them and finds them acceptable. Therefore, DFSER Confirmatory Item 6.5.1-1 is resolved.

Appendix 6B to Chapter 6 of the SSAR addresses the compliance of the instrumentation for the SGTS with the requirements in Table 6.5.1-1 of SRP Section 6.5.1. Before the DFSER was issued, GE identified in Appendix 6B several deviations from the instrumentation requirements listed in SRP Table 6.5.1-1. GE attributed these deviations as being partly due to the single filter train proposed for the SGTS. In the DFSER, the staff stated that in addition to its concern regarding the deviations that stem from the use of a single filter train, it was concerned about deviations that cannot be attributed to a single filter train design. Therefore, as stated above, the staff required GE to revise Appendix 6B to reflect the redundant filter trains and additionally justify any remaining deviations from SRP Table 6.5.1-1. The staff identified its concern regarding the deviations as Open Item 6.5.1-1 in the DFSER. GE has amended Appendix 6B and has provided acceptable justifications for the deviations. The staff has reviewed GE's justifications for compliance of the SGTS with SRP Table 6.5.1-1 (minimum instrumentation, readout, recording and alarm provisions for ESF atmosphere cleanup systems) and finds that the proposed instrumentation and controls in the main control room

(MCR), as well as selected local panels, meet the above guidance, and are acceptable. Although it would prefer that GE provide all of the local instrumentations and controls described in the above guidance for operational efficiency and convenience, the staff finds that the SGTS instrumentation and controls meet the above guidance on the basis of the following:

(1) The instrumentation and control (i.e., readout, recordings, and alarms) provisions for the related parameters (i.e., pressure drop, temperature, humidity, flow, etc.) are available and would be made accessible in the MCR.

(2) Plant-specific procedures will be developed to respond to these instrumentation and controls.

(3) The filtration train fully complies with the regulatory position of RG 1.52.

(4) The acceptance, preoperational, and surveillance testing meet the intent of ASME N509 and N510.

(5) Unfiltered inleakage will be controlled by the use of welded filtration housings (GE has stated that the advanced design of the filter housing and flow pattern would virtually eliminate any untreated filter bypass).

(6) SGTS dose analysis is bounded by the maximum system flow.

Therefore, DFSER Open Item 6.5.1-1 is resolved.

During its meeting with the staff on May 5, 1992, GE committed to provide an analysis to demonstrate that the use of the system during the inerting, de-inerting, pressure control, or purging of the primary containment during normal plant operation will not impair its functional capability during a DBA. Therefore, the staff identified the submittal of the analysis as Open Item 6.5.1-2 in the DFSER. However, the staff considers that the analysis need not be performed, provided the use of the SGTS during power operation is limited to no more than 90 hours per year (approximately 1 percent of the time). BY SSAR Amendment 32, GE stated that normal operation of the SGTS would be much less than 90 hours per year for both trains combined during startup, power, and hot shutdown modes of operation. However, if 90 hours of operation per year for either train is to be exceeded, the COL applicant is required to provide functional damage analyses to demonstrate that the SGTS is capable of performing its intended function in the event of a LOCA, since this would be a change to SSAR certified design information. The staff finds this acceptable. Therefore, DFSER Open Item 6.5.1-2 is resolved.

The staff concludes that the SGTS has a removal efficiency of 99 percent for all forms of radioiodine. It further concludes that the system meets the acceptance criteria of SRP Section 6.5.1 and is, therefore, acceptable.

6.5.2 Containment Spray System

SRP Section 6.5.2 does not apply to the ABWR plant.

6.5.3 Fission Product Control Systems and Structures

The staff reviewed fission product control systems and structures in accordance with SRP Section 6.5.3. The ABWR design includes two fission product control systems and structures. These are the primary containment, which includes the suppression pool with scrubbing and retention capability, and the secondary containment, which includes the SGTS with filtration capability.

Primary Containment: The primary containment is a cylindrical steel-lined reinforced-concrete structure that forms a limited leakage boundary for fission products released to the containment atmosphere following a LOCA or any other accident that releases lesser amounts of fission products. The structure is divided by a reinforced-concrete diaphragm floor and the reactor vessel pedestal into the upper and lower drywell, and a suppression chamber (wetwell). The diaphragm floor will be rigidly attached to the reactor pedestal and the containment wall. The diaphragm floor includes a liner to prevent steam bypass from the upper drywell to the suppression chamber airspace during any accident. The primary containment is totally enclosed within the reactor building, a portion of which forms the secondary containment. GE has assumed a design leak rate of 0.5 percent per day of the free containment volume for the duration of the accident (for further discussion on leak rate, see Section 15.4.4.1 of this report). A test program will be implemented to confirm leak integrity of the primary containment structure (see Section 6.2.6 of this report). The primary containment provides a passive barrier to limit leakage of airborne radioactive material following a LOCA by immediate closure of containment isolation valves except for those necessary for ECCS and ESF functions. Information on the primary containment design, its isolation methods, and isolation times are given in Sections 6.2.1 and 6.2.4 of this report. This information includes an evaluation of the suppression pool bypass leakage area of 0.0046 m^2 (0.05 ft^2) assumed by GE. GE originally assumed a decontamination factor (DF) of 10 for the elemental and particulate forms of iodine resulting from scrubbing and retention in the suppression pool. In its radiological consequence analysis, the staff conservatively assumed a decontamination factor of 2 (equivalent to suppression pool steam bypass of 50 percent). The staff finds that the assumed pool DF of 2 for elemental and particulate forms of radioactive iodine meets the regulatory position in SRP Section 6.5.5, "Pressure Suppression Pool as a Fission Product Cleanup System," and is acceptable as discussed in Section 15.3 of this report. Therefore, the design features of the primary containment discussed in this section are acceptable for controlling fission product release during an accident.

Secondary Containment: The secondary containment is a reinforced-concrete building that forms an envelope surrounding the primary containment above the basemat. It encloses the penetrations through the primary containment and all those systems external to the primary containment that could become a potential source of radioactive release after an accident. Following an accident, the secondary containment normal HVAC paths are secured. The SGTS will maintain the secondary containment at a negative pressure of 6.4 mm (1/4 in.) of water gauge or greater relative to the surrounding spaces within 20 minutes from the time the secondary containment is isolated and process the effluent gases through a filter train to remove airborne iodines and particulates. GE has assumed a draw-down time of 20 minutes for achieving this negative pressure in its LOCA dose analysis. GE states in SSAR Section 6.5.1.3.1(5) that the COL applicant will perform a secondary containment draw-down analysis to demonstrate the capability of the SGTS to maintain the design negative pressure following a LOCA, including inleakage from the open, nonisolated penetration lines identified during construction engineering and in the event of the worst single failure of a secondary isolation valve to close. The SGTS will filter airborne fission products (iodines and particulates) leaking from the primary containment before their release to the environs. GE assumed an inleakage rate of 50 percent of the secondary containment free volume per day at a differential pressure of 6.4 mm (1/4 in.) water gauge with respect to the surrounding spaces. The design of the secondary containment is discussed in Section 6.2.3 of this report. The design of the SGTS is discussed in Section 6.5.1 of this report.

On the basis of the above information and its evaluation of the SGTS in Section 6.5.1 of this report, the staff concludes that GE has demonstrated 99-percent removal efficiency by the SGTS filter train for all forms of iodine. The staff further concludes that the fission product control systems and structures provided in the ABWR design have the capability to reduce the DBA doses to within 10 CFR Part 100 limits (for further information, see Chapter 15 of this report).

The staff concludes that the fission product control systems and structures provided for the ABWR meet the acceptance criteria of SRP Section 6.5.3 and are, therefore, acceptable.

6.6 Inservice Inspection of Class 2 and 3 Components

A detailed evaluation of the inservice inspection (ISI) of the reactor coolant pressure boundary is given in Section 5.2.4 of this report.

The staff's evaluation of SSAR Section 6.6 was not complete in the DSER (SECY-91-153) and was therefore identified as Open Item 31. The staff reviewed SSAR Section 6.6, and Open Item 31 was resolved based on the following discussion.

SSAR Section 6.6 and Table 6.6-1 describe certain commitments and plans for the preservice inspection (PSI) and ISI programs for ASME Code, Class 2 and 3 components. GE discussed basic inspection concepts and general ASME Code provisions because the requirements of the NRC regulations might be controlled by the date of order of each specific component subject to examination. The staff review was performed in accordance with SRP Section 6.6, except as explained below.

Throughout the service life of the plant, the COL applicant has the overall responsibility for the ISI of the Class 2 and 3 components. However, other organizations also contribute to the examination activity, including the reactor vendor and the architect-engineer. The NRC staff and GE discussed the issue of ISI requirements on March 25 and 26, 1992, and decided that the COL applicant is responsible for the development of the psi and ISI programs for all ASME Code, Class 2 and 3 components. This was identified as DFSER COL Action Item 6.6-1.

GE included this information in SSAR Section 6.6.9, which states that the COL applicant will develop psi and ISI program plans as outlined in Section 6.6 of the SSAR. This is acceptable.

6.6.1 Requirements of 10 CFR 50.55a, "Codes and Standards"

Pursuant to 10 CFR 50.55a(g)(3), ASME Code Class 2 and 3 components should be designed and provided with access to enable the performance of inservice examination of such components and should meet the preservice examination requirements in ASME Code, Section XI, applied to the construction of the particular component. The applicable construction code will be determined by paragraph NCA-1140 of ASME Code, Section III, but the construction code that is selected must be incorporated by reference in 10 CFR 50.55a(b).

All items of a nuclear power plant may be constructed to a single edition of the ASME Code, or each item may be constructed to individually specified code editions and addenda. NCA-1140 of ASME Code, Section III, states, in part, that in no case should the ASME Code edition established in the design specification be earlier than 3 years before the date that the nuclear power plant construction permit application is docketed. In addition, NCA-1140 permits the use of later code editions by mutual consent of the owner and certificate holder, that is, an organization holding a valid N, NV, NPT, or NA certificate of authorization issued by ASME.

Throughout the service life of a boiling or pressurized water-cooled nuclear power facility, components (including supports) that are classified as ASME Code, Class 2 and 3, must meet the requirements, except design and access provisions and preservice examination requirements, set forth in ASME Code, Section XI, that become effective after the editions specified in 10 CFR 50.55a(g)(3) and are incorporated by reference in 10 CFR 50.55a(b), to the extent practical within the limitations of design, geometry, and materials of construction of the components. The inservice examination of components conducted during the initial 10-year interval must comply with the requirements of the latest edition and addenda of ASME Code, Section XI, incorporated by reference in 10 CFR 50.55a(b) on the date 12 months before the date of issuance of the operating license. The regulations also require that the ISI program be updated for each subsequent 10-year interval to comply with the ASME Code incorporated by reference in 10 CFR 50.55a.

6.6.2 Application of the Codes and Standards Rule to Standard Design Certification

Compliance with the requirements of the regulations for PSI and ISI must be based on commitments for or by the COL applicant. The applicable ASME Code(s) for both design and ISI are not known at the time of the design certification application. In addition, the regulation permits the COL applicant the option to change to a newer edition of ASME Code, Section XI, by component, during the construction of the plant.

The staff reviewed the information in the SSAR that emphasized design access and the use of effective nondestructive examination (NDE) methodology. Because PSI requirements are established and known at the time each component is ordered, 10 CFR 50.55a(g) does not have provisions for relief requests for impractical examination requirements. ASME Code, Section XI, does have provisions to use certain shop and field examinations in lieu of the onsite preservice examination. Therefore, the COL applicant will incorporate plans for NDE during design and construction in order to meet all access requirements of the regulations. This was identified as DFSER COL Action Item 6.6.2-1.

GE included this information in SSAR Section 6.6.9, which states that the COL applicant will incorporate plans for NDE during design and construction in order to meet all access requirements of the regulations. This is acceptable.

ASME Code, Section XI, states that the preservice examination should be conducted with equipment and techniques equivalent to those that are expected to be used for subsequent inservice examinations. Improvements in the ultrasonic testing of Class 2 and 3 components will occur in the near future. ASME has published ASME Code, Section XI, Appendix VII, "Qualification of Nondestructive Examination Personnel for Ultrasonic Examination," and Appendix VIII, "Performance Demonstration for Ultrasonic Examination Systems." The NRC staff has published in the Federal Register its intent to reference in 10 CFR 50.55a(b) the edition of ASME Code, Section XI, that includes the published Appendix VII. In addition, the staff has established a technical contact to coordinate the implementation of Appendix VIII. Therefore, the PSI program for the ABWR should include provisions that ultrasonic testing be performed in accordance with Appendices VII and VIII pursuant to 10 CFR 50.55a(g)(3). However, the staff did not identify the above as a separate issue because it was covered by DFSER Open Item 5.2.4-1. See Section 5.2.4 of this report.

The requirements for the initial ISI program will be determined by the ASME Code in effect 1 year before the issuance of an operating license. The COL applicant is required to meet the ISI requirements "to the extent practical within the limitations of design, geometry and materials of construction of the components." The regulations have provisions for the staff evaluation, on written request, of new ASME Code requirements determined by a licensee to be impractical for its facility.

6.6.3 Staff Evaluation

A PSI program and an ISI program will be prepared for each ABWR plant. GE states that provisions have been made in the design and layout of ASME Code, Class 2 and 3 systems to allow for access for the examinations required in ASME Code, Section XI, Articles IWC-2000 and IWD-2000, and as defined in the ISI program. As a result of a meeting between GE and the NRC staff on March 25 and 26, 1992, it was determined that the COL applicant is responsible for the development of the PSI and ISI programs for all ASME Code, Class 1, 2, and 3 &components. The NRC staff reviewed the above information regarding design access for the preservice examination and finds the information pertaining to planning adequate. The staff notes that the SSAR Section 6.6 states that the COL applicant will meet the

requirements of the NRC regulations defined in 10 CFR 50.55a.

The SSAR Section 6.6.3.2 states that the examination techniques to be used for ISI will include radiographic, ultrasonic, magnetic particle, liquid penetrant, eddy current, and visual examination methods. For all examinations, both remote and manual, specific procedures will be prepared describing the equipment, inspection technique, operator qualifications, calibration standards, flaw evaluation, and records. These techniques and procedures will meet the requirements of Articles IWC-2000 and IWD-2000 in the edition of ASME Code, Section XI, in effect as stated in 10 CFR 50.55a(g). GE has revised SSAR to state that personnel performing ultrasonic examinations will be qualified in accordance with ASME Code, Section XI, Appendix VII. Ultrasonic examination systems will be qualified in accordance with an industry-accepted program for implementing ASME Code, Section XI, Appendix VIII. GE considered the supplemental examinations recommended in GE service information letters (SILs) and rapid communication service information letters (RICSILs) for previous BWR designs as not applicable to the ABWR. In the ABWR design either of the following has been eliminated: components addressed by the SILs or RICSILs or the need for the examination by eliminating crevice designs and using materials resistant to the known degradation mechanisms, such as intergranular stress corrosion cracking, on which the SIL and RICSIL examinations were based.

The staff finds these commitments acceptable.

6.7 High-Pressure Nitrogen Gas Supply System

The ABWR design includes four compressed air systems: the instrument air system, the service air system, the high-pressure nitrogen (HPIN) gas supply system, and the atmospheric control system (ACS). In SSAR Section 6.7, GE supplies information on the design of the HPIN system. The staff reviewed the information in accordance with SRP Section 9.3.1.

The HPIN system consists of both a nonessential (i.e., non-safety-related) and an essential systems. A single nonessential system will provide a continuous nitrogen supply to all pneumatically operated components in the primary containment during normal operation. As noted in Section 6.5.2 of this report, during normal operation, the nitrogen gas evaporator/storage tank will supply the HPIN system via the makeup line to the ACS. The essential system has two independent divisions. Each division contains a safety-related emergency stored nitrogen supply capable of supplying 100 percent of the requirements of the division being serviced. Nitrogen gas for the essential system will be supplied from HPIN gas storage bottles. There are tielines between the nonessential system and each division of the essential system. Each tieline has a motor-operated shutoff valve.

Because the HPIN system is one of the four systems that will perform the functions addressed in SRP Section 9.3.1, the review of this system was performed as part of an integrated review of the ABWR compressed air systems. The results of this review are given in Section 9.3.1 of this report.

6.8 Main Steam Isolation Valve Leakage Control System

An MSIV leakage control system will not be used in the ABWR, as discussed in Section 10.3 of this report.

7 INSTRUMENTATION AND CONTROLS

7.1 Introduction

The staff has reviewed the GE Nuclear Energy (GE) advanced boiling water reactor (ABWR) standard safety analysis report (SSAR) and associated reference material. A draft safety evaluation report (DSER) was issued on September 18, 1991, attached to SECY-91-294, "Draft Safety Evaluation Report on the General Electric Boiling Water Reactor Design Covering Chapter 7 of the Standard Safety Analysis Report." The applicant responded to the open issues identified in the DSER by letter dated February 3, 1992. A copy of the DSER was also provided to the Advisory Committee on Reactor Safeguards (ACRS).

A draft final safety evaluation report (DFSER) was issued on October 14, 1992, attached to SECY-92-349, "Draft Final Safety Evaluation Report on the General Electric Boiling Water Reactor Design Covering Chapter 7 of the Standard Safety Analysis Report." The DFSER has also been provided to the ACRS. The applicant responded to the open issues identified in the DFSER by several submittals described in this report. The following sections describe the staff's final safety evaluation findings. This report describes the resolution of the open items from the DSER and DFSER.

The staff has reviewed Chapter 7 of the SSAR submittal through Amendment 35. Chapter 7 contains the primary instrumentation and control (I&C) descriptions and commitments. Chapter 1 of the SSAR contains most of the associated instrumentation block diagrams. A few items from other chapters of the SSAR are included in this evaluation, as indicated. The staff has reviewed both proprietary and non-proprietary information from the SSAR and other docketed materials.

7.1.1 Acceptance Criteria

The acceptance criteria used as the basis for the staff's evaluation of the I&Cs for the ABWR are set forth in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition," and 10 CFR Part 52. The primary sections of NUREG-0800 used for this portion of the review are Section 7, "Instrumentation and Control," and Section 9.5.2, "Communications Systems." The Commission issued review guidance in staff requirements memoranda (SRM), dated June 26, 1990, and February 15, 1991, pertaining to SECY-90-016, "Evolutionary Light Water Reactor Certification Issues and Their Relationship to Current Regulatory Requirements," and SECY-90-377, "Requirements for Design Certification under 10 CFR Part 52," respectively.

The staff developed the necessary acceptance criteria to cover the certification of those systems because the SRP presently does not address design certification, or the newer digital technology used in these ABWR I&C systems (the SRP was last revised in 1984). The additional acceptance criteria and conformance to the SRP are discussed in the applicable sections of this report. This report also describes certain items which will be included in the certified design material (CDM) which includes the design description; inspections, tests, analyses and acceptance criteria (ITAAC), site parameters, and interface requirements for design certification. The description of the CDM development process, bases, and acceptability is primarily discussed in Section 14.3 of this report. This report discusses only those CDM areas which relate to the I&C systems' certified design process in addition to specific design characteristics.

The staff referenced its evaluation of previously-reviewed plant designs and topical reports, as stated.

7.1.2 Method of Review

In the DSER (SECY-91-294) the staff discussed the following items: systems which have had significant design changes compared to previously-reviewed and-accepted designs; the appropriate level of detail necessary for design certification; and design issues where more information was needed in order for the staff to make a finding of acceptability. Since issuing the DSER the staff has concentrated on resolving the open issues in the DSER and reviewing the submittals of the CDM and technical specifications (TS). The DFSER (SECY-92-349) described the remaining open issues and the acceptance of design process commitments in lieu of design details for certain rapidly developing technologies such as software design.

In addition to reviewing the SSAR, the staff also visited GE's offices and reviewed documents associated with the SSAR material. All documents relied upon for the safety evaluation that are not included in the SSAR have been docketed.

7.1.3 General Findings

GE has identified the I&C systems which are important to safety in accordance with Regulatory Guide (RG) 1.70, Revision 3, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," November 1978. Each of the safety systems identified in the SSAR also has an associated Design Description and ITAAC. Some aspects of the design which affect all of the I&C systems (such as the software design process) are addressed in an I&C CDM section rather than repeated for each individual I&C system CDM. Most of the non-safety systems are also described in a CDM.

The ABWR I&C systems are significantly different from I&C systems in previously-approved designs. The primary differences result from using digital microprocessor-based I&C systems with multiplexed data links in place of the analog electronics, relay logic, and hardwired systems previously approved by the staff. The staff previously reviewed designs using digital equipment similar to that proposed for the ABWR; however, the ABWR significantly extends the quantity and scope of system coverage of the digital microprocessor applications and, in conjunction with the 10 CFR Part 52 design certification process, requires a unique review which is not directly comparable with any past licensing review. The differences from past license reviews are address in greater detail in this chapter.

The comparison of the ABWR to the GESSAR II design in SSAR Table 7.1-1 is not a sufficient basis for accepting the ABWR I&C design because, as the table shows, of the significant differences in the implementation of the I&C system design when using the ABWR digital equipment.

7.1.3.1 Standard Review Plan Criteria

The acceptance criteria listed in Table 7.1 of this report are identified in the Commission regulations and Standard Review Plan (SRP) Chapter 7, "Instrumentation and Controls." The SRP also provides some review guidance and acceptance criteria not found in the specified standards, General Design Criteria (GDC) and other references. Table 7.1 does not consider the degree of conformance of the ABWR with the requirements or the method of implementation of the I&C system design, but simply lists the SSAR sections where the standards and criteria are mentioned in the SSAR. Many of the standards and criteria discussed in SSAR Chapter 7 and Appendices A, B, and C are applicable to the previous SSAR sections which describe the I&C systems without being specifically listed and discussed in those sections. The request for additional information (RAI) column refers to the staff RAI and the GE response to them which relate to specific standards and criteria. The RAIs and responses are provided in Chapter 20 of the SSAR. The conformance to and any exceptions from the standards and criteria are discussed in this chapter. SSAR Table 7.1-2, "Regulatory Requirements Applicability Matrix for I&C Systems," also provides a similar matrix of the SRP criteria to that in this report, but compares them with the I&C systems rather than the SSAR sections. In general, GE committed to meet the SRP requirements without exception. The few exceptions GE requested are noted in SSAR Section 1.8 and the applicable sections of this chapter.

10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," contains the GDC. Chapter 3.1 of the SSAR discusses compliance with the GDC generally and references other SSAR chapters for specifics. Table 7.1-2 of the SSAR lists the GDC applicable to the I&C systems.

The SSAR Chapter 3 discussion of GDC 3, "Fire Protection;" GDC 26, "Reactivity Control System Redundancy and Capability;" GDC 27, "Combined Reactivity Control System Capability;" GDC 30, "Quality of Reactor Coolant Pressure Boundary;" GDC 37, "Testing of Emergency Core Cooling System;" GDC 42, "Inspection of Containment Atmosphere Cleanup Systems;" GDC 43, "Testing of Containment Atmosphere Cleanup Systems;" GDC 55, "Reactor Coolant Pressure Boundary Penetrating Containment;" and GDC 56, "Primary Containment Isolation;" referred to Chapter 7 of the SSAR for further discussion of each GDC. However, these GDC did not appear in the SSAR Chapter 7 GDC discussions. The topics generally are discussed in other sections of the SSAR.

GDC 2, "Design Bases for Protection Against Natural Phenomena;" GDC 15, "Reactor Coolant System Design;" GDC 16, "Containment Design;" GDC 38, "Containment Heat Removal;" and GDC 44, "Cooling Water," were listed in Chapter 7, but there was no corresponding reference to Chapter 7 in the Chapter 3 evaluations. The GDC discrepancies were listed as Open Item 7.1.3.1-1 in the DFSER. GE revised the SSAR to include references in Chapter 3 to Chapter 7 for GDC 2, 15 and 38. GE also clarified that the GDC discussed in Chapter 3 apply to the equipment included in Chapter 7, regardless of whether there is a specific discussion in the I&C sections in Chapter 7 or not. The staff concludes that this is acceptable; therefore, this open item is resolved.

The specific standards to which GE has committed are a significant consideration for the staff's safety findings. The most important aspects of those criteria, which are required to be certified by rulemaking, are included in the CDM and are discussed in the staff's evaluation.

7.1.3.2 Electric Power Research Institute Requirements Document

Electric Power Research Institute (EPRI) prepared a document of technical requirements, referred to as the advanced light water reactor (ALWR) Utility Requirements Document (ALWR RD), intended by EPRI to be applies to the design of ALWR power plants, including the ABWR.

Table 7.1 SRP Acceptance Criteria

SRP CRITERIA GDC	ABWR SSAR Chapter 7											
	7.1	7.2	7.3	7.4	7.5	7.6	7.7	7.8	7.A	7.B	7.C	RAI
10CFR50.55A	X	X	X	X	X	X			X			
10CFR50APPB	X	X	X	X	X	X			X	X		X
GDC(GENERIC)	X	X	X	X	X	X			X	X		
GDC 1	X	X	X	X	X	X			X	X		X
GDC 2	X	X	X	X	X	X			X			X
GDC 4	X	X	X	X	X	X			X			X
GDC 10	X	X				X						
GDC 13	X	X	X	X	X	X	X		X			
GDC 15	X	X	X	X		X	X		X			
GDC 16	X	X	X			X			X			
GDC 19	X	X	X	X	X	X	X		X			
GDC 20	X	X	X	X		X						
GDC 21	X	X	X			X						
GDC 22	X	X	X			X						
GDC 23	X	X	X			X						
GDC 24	X	X	X			X						
GDC 25	X	X		X		X						
GDC 28	X					X			X			
GDC 29	X	X	X			X			X			
GDC 33	X		X	X		X			X			
GDC 34	X		X	X		X						
GDC 35	X		X	X		X						
GDC 38	X		X			X						
GDC 41	X		X			X						
GDC 44	X		X			X						

IEEE STANDARDS IN SRP	ABWR SSAR Chapter 7											
	7.1	7.2	7.3	7.4	7.5	7.6	7.7	7.8	7.A	7.B	7.C	RAI
279-1971	X	X	X	X	X	X	X		X			
317-1972												X
323-1974	X	X							X			
336-1971												
338-1971	X	X	X	X	X	X			X			
379-1972	X	X	X									
384-1974	X	X	X	X	X	X	X		X			

OTHER GUIDANCE IN SRP	ABWR SSAR Chapter 7											
	7.1	7.2	7.3	7.4	7.5	7.6	7.7	7.8	7.A	7.B	7.C	RAI
ANSI/ANS 4.5					X							
ANSI N41.15	X	X	X	X					X			
ISA 67.02	X	X	X	X	X	X	X		X			
NUREG-0694			X		X							
NUREG-0696	X	X	X	X	X	X	X		X			
NUREG-0718	X	X	X	X	X							
NUREG-0737	X	X	X	X	X	X	X		X			

Table 7.1 SRP Acceptance Criteria (continued)

TMI ACTION PLAN ITEMS IN SRP	ABWR SSAR Chapter 7											
	7.1	7.2	7.3	7.4	7.5	7.6	7.7	7.8	7.A	7.B	7.C	RAI
II.D.3	X		X		X							
II.E.1.2	NOT APPLICABLE											
II.E.4.2	X		X									
II.F.1	X				X	X						
II.F.3	X		X		X	X						
II.K.1	X	X	X									X
II.K.2	NOT APPLICABLE											
II.K.3	X	X	X				X					
REGULATORY GUIDES IN SRP	ABWR SSAR Chapter 7											
	7.1	7.2	7.3	7.4	7.5	7.6	7.7	7.8	7.A	7.B	7.C	RAI
RG 1.22	X	X	X	X		X						
RG 1.47	X	X	X	X		X			X			
RG 1.53	X	X	X	X		X			X			
RG 1.62	X	X	X	X	X	X			X			
RG 1.75	X	X	X	X	X	X	X	X	X			
RG 1.89	X	X	X	X	X	X			X			
RG 1.97	X		X		X	X						X
RG 1.105	X	X	X	X	X	X						
RG 1.118	X	X	X	X	X	X						
RG 1.151	X	X	X				X					
BRANCH TECHNICAL POSITION IN SRP	ABWR SSAR Chapter 7											
	7.1	7.2	7.3	7.4	7.5	7.6	7.7	7.8	7.A	7.B	7.C	RAI
BTP ICSB 3	X		X	X		X						
BTP ICSB 4	NOT APPLICABLE											
BTP ICSB 12	X	X										
BTP ICSB 13	NOT APPLICABLE											
BTP ICSB 14	NOT APPLICABLE											
BTP ICSB 16	DELETED											
BTP ICSB 20	X		X									
BTP ICSB 21	X	X	X			X						
BTP ICSB 22	X	X	X			X						
BTP ICSB 26	X	X										

Volume II, "Evolutionary Plant," specifically applies to the ABWR design. By letter dated June 12, 1990, GE provided a comparison of the ALWR RD, Volume II, and the ABWR SSAR. The EPRI requirements pertaining to the I&C systems are contained primarily in Chapter 10 of Volume II of the ALWR RD, "Man-Machine Interface Systems." GE concluded in their letter that a detailed comparison of the ABWR SSAR to the many ALWR RD requirements showed that the ABWR complied with all but a few of the EPRI ALWR requirements.

GE did not document its detailed comparison, nor did the staff compare the EPRI's requirements to the entire ABWR design for applicability. In addition, there are many requirements in the ALWR RD without any corresponding references to published standards. There are also many requirements in the ALWR RD which allow a vendor different options for implementation in a design, or provide a caveat that the requirement should be implemented to the extent practical. In August 1992, the staff issued the final safety evaluation report for the

Volume II of the ALWR RD, and requested that GE clearly document in detail how the ABWR design differs from that in the ALWR RD and also how it complies with EPRI's requirements. This was identified as DFSER Open Item 1.1-1. The resolution of this issue is described in Section 1.1 of this report.

7.1.3.3 Additional Criteria

Table 7.2 of this report lists criteria and standards discussed in the ABWR SSAR but are not included in the SRP criteria in Table 7.1 of this report. However, some of the standards in Table 7.2 are referenced in standards included in Table 7.1 and, therefore, could be considered as part of the SRP listings. They are listed in Table 7.2 for convenience and completeness. Table 7.2 also includes the standards and criteria which GE has committed to use since the DSER was issued to address the new technology issues not addressed in the SRP. Some of these items are included in SRP chapters other than Chapter 7. Additional comments regarding use of future standards to accommodate changing I&C digital technology are discussed elsewhere in this chapter.

ABWR SSAR Table 7.1-2, "Regulatory Requirements Applicability Matrix for I&C Systems," lists the ABWR I&C systems and the requirements which apply to each system. The safety system logic and control (SSLC) and essential multiplexing system (EMS) are not explicitly defined in SSAR Table 7.1-2. The staff noted that GE has defined the "reactor protection (trip) system (RPS)" as a system that provides sensor signals and reactor trip outputs, and that uses the SSLC as the means for implementing trip logic. The SSLC in turn uses the EMS as the means for interfacing with sensors and actuated equipment except where the devices are hardwired. Since the commitment to the SRP standards and criteria as shown in SSAR Table 7.1-2 includes the SSLC and EMS, the staff finds the commitment acceptable.

In the DSER, the staff determined that SSAR Table 7.1-2 provides commitments to the acceptance criteria prescribed by the SRP with one exception. The table in the SSAR did not include an explicit commitment to GDC 1, "Quality Standards and Records," of 10 CFR Part 50. Although GDC 1 was addressed in Chapter 3 of the SSAR, the staff believed that an explicit commitment should be provided in Chapter 7 and identified this as an open issue in the DSER (Open Item 5). Specifically, GDC 1 requires that systems and components important to safety be designed, fabricated, and tested to quality standards commensurate with the importance of the safety functions to be performed. In addition, this criterion also requires that GE identify and evaluate applicability, adequacy, and sufficiency of generally recognized codes and standards.

In the DSER, the staff concluded that acceptance criteria for software-based digital I&C systems were not addressed in sufficient detail by the regulatory positions and the industry standards referenced by GE in the SSAR at the time the DSER was written. No standards were identified regarding electromagnetic compatibility (EMC), multiplexor architecture, communications protocols, and software design. GE has since committed to industry standards listed in SSAR Appendix 7A for these areas, and the staff concludes that the ABWR design conforms with GDC 1 in that GE has committed to use the appropriate industry standards for the type of equipment to be used in the implementing the design. Therefore, this open issue is resolved.

SRP Appendix 7-B, "General Agenda, Station Site Visits," includes the following statement:

> An important part of the review at the operating license stage is a site visit. It is preferable to have the site visit sometime before the completion of the drawing review. The purpose of the site visit is to supplement the review of the design based on the drawings and to evaluate the actual implementation of the design as installed at the site. The NRC Regional Office having jurisdiction over the plant under consideration should be notified ahead of time of the visit so that the regional inspectors can become familiar on a first-hand basis with findings that may require followup action. Since proper implementation of design is the ultimate goal of the technical review process, the importance of a site visit is self-evident.

SRP Appendix 7-B provides a general agenda for the station site visit which includes verification of layouts, separation and isolation, test features and potential for damage due to fire and flooding, or other environmental effects. Because the design certification will be issued prior to the selection of a construction site, this SRP review item cannot be completed at this time. The method to be used to address the necessary inspection tasks for design certification will be addressed through the ITAAC process and commitments to preoperational tests described in SSAR Chapter 14. The review described in SRP Appendix 7-B will be accomplished as part of the testing done for the combined license (COL).

Table 7.2 Chapter 7 Additional Requirements

CRITERIA/ STANDARD	ABWR SSAR Chapter 7											
	7.1	7.2	7.3	7.4	7.5	7.6	7.7	7.8	7.A	7.B	7.C	RAI
GDC 12						X						
GDC 17												X
GDC 18												X
GDC 54					X							
GDC 55					X							
GDC 56					X							
GDC 57					X							
RG 1.26		X										
RG 1.28	X	X	X	X	X	X			X			
RG 1.30	X	X	X	X	X	X			X			
RG 1.32	X	X	X	X	X	X			X			
RG 1.38	X	X	X	X	X	X			X			
RG 1.58					X							
RG 1.64	X	X	X	X	X	X			X			
RG 1.70	X	X	X	X	X	X	X	X	X			X
RG 1.74					X							
RG 1.88	X	X	X	X	X	X			X			
RG 1.100	X	X	X	X	X	X			X			
RG 1.123	X	X	X	X	X	X			X			X
RG 1.144	X	X	X	X	X	X			X			
RG 1.146					X							
RG 1.152	X	X	X	X	X	X			X	X		X
RG 1.153	X	X	X	X	X	X			X			X
IEC 801-2	X	X	X	X	X	X			X			X
IEC 880										X		
IEEE 344	X	X	X	X	X	X						
IEEE 352	X	X										X
IEEE 472	X	X	X	X	X	X			X			X
IEEE 518		X										X
IEEE 603	X	X	X	X	X	X			X			X
IEEE 730										X		
IEEE 7-4.3.2	X	X	X	X	X	X			X	X		X
IEEE 802.2	X	X	X	X	X	X			X			X
IEEE 802.5	X	X	X	X	X	X			X			X
IEEE 828	X	X	X	X	X	X			X	X		X
IEEE 829	X	X	X	X	X	X			X	X		X
IEEE 830	X	X	X	X	X	X			X	X		X
IEEE 1012	X	X	X	X	X	X			X	X		X
IEEE 1033	X	X	X	X	X	X			X	X		X
IEEE 1042	X	X	X	X	X	X			X	X		
IEEE 1228	X	X	X	X	X	X			X	X		
NEMA 4		X	X									
ASME NQA 2.7										X		
NEDO 24708			X									
NEDC 31336		X										X
NEDO 31439												X

Table 7.2 Chapter 7 Additional Requirements (continued)

CRITERIA/ STANDARD	ABWR SSAR Chapter 7											
	7.1	7.2	7.3	7.4	7.5	7.6	7.7	7.8	7.A	7.B	7.C	RAI
NEDE 31906				X								X
ANS 4.5					X							
NUREG 0493	X	X	X	X	X	X	X	X	X	X	X	X
NUREG 0588					X							
NUREG CR3453									X			
NUREG -0308									X			
NUREG -0491									X			
ISA 67.04	X	X	X	X	X	X				X		X
10CFR21	X	X	X	X	X	X				X		X
MIL HDBK 251	X	X	X	X	X	X				X		X
MIL HDBK 217E	X	X	X	X	X	X				X		X
MIL STD 2167	X	X	X	X	X	X			X	X	X	X
MIL STD 461C	X	X	X	X	X	X				X		X
MIL STD 462	X	X	X	X	X	X				X		X
10CFR50.46	X		X									
10CFR50.49	X	X	X	X	X	X				X		X
10CFR50.62				X						X		X
GL 83-08										X		X
GL 84-23										X		X
ISO 7498		X							X	X		X
ANSI C37.90.2	X	X	X	X	X	X			X	X		X
ANSI C62.41	X	X	X	X	X	X			X	X		X
ANSI C62.45	X	X	X	X	X	X			X	X		X
ANSI C63.12	X	X	X	X	X	X			X	X		X
ANSI X3T9.5	X	X	X	X	X	X			X	X		X

Because the ABWR has been submitted for design certification, the requirements of 10 CFR Part 52 apply in addition to those of 10 CFR Part 50. 10 CFR Part 52 requires a level of design detail beyond a simple commitment to conformance with the existing requirements. 10 CFR 52.47(a)(2) requires that:

> The application must contain a level of design information sufficient to enable the Commission to judge the applicant's proposed means of assuring that construction conforms to the design and to reach a final conclusion on all safety questions associated with the design before the certification is granted. The information submitted for a design certification must include performance requirements and design information sufficiently detailed to permit the preparation of acceptance and inspection requirements by the NRC, and procurement specifications and construction and installation specifications by an applicant. The Commission will require, prior to design certification, that information normally contained in certain procurement specifications and construction and installation specifications be completed and available for audit if such information is necessary for the Commission to make its safety determination.

10 CFR 52.47(b)(1) also states that, ". . . this rule must provide an essentially complete nuclear power plant design except for site-specific elements." The following sections of this report describe the information provided by GE and the staff's conclusions concerning conformance with the SRP criteria, additional criteria necessary to address newer digital I&C technology, and the above requirements of 10 CFR Part 52.

In the DSER, the staff concluded that GE had adequately committed to the SRP requirements but had not provided design information detailed enough for the staff to conclude that the plant will be built in accordance with those standards. This was identified as an open issue in the DSER. The resolution of this open issue was based on a digital I&C system design process and the use of a phased approach to design implementation through the ITAAC. In addition, certain restrictions on changing SSAR commitments have been incorporated to ensure staff agreement prior to any changes to key details of the design process. This approach formed a substantial part of the basis for staff's acceptance of the I&C systems for certification under 10 CFR Part 52.

The CDM describes the basic design process to be followed during the design of the I&C hardware and software and will include specific acceptance criteria. The design process CDM will be in the same format as the system CDMs. The I&C system design process is described below. Additional general information on this aspect of the review process is provided in SECY-92-287, "Form and Content for a Design Certification Rule."

CDM Process

The ABWR I&C systems described in the SSAR lack substantial design detail. Therefore, the following descriptions of the process in the SSAR and CDM for development of the detailed design, together with certain SSAR change restrictions are key features of the final safety determination for the I&C systems in this report and are included in the design certification rule for the ABWR. The staff's audit and approval function following design certification is also a critical feature of the acceptance of the CDM and SSAR descriptions for the I&C systems. The staff proposed in the DFSER that this design process be described in the appropriate CDM. The CDM has been revised to include the phased development process, however, the staff audit participation is only described in this report.

In the design description section of the CDM, a textual and graphical description of the fundamental characteristics of the various I&C systems is provided. The CDM includes a summary of the systems and design commitments which will be certified for the life of the plant. The items in the design description cannot be changed over the life of the plant without rulemaking and, therefore, the details of the design which can be changed are not described. The ITAAC section of the CDM provides a general description of the activities that will be performed by the COL applicant to verify that the I&C system has been built in accordance with the certified design description and commitments. The SSAR provides additional information on the implementation of the design. The SSAR provides additional detail (unlike the CDM) by reference to specific standards for the designers (COL applicant) to follow. The staff will audit the COL applicant's activities at appropriate review points to ensure that the detailed design is in conformance with the design descriptions and ITAAC.

It is premature to complete the final design details for the microprocessor and digital control technology aspects of the I&C system design of the ABWR before the COL is issued. This is 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 approach that will be used is to "lock in" a design process and the specific acceptance criteria which, if met, would result in a design which is acceptable. At the same time, the functional system description, commitment to standards, and commitment to a structured design process in the CDM must be sufficient for the staff to make its final safety determination. This process was described in SECY-92-053, "Use of Design Acceptance Criteria During 10 CFR Part 52 Design Certification Reviews."

The CDM was not submitted by GE as part of the original design certification application. Subsequently, a small group of system design description and ITAAC were submitted as a pilot program in an effort to reach agreement on the general scope of the design description and ITAAC for all systems. The staff reviewed the RPS design description and ITAAC as representative of the I&C systems. The staff provided comments whose resolutions were incorporated into the documents. In general, the descriptions and ITAAC that were proposed by GE were not sufficiently detailed or complete for a staff safety finding. Therefore, the staff concluded that it was necessary to review both the SSAR and the CDM and to require more CDM information in order to reach a safety conclusion. Because the CDM was still under review when the DFSER was issued, the staff identified this as DFSER Open Item 7.1.3.3-1. The CDM has been submitted, reviewed, and found acceptable. The acceptance of the CDM is addressed in Section 14.3 of this report. Therefore, this item is resolved.

GE has not finalized the hardware and software design for the ABWR digital I&C systems. Therefore, the staff used the two-part approach described in SECY-92-053 to reach its safety finding for design certification. In reviewing the I&C systems, the first part of this approach involved a detailed functional review at the block diagram level to ensure the applicant has appropriately implemented the Commission's requirements related to postulated single failures, common mode failures, signal isolation, and other aspects of the staff's review that are typical for any safety-related I&C system, including analog control systems such as those in current operating nuclear plants. This review confirmed that the detailed functional requirements for the I&C systems are established.

The second part of the staff's approach to reach certification safety finding addresses the adequacy of the digital control systems implementation with respect to the system functional requirements. This relies upon a formal design implementation process with a phased ITAAC program for design development within certain predefined constraints and limits.

This report documents the results of the staff's review of the ABWR I&C systems at the functional block diagram level as described above for the first part of the two-part approach documented in SECY-92-053.

Certified Design Material

Because the CDM and SSAR reviews are interrelated, it is necessary for the two parts of the review to be considered integrally rather than individually. The following section uses the computer development process to illustrate the implementation of the CDM and the NRC staff involvement. This process is also used for multiplexor design, setpoint methodology, and EMC and equipment qualification verification.

Computer Development

The primary function of the computer is to implement the functional I&C requirements described in the CDM and the SSAR for the ABWR systems. The decomposition of the functional system (SSLC, RPS, ARI, etc.) requirements to specific computer hardware and software components to accomplish 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 ITAAC to be performed, and the appropriate acceptance criteria. The ITAAC for computer hardware and software (included in the I&C CDM) differs from the majority of the system ITAAC in that the acceptance criteria include acceptance criteria for the certified design process (referred to as design acceptance criteria in SECY-92-053). 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 (included in the CDM Section 3.4, "Instrumentation and Control") describes several design stages.

The following design stages (the software lifecycle) are necessary for the development and operation of both safety-related and non-safety-related software.

(1) planning
(2) requirements
(3) design
(4) implementation
(5) integration
(6) validation
(7) Installation
(8) operation and maintenance

The staff considers these stages to be necessary for the development of a computer system of sufficient quality to adequately perform its design function. The stages are based on generic activities that represent software and system lifecycle models, and encompass existing national standards and input from ongoing efforts in the revisions of national and international standards for computer systems in nuclear power plants, and expert opinion.

GE has revised the CDM and SSAR to provide a similar lifecycle definition but selected the following categories:

(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, while the SSAR adds the set of guidelines and standards that will provide more detailed criteria for the development of the design process. The CDM has been written to encompass the most important aspects from the standards. The SSAR set of standards and criteria encompass the guidance that will be used to start the computer software and hardware design process by generating the plans for the computer design throughout the lifecycle. Therefore, the staff concludes that these stages are acceptable for the development and operation of safety-related and non-safety-related computer systems.

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 Appendix B of 10 CFR Part 50, "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 DFSER identified the following documents which are typical of a software development program.

- 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 describes a development process, tool documentation, and products developed according to the software lifecycle.

- The verification and validation (V&V) plan describes the method to ensure that the requirements of each phase or stage are fully and accurately implemented in the next phase.

- The software safety plan 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 combined these plans into an SMP, a CMP, and a V&V plan. The NRC will perform audits and inspections of these plans and their implementation at each appropriate phase or stage of the lifecycle.

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. 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. Figure 7.1-1 (from Section 3.4 of the CDM) of this report shows the stages or phases described by GE. The stages described in the DFSER, including the Figure 7.1-2 shows the stages described in the DFSER, including the NRC audit and the COL applicant conformance review points. These stages correspond closely 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 the design implementation is scheduled to begin. The COL applicant is required to 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 and/or changing the system design. The NRC staff will issue an inspection report for each stage ITAAC and will identify any open issues which require resolution. Significant open issues which are not resolved could result in the NRC staff concluding that the ITAAC has not been satisfactorily completed.

At each stage, the COL applicant must verify that the design development is in accordance with the certified design process and that the detailed design developed (through that stage) is 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.

Certain ITAAC, such as those for software development, will be repeated for all the products which use software. All of the products to be verified by the ITAAC process must be complete before the ITAAC itself is complete. The COL applicant will also provide a description of the next stage of design development and associated testing, analysis, and acceptance criteria in enough detail that the NRC staff can determine whether or not the proposed design development and testing are consistent with the certified design process and the subsequent ITAAC. This phased process will continue until all ITAAC steps for all the safety-related software are complete.

The certified design description and design development process will 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 CDM, or the applicant 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. The applicant will be required to use the approved soft-ware change procedure based upon the certified design development process for the operation stage of the lifecycle. The method of incorporating software changes will be consistent with the SSAR change limitations defined in this report.

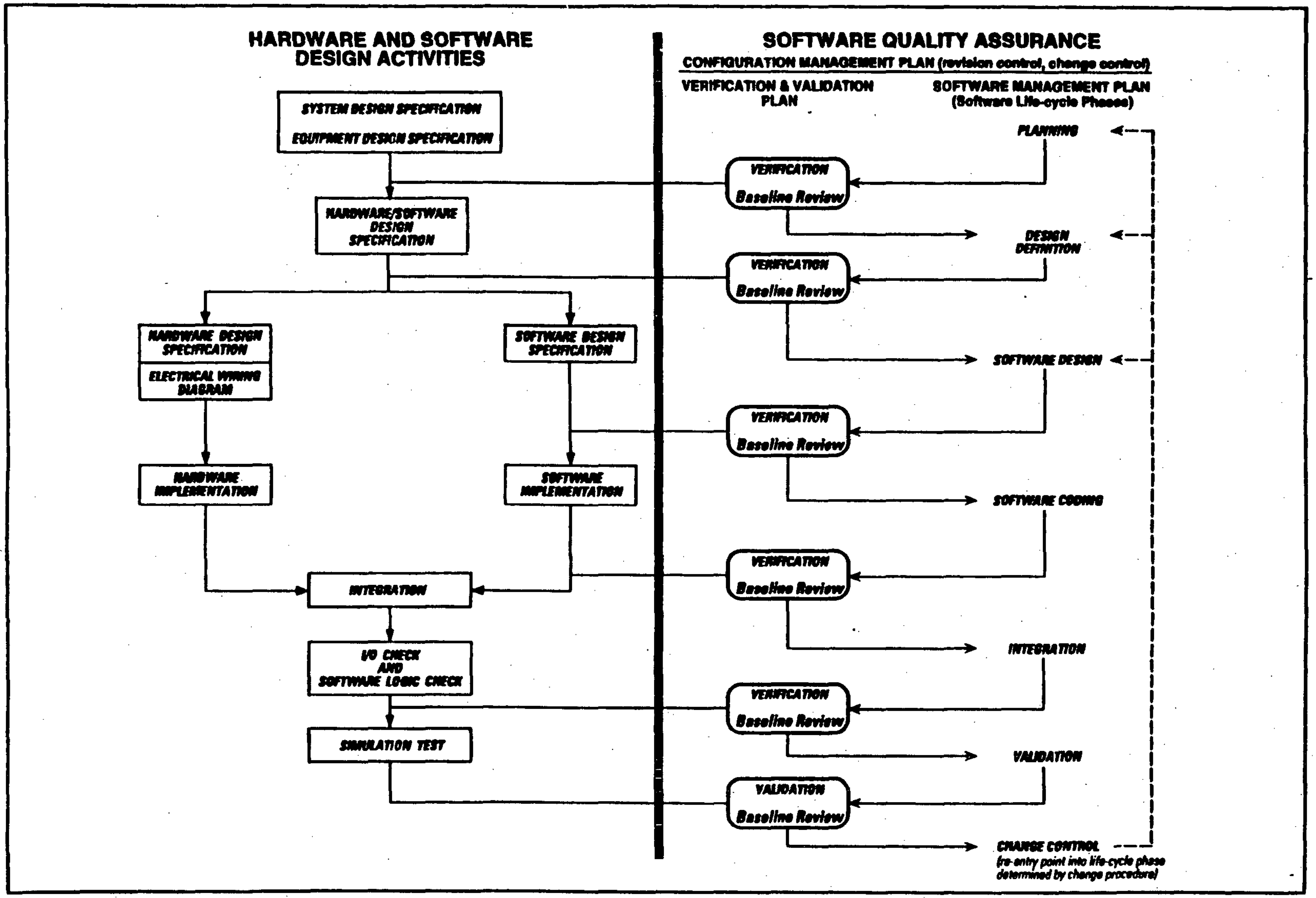


Figure 7.1-1 Integrated hardware/software development process

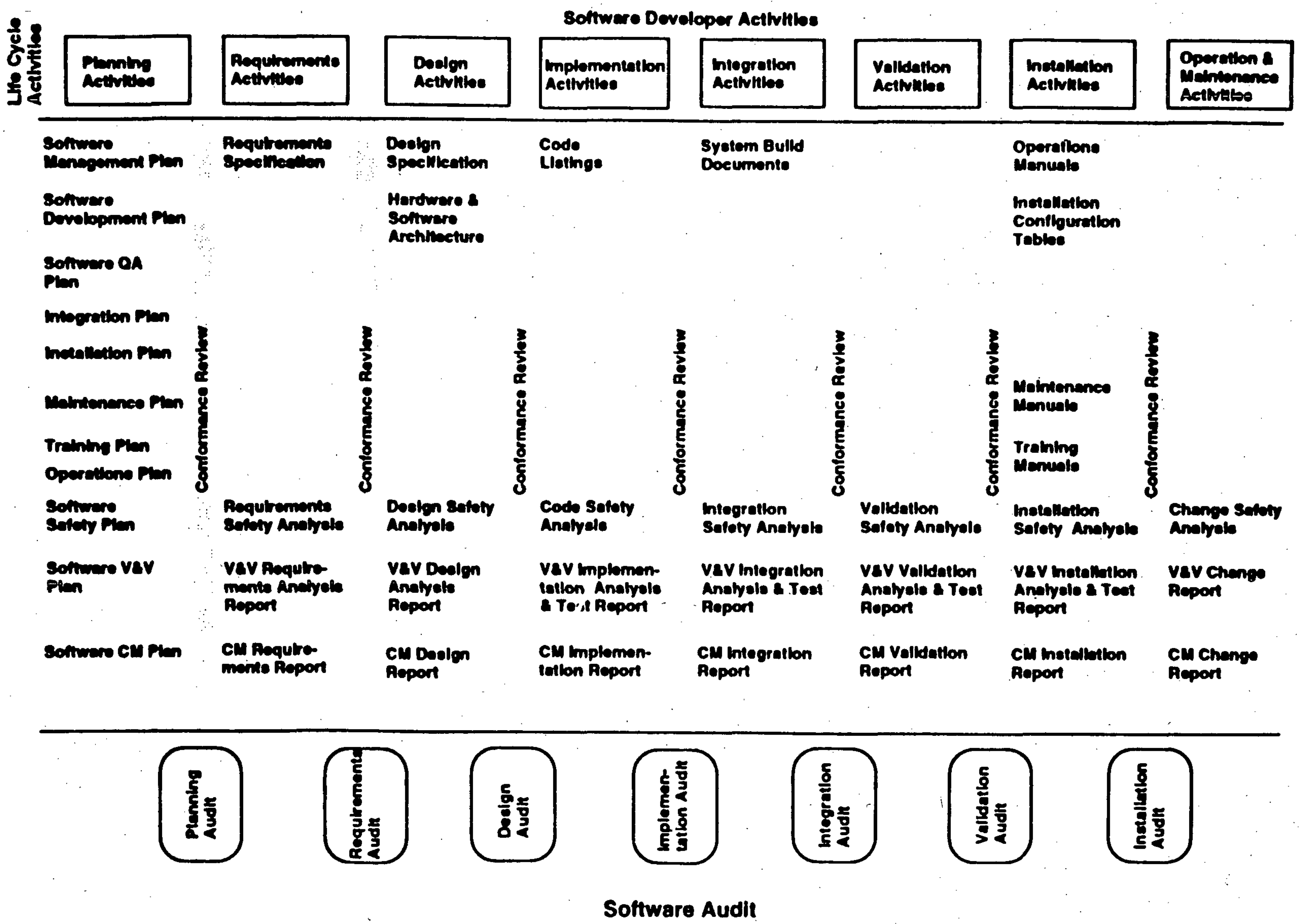


Figure 7.1-2 Flow of document through the software life cycle

The commitments described in the SSAR and related (docketed) documents provide acceptable methods and descriptions of the implementation of the CDM. The determination that the plant has been constructed in accordance with the CDM will require the use of the more detailed information contained in the SSAR and the documents which will be produced during the design phases. The SSAR commitments are based on postulations of how the design will be implemented which may change when detailed design begins. Therefore, some of the SSAR commitments may be changed. The acceptance of the ABWR design is based upon assumptions by the staff that the criteria that have been accepted will still be the proper criteria when the design is implemented. If a significant change in design process is selected by the design implementer, it may be appropriate to select different standards. Changes to SSAR commitments that affect the technical design of the I&C systems, including the design process, design implementation, and the NRC staff review as described in this report, must be submitted to the NRC for review prior to implementation. The majority of the SSAR material may be revised through a "50.59-like process" as described in SECY-92-287. This report specifically identifies those areas which must be reviewed by the staff prior to change by a licensee. The areas that the staff has determined must be submitted for review of proposed changes include computer design (hardware and software), multiplexor design, setpoint methodology, and EMC. These areas are designated with the following statement:

> Any changes to this commitment would involve an unreviewed safety question and, therefore, require NRC review and acceptance prior to implementation. Any requested changes to this commitment shall either be specifically described in the COL application or submitted for license amendment after COL issuance.

The process to be used for software development and implementation is in compliance with the specified industry standards governing those activities (Tables 7.1-1 and 7.1-2), the requirements of 10 CFR Part 50, Appendix B, QA program for I&C systems and the requirements listed in CDM 3.4. In particular, the vendor implementation and the NRC evaluation of the safety-related software QA program is accomplished under a 10 CFR Part 50, Appendix B, QA program.

Based on the additional commitments to standards in the SSAR, increased level of design detail in the design description submitted, and the use of ITAACs (including NRC inspections at various stages) to allow design detail to be developed after design certification, the DSER open issue on level of detail for digital I&C system design is resolved.

Prototyping

In discussing the level-of-detail open issue in the DSER (Open Item 1), the staff stated that it expected prototype testing of new technology to confirm expected safety performance would be required to confirm potential systems interactions, and to allow the staff to reach its safety determination on systems which may not have extensive operating experience. Based on the limited design information available for the interconnected RPS, ESF, EMS, and SSLC systems, the staff concluded that prototypes would be needed to demonstrate acceptable performance of these systems.

The term "prototype" has been used in several different ways. The staff identified several types of feasibility or demonstration testing that have at times been referred to by various parties, including the NRC, as prototype testing or prototyping. There are probably types other than the ones listed below that could be referred to as prototyping, but the staff believes that the following discussion will serve to clarify the issue of prototyping in the level-of-detail issue. The first example of prototyping, which occurs prior to design certification, is the need for a demonstration of the basic technology, or a demonstration of an example of a possible implementation of the proposed system. An example of this is the GE NUMARC digital equipment product line as a demonstration that microprocessors can be used to perform a function such as neutron monitoring which requires data transmission, comparison to setpoints, and logic decisions similar to the functions that will be required for the ABWR design. The staff considered the previous demonstrations of digital technology and does not require further demonstration of the feasibility of digital implementation of the I&C design. The staff also does not believe that there is a need to put acceptance criteria into the ITAAC for this prototype point. The design process description and the assessment of proven technology required in the equipment selection process is adequate.

The second example of prototyping occurs during the selection of the equipment vendors. There may be some prototyping to demonstrate capabilities or compatibility of specific hardware selected with the rest of the design. The staff does not believe that this demonstration is necessary to make a safety determination based on the system integration incorporated into the digital system design process.

The third prototyping example (probably at many places along the design implementation process) could include the development of specific system- aspect prototyping. For example, this may include breadboarding and testing specific components and different software languages to effectively implement the neutron monitoring algorithms. The staff does not believe that this type of prototyping needs to be described in the ITAAC because the final selected process or component will be assessed under the design process without a need to assess methods not selected for the design.

The fourth prototyping example could be identified on a typical design process as "perform hardware/software integration testing." The staff has concluded that this is an appropriate point to audit in the design process, and it will be part of the verification of the ITAAC at that stage to demonstrate that the design has been implemented correctly. This point includes a verification of the real-time performance capabilities of the system.

The items incorporated in the ITAAC include the development of the test and evaluation procedures as the design commitment, inspections to verify that the procedures are being used correctly as the inspection activity, and the completion of the testing, documentation of the results, and the feedback of the results to the design process as the acceptance criteria. The staff believes that the inclusion of this prototyping (hardware/software integration testing) in the ITAAC serves two primary purposes. The first is to verify that the design process is being implemented as certified and that the previous requirements have been met. The second is to establish the more detailed acceptance criteria for the next stage of the design process. The staff believes that this prototype needs, as a minimum, one full channel of the SSLC using hardware and software similar to the intended final product to adequately demonstrate that the design process stage has been completed.

The fifth example of prototyping (not usually referred to as prototype) is the final factory acceptance test. Factory acceptance testing will be covered by the ITAAC and has similar requirements to the previous prototype example, except that the test will be with the final software and hardware to be installed in the plant and would include all channels of the I&C system.

The last prototype example (also not usually referred to as prototype) is the final test and inspection, prior to fuel load, as installed in the plant. This test will be in the ITAAC and the SSAR Chapter 14 preoperational test program. The acceptance criteria will be a successful demonstration that all the original criteria in the design certification have been met.

GE agreed with the use of prototypes as described above and included prototyping in the ITAAC, as previously stated. The required I&C system testing is described in the ITAAC and SSAR. The COL applicant will also perform a series of tests of the I&C system following fuel load under 10 CFR Part 50 guidelines that are not included in the ITAAC. Therefore, the issue of prototyping in the level-of-detail open issue (Open Item 1) is resolved.

7.1.4 Specific Findings

In DSER Section 7.4.1.2, the staff stated that the standby liquid control system (SLCS) lacked automatic initiation capability. Subsequently, GE added automatic initiation capability to this system and revised SSAR Section 7.4.1.2 to reflect this addition. However, SSAR Section 7.1.1.4.2, SLCS, was not revised to include the automatic initiation feature. Therefore, GE's commitment to revise SSAR Section 7.1.1.4.2 was identified as DFSER Confirmatory Item 7.1.4-1. The automatic initiation feature is also included in the CDM and will be verified during the ITAAC process. GE revised SSAR Section 7.1.1.4.2 to reflect automatic initiation capability for the SLCS. GE has also included this information in the SSAR and the staff finds it to be acceptable. This item is resolved.

In response to NRC Bulletin 88-07 (June 15, 1988), "Power Oscillations in Boiling Water Reactors (BWRs)," GE committed to implement the BWR Owners Group (BWROG) resolution which includes the installation of an Oscillation Power Range Monitor (OPRM) system. This was identified as DFSER Confirmatory Item 7.2.1-2. The SSAR and CDM have been revised to reflect this change and, therefore, this item is resolved. The resolution is discussed further in Sections 7.2.1 and 7.6 of this report.

The CDM for the neutron monitoring system (NMS) (including the OPRM) had not been submitted for review in time for the DFSER and, therefore, the OPRM CDM was DFSER Open Item 7.1.4-1. GE revised the SSAR to incorporate this information; therefore, this item is resolved.

The OPRM had not been included in the draft RPS ABWR TS and, therefore, this was listed as DFSER technical specification (TS) Item 7.1.4-1. The OPRM has since been added to the TS, therefore, this TS item is resolved. The TS is described in Chapter 16 of the SSAR.

SSAR Section 7.1.2.1.6, "Protection System Inservice Testability," described an integrated self-test provision built into the SSLC microprocessors. The SSAR described this feature as "safety associated" rather than "safety-related" consistent with the SSLC. The DSER identified this as an open issue (Open Item 7). GE committed to qualify the self-test features as Class 1E (safety-related) for microprocessors performing safety functions and has revised the SSAR to reflect this commitment. This DSER open issue is, therefore, resolved.

Based on the above discussion, the staff concludes that GE has proposed an acceptable approach for design certification of the I&C systems under 10 CFR Part 52. All DSER and DFSER open items specific to the I&C systems design are resolved.

7.2 Reactor Protection System

7.2.1 General System Description

The reactor protection system (RPS) is described in Section 7.2 of the SSAR. Certain aspects of the RPS design are also discussed in SSAR Appendices 7A, 7B, and 7C. The RPS is also described in the CDM and is included in the TS. As with the other I&C systems, the RPS CDM provides the design description and ITAAC for the functional requirements of the RPS. Other CDM sections that also apply include the general I&C and EMS descriptions. The RPS includes those power sources, sensors, communication links, software/firmware, initiation circuits, logic matrices, bypasses, interlocks, racks, panels and control boards, and actuation and actuated devices that are required to initiate a reactor trip. The RPS is designed to automatically initiate the rapid insertion of the control rods of the reactivity control system to ensure that the specified acceptable fuel design limits are not exceeded. Manual initiation is also provided. The RPS also provides status information to the operator and status and control signals to other systems and annunciators. The RPS is qualified as a Class 1E safety system and will be environmentally and seismically qualified. The alternate rod insertion (ARI) capability via the fine motion rod control drive is discussed in Section 7.4 of this report.

The RPS is an open loop system with no feedback control other than operator initiation in response to display indication. The RPS performs four major functions: sense, command, execute, and display. At this highest level functional description, the ABWR RPS is similar to that previously reviewed and accepted for the GESSAR II design. The primary differences in the I&C portion of the ABWR design are: (1) the technology of the digital microprocessor-based logic and multiplexed data communications systems which is now used for most of the I&C design, and (2) the sharing of equipment for logic and display functions with other safety systems. A representation of the RPS configuration is provided in Figure 7.2-1 of this report for reference.

The RPS is implemented in the ABWR using the SSLC system which is also shared with the main steamline (MSL) isolation valves (MSIV) and the engineered safety feature (ESF) systems. ESF systems which will share the SSLC are:

(1) neutron monitoring system
(2) process radiation monitoring (PRM) system
(3) nuclear boiler system (NBS)
(4) leak detection and isolation system (LDS)
(5) residual heat removal (RHR) system
(6) reactor core isolation cooling (RCIC) system
(7) high-pressure core flooder system
(8) reactor building cooling water
(9) reactor service water (RSW)
(10) HVAC emergency cooling water
(11) diesel generator
(12) electrical power distribution system
(13) standby gas treatment system (SGTS)
(14) atmospheric control system
(15) safety-related heating, ventilation, and air conditioning (HVAC)
(16) suppression pool temperature monitoring (SPTM) system
(17) automatic depressurization subsystem
(18) high-pressure nitrogen gas supply
(19) fuel pool cooling and cleanup
(20) radioactive drain transfer system
(21) flammability control system

The following descriptions generally apply to the above ESF systems and the RPS. Differences are discussed in the ESF system Section 7.3 and other applicable sections of this report.

The SSLC is included in the I&C CDM. Figure 7.2-2 of this report is a representation of the SSLC logic and control block diagram (from the CDM) which shows the interconnection of the RPS and ESF systems. The figure also shows the inherent interrelationship with the EMS. Specific components and attributes of the SSLC are discussed in the following sections of this report.

The first primary function of the RPS is to sense the condition of certain parameters of the plant and provide accurate information to the command (comparison and calculation) section of the RPS, which in turn will initiate the execution of a reactor trip (scram) when the predetermined conditions have been met. The RPS will initiate a reactor scram when any one of the following conditions occur:

(1) neutron monitoring system conditions exceed acceptable limits
(2) high reactor pressure
(3) low reactor water level (level 3)
(4) high drywell pressure

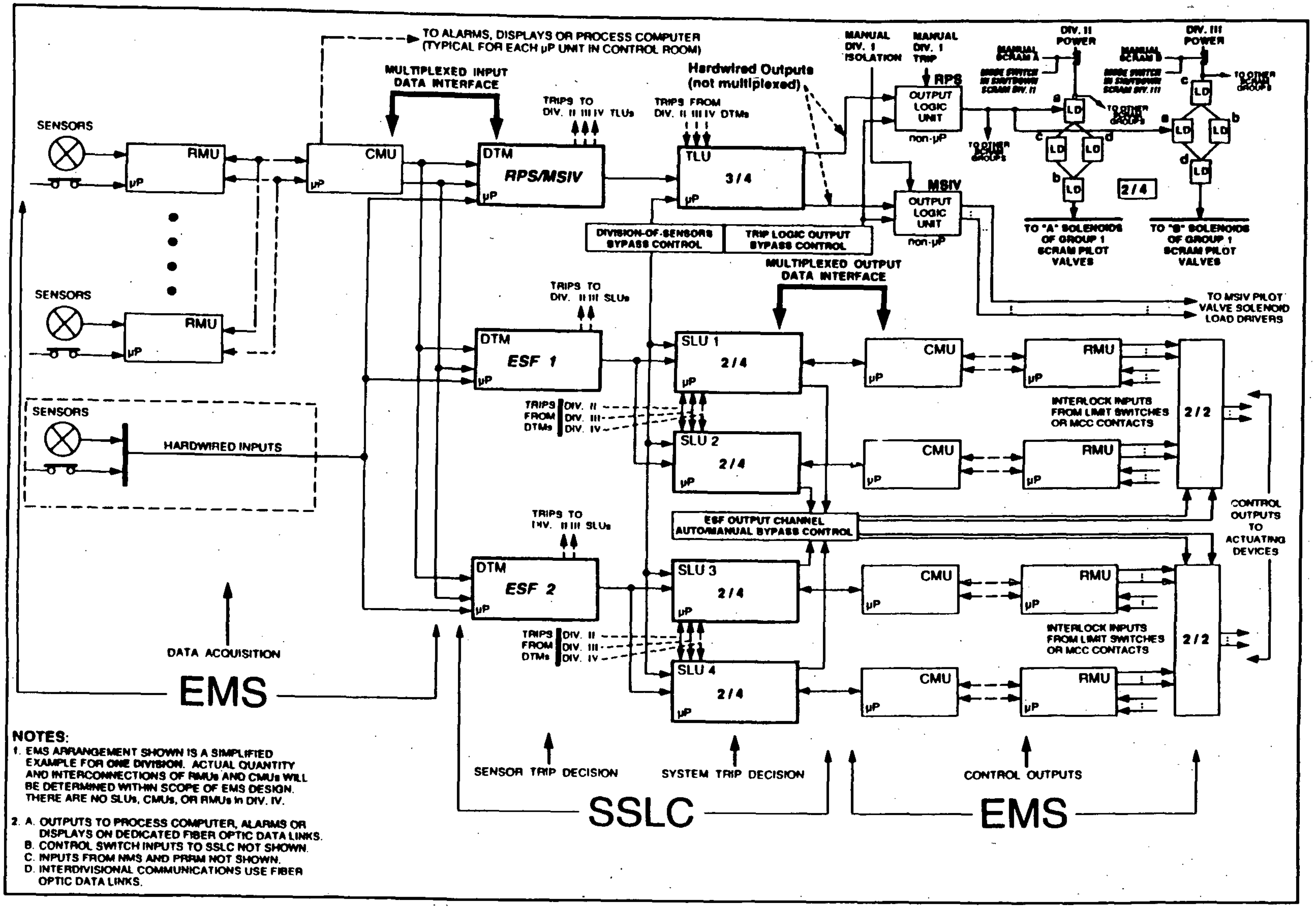


Figure 7.2-1 Safety system logic and control block diagram

(5) MSL isolation
(6) low control rod drive (CRD) charging header pressure
(7) high MSL radiation
(8) turbine stop valve closed
(9) turbine control valve fast closure
(10) operator initiated manual scram
(11) high suppression pool temperature

The specific setpoints at which a reactor trip decision is reached are discussed in the setpoint (Section 7.2.7) and TS (Section 7.11) sections of this report. The setpoint methodology is also included as a part of the I&C CDM and SSAR Appendix 7B. The specific setpoints to be used for plant operation will be determined and verified using a structured process described in the CDM after design certification but prior to reactor fuel load.

SSAR Section 7.2.1.1.4.2 lists the initiating RPS circuits. When the DFSER was prepared, the SSAR list included most of the above listed items plus a reactor scram on high seismic activity. GE has stated that this latter item was inadvertently included and would be deleted. This is acceptable to the staff and was identified as DFSER Confirmatory Item 7.2.1-1. The SSAR figures and several additional textual sections throughout SSAR Section 7.2 have been revised to reflect the removal of the seismic trip. This confirmatory item is resolved.

The high suppression pool temperature trip was added as an RPS signal after the DFSER was issued.

The NMS will provide separate, isolated, bistable startup range neutron monitor (SRNM) trip and average power range monitor (APRM) trip signals to all four divisions of RPS trip logics. The NMS provides trip signals directly to the divisional trip logic units (TLU). The NMS does not use the EMS to transmit its data. The NMS will also not use the digital trip modules (DTMs) for the determination of tripped or non-tripped status, but will perform that function within the NMS modules themselves.

The NMS provides trip signals and information from the SRNM and power range neutron monitor (PRNM). The PRNM includes both the local power range monitor (LPRM) and the APRM. The NMS monitors neutron flux from a source range of 1.E+3 nv to 125 percent of rated power. Trip signals will be provided for SRNM upscale, SRNM short period, SRNM and APRM inoperative, APRM rapid flow decrease (reactor internal pump trip), and APRM upscale (flux and thermal power).

The staff determined that the SSAR includes commitments to compliance with IEEE 279-1971 for the NMS. In addition, in response to NRC Bulletin 88-07, "Power Oscillations in Boiling Water Reactors," GE committed to implementing the BWROG resolution by incorporating an OPRM system in the ABWR design. This was DFSER Confirmatory Item 7.2.1-2. The OPRM has been added to the ABWR SSAR, and therefore, this item is resolved. The NMS (and the OPRM) is described in more detail in Section 7.6 of this report.

The second primary functions of the RPS is the command function, which is implemented using the logic of the SSLC. The RPS automatically initiates rapid insertion of the control rods to scram the reactor when warranted by any one of the predetermined conditions (listed above). The scram is initiated by means of four redundant divisions of sensor channels, trip logic, and trip actuators, and two divisions of manual scram controls and scram logic circuitry. In most instances, the EMS (described in Section 7.2.2.1 of this report) encodes the analog sensor channel output signal or contact position into a digital message, then transmits the message through an optical data link to a central decoder, which presents the signal to the DTM. In this manner, the EMS transmits data to the DTM (described in Section 7.2.2.2 of this report) from the reactor pressure transducer, reactor water level (Level 3) transducer, drywell pressure transducer, and CRD charging header pressure transducer. One exception to this EMS-to-DTM pathway is the manual scram, which bypasses both the EMS and DTM, since it is hardwired from the main control panel directly to the output logic unit (OLU). Other exceptions include the main steam isolation valve position switches, which are hardwired to the DTM, as are the MSL radiation monitors, turbine stop valve closure sensors, and turbine control valve fast closure sensor. Signals from these sensors are transmitted directly to the DTM, without passing through the EMS. However, the SSAR text and figures originally stated that the turbine inputs were multiplexed. This was DFSER Confirmatory Item 7.2.1-3. GE revised the SSAR to state which sensor inputs use the EMS and which are hardwired. The revised SSAR states that the turbine inputs are hardwired. This item is, therefore, resolved.

The DTM compares the sensor data (either received directly from the sensor or through the EMS) to a preestablished setpoint and determines if the individual sensor is in a tripped or non-tripped state. The data link between the sensors and the DTM is a fiber distributed data interface (FDDI), a bi-directional, fiber optic link in which two redundant token ring networks provide the data path protocols. Each sensor input is multiplexed through each of the two links so that the loss of a single direction of an EMS link will not prevent the SSLC from receiving the sensor information.

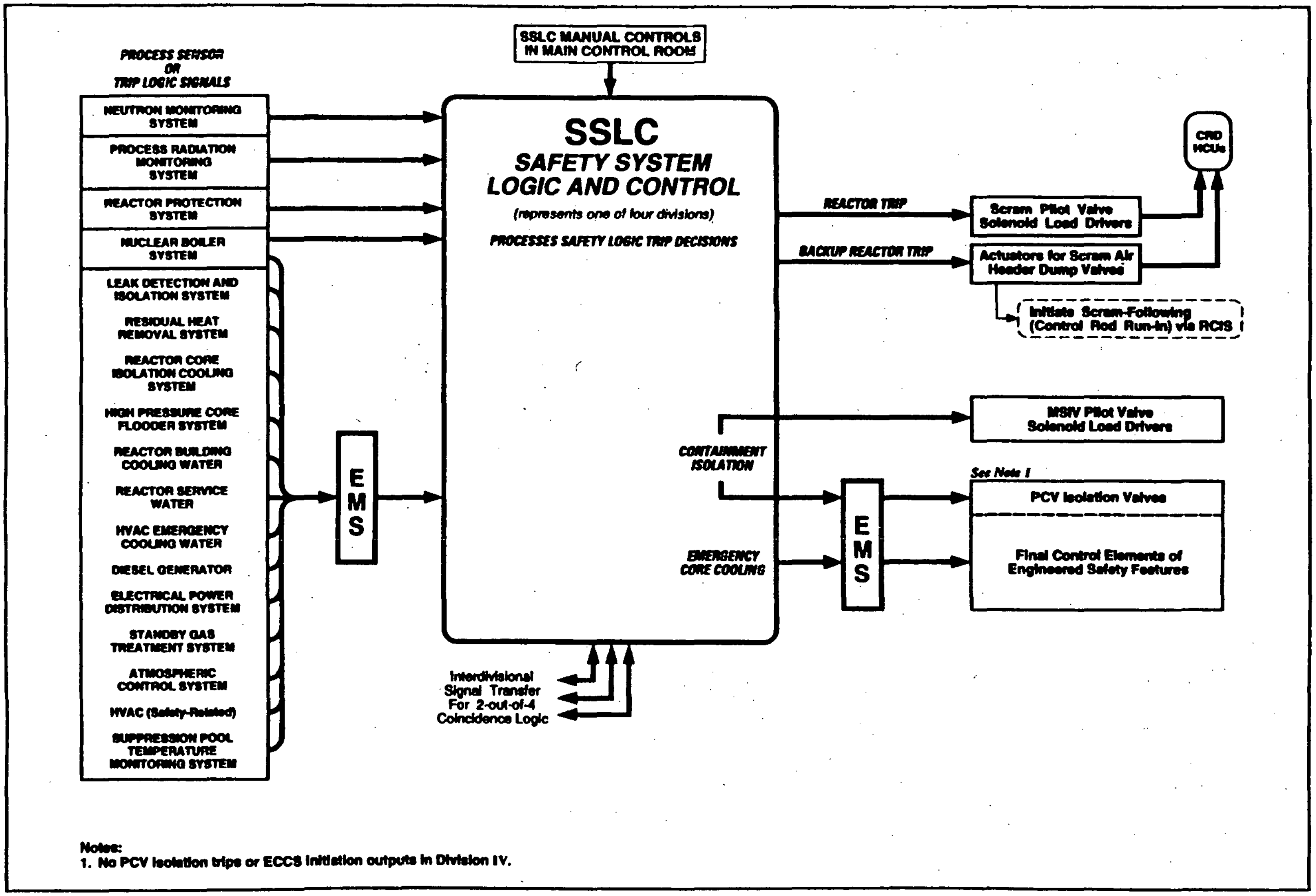


Figure 7.2-2 Safety system logic and control (SSLC) control interface diagram

The RPS is a four division system where each parameter is monitored by four sensor channels, one in each division. GE defined "sensor channel" in the TS as sensor, data acquisition, and data transmission hardware and software to the input of a bistable or voter in the TLU or safety system logic unit (SLU). A division of sensor channels is the total set of sensor channels which are input to the SSLC for the RPS, MSIV actuation and ESF. Each division of sensor channels is powered from the respective Class 1E power supply. Unless otherwise specified, a "division" of RPS (or MSIV or ESF) is the set of sensor channels, the logic channel, and output channel which are powered from the same division of electrical power. This definition of division also specifies physical and electrical separation. To ensure consistency, a "logic channel" is defined as the hardware and software that process the sensor channel inputs to produce an identifiable actuation signal within a division.

Certain sensed parameters have more than the four individual division sensors which would provide one signal per logic per channel input. The SRNM, for example, has ten channels which feed into the four RPS logic channels (three each to Divisions I and III and two each to Divisions II and IV). The MSIV position and turbine control valve fast closure each provide eight logic channel inputs.

Trip signals from the four channels of DTMs (and the NMS input) form four two-out-of-four coincidence logic matrices, one for each division of trip logic. A trip output from a DTM will also indicate the parameter which has exceeded its setpoint. If the parameter is the NMS, the indication to the operator will be that the trip was caused by the NMS without specifying the particular aspect within the NMS which caused the trip. The two-out-of-four comparison is accomplished in the TLU and the voting is done by local coincidence logic which requires the two tripped signal inputs to be from the same parameter. Because the RPS is designed to be fail-safe, the actual implementation of the logic is three-out-of-four channels not tripped. The TLU is described in Section 7.2.2.3 of this report. Trip signals from each division TLU go to the RPS OLU where a hard logic (not software/firmware) unit (similar in concept to the GESSAR II design) interconnects the divisional two-out-of-four TLU trip signals to the control rod scram pilot valves. The OLU is described in Section 7.2.2.4 of this report. Each final scram logic provides four output signals, each of which operates a load driver. A scram occurs when two or more divisions of TLUs are tripped. The two-out-of-four vote will occur twice -- once in a vote of sensor parameters in each TLU and again in a vote of RPS division output logic at the load drivers.

The RPS interfaces with the CRDs through its own pair of solenoid scram pilot valves to perform the "execute" function. There are two scram valves for each CRD. A rod scram is initiated when both (two-out-of-two) solenoids of a scram pilot valve are de-energized. When de-energized, the scram pilot valve vents the air that holds the scram valve closed. Opening of the scram valve allows the pressurized (at greater than RCS pressure) control rod water to act on the CRD piston resulting in the rapid insertion of the rods. With this arrangement, a scram of all rods is initiated if any two (or more) of the four divisional logics (TLUs) are tripped.

The OLU output is used as an input to two backup scram logic circuits. The backup scram is an energize-to-scram logic arrangement. The backup scram logic is 125 Vdc powered rather than the de-energize to scram 120 Vac power of the scram pilot valves discussed above. When the relays for the backup logic are tripped, the relay contacts will energize the air header dump valve solenoids and initiate the scram. This is a diverse means of reactor scram.

The ARI can also automatically (and manually) insert the control rods by means of the fine motion control rods or the ARI scram valves in case of off-normal conditions and is discussed in Section 7.4 of this report. The SLCS can also automatically (and manually) shut down the reactor and is discussed in Section 7.4.1.2 of this report.

For manual scram, four push-button switches are provided, one for each divisional trip logic. Additionally, hardwired manual scram is provided by two pushbuttons in the power supply circuits for the pilot scram solenoids. Actuation of any two (or more) of the four divisional switches or both hardwired manual scram pushbuttons will scram the reactor by means of the load drivers for automatic scram. The reactor can also be scrammed by setting the mode switch to the "shutdown" position. These provisions in the RPS design for manually scramming the reactor conform with the requirements of IEEE-279 and the guidelines of RG 1.62 on manual initiation of protective actions.

The display function of the RPS will be accomplished with a combination of Class 1E fixed mimic displays and Class 1E divisional visual display units (VDUs). SSAR Table 18F-1 (also 18F-2 and 3), "Inventory of Controls (displays and alarms) based upon the ABWR EPGs and PRA," provides a listing of the inventory of controls and displays required to execute all emergency procedure guideline (EPG) steps, the alarms required to alert the operator to perform the steps, and the displays to judge that the actions have been initiated or accomplished. The human factors aspects of the controls and displays, and the staff conclusions are addressed in SSAR Chapter 18 and the corresponding section of this report. There are also RPS inputs to the non-Class 1E plant computer through isolation devices.

The RPS will be periodically tested during plant operation as defined by the TS discussed in Section 7.11 of this report. In addition to the standard operator initiated surveillance, the safety system logic portion of the RPS is tested by the self-test subsystem (STS). The STS is a software-based self diagnostic system which continuously and automatically tests the SSLC components and interconnections. A more complete description of this system is discussed in Section 7.2.2.5 of this report. The provisions for testing in the design of the RPS and the periodic testing of the RPS, as described in the SSAR, conform with guidelines of RG 1.22, "Periodic Testing of Protection System Actuation Functions" dated February 1972 and IEEE Standard 338, "Criteria for the Periodic Testing of Nuclear Power Generating Station Safety Systems" as supplemented by RG 1.118, "Periodic Testing of Electric Power and Protection Systems" Revision 2. The commitments in the SSAR to the design basis requirements for the capabilities for sensor checks and test and calibration are consistent with the requirements of GDC 21, "Protection System Reliability and Testability."

Complete electrical and physical separation must be maintained between the four RPS divisions in order to meet the criteria of IEEE Standard 279-1971. This requirement is included in the CDM and will be verified during the implementation of the ITAAC. The staff concluded in the DSER that isolation of information (error handling) must also be addressed because of the extensive use of multiplexors and software. This aspect of the multiplexors is discussed in Section 7.2.2.1 of this report.

A trip of any sensor or logic channel will be annunciated and will cause that channel to lock in the trip mode until manually reset. The RPS will be fail-safe in that a loss of power to a channel will result in that channel going to the tripped condition. Other failures such as a break in a communications link will be detected and the self diagnostic of the individual microprocessor will put the output to the tripped state.

The design of the ABWR RPS is significantly different than in previous BWRs in the method the high level functional design is implemented. The staff reviewed the SSAR and found that conceptual design description to be a functional block diagram level of detail. The staff found that the hardware design documents for the SSLC did not state details such as bus protocol, bus data capacity, provisions for hardware level interrupts, the size of the memory, the speed and size of the microprocessor, the format of the status panel, hardware based interlocks, and type of display media. The means of addressing the above design details for design certification is described in Section 7.1 of this report and is discussed further below.

The functional level block diagram of the safety-related signal paths is simple and direct. However, in a microprocessor-based system, the software implied in the blocks of the system diagram can mask much of the safety system's design complexity. Several of the significant issues involving the use of microprocessor-based systems and safety functions have been presented to the Commission in SECY-91-292 (September 16, 1991), "Digital Computer Systems for Advanced Light Water Reactors."

The staff concluded in the DSER that the SSLC design was not "essentially complete" as required by 10 CFR Part 52 because of the complexity of the ABWR I&C system design and the lack of design details for certification. The staff found that the design description presented for the SSLC was ambiguous because the design material was dispersed in the submittal and often contradictory. For example, data about the TLU trip (RPS division trip) status was not originally listed in the SSAR as an output to the operator.

The staff reviewed the SSLC system design specification, which was described by GE as a procurement level document. The staff considered the documentation for the SSLC to be inadequate for design evaluation and not in conformance with the design certification requirements for level of detail. The staff, therefore, concluded that the software design and development aspects of the SSLC were not described to a sufficient level of detail. GE did not adequately disclose the details of the design to enable the staff to reach a final conclusion on the acceptability of the design for certification, and this was an open issue in the DSER.

Since the issuance of the DSER, GE has provided additional detail, committed to industry standards appropriate for the digital system design, and submitted the CDM which includes the I&C system design process. The staff concludes that the combination of the improved level of detail provided, the commitments to standards, and the incorporation of the design process into ITAAC with extensive NRC auditing during design development and implementation provides the basis for a final safety determination for certification, and is sufficient to resolve this issue. In Section 7.1, the staff discusses the issue of level of design detail. This open issue from the DSER is, therefore, resolved.

The staff also reviewed the SSAR for conformance to IEEE Standard 279-1971 and ancillary requirements for meeting the single-failure criterion, independence, control and protection system interaction (isolation), testing, bypass and bypass indication (including removal of bypass), and manual initiation. To meet the single-failure criterion described in Section 4.2 of IEEE-279 and IEEE-379, "Standard Application of the Single-Failure Criterion to Nuclear Power Generating Station Class 1E Systems," GE committed in the SSAR to compliance with RG 1.53, "Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems." The SSAR states that the RPS will comply with GDC 22 on protection system independence, the requirements in IEEE-279 on channel independence, and the guidance for physical and electrical independence of the instrumentation system in RG 1.75, which endorses IEEE Standard 384. Conformance to RG 1.47, "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems," is described in the SSAR and discussed further in Section 7.2.3 of this report. The RPS design includes provisions to meet the requirements of IEEE-279 on channel and operating bypasses. The RPS performs no control functions, interlocks with control systems are to be through isolation devices, and the channels of the RPS are to be electrically isolated and physically separated in order to meet the criteria of GDC 24 on separation of protection and control systems and the requirements of IEEE-279 for control and protection system interaction. The RPS description and drawings in the SSAR describe a clear commitment to the above requirements. The CDM describes the GE commitments to the most significant requirements from the various standards and criteria (though not the specific standards themselves). These commitments will be verified during the ITAAC implementation. The standards themselves and other less significant criteria referenced in the SSAR and this report are expected to change over time and are not "locked in" by the CDM.

In the DSER, the staff noted that the SSAR did not present a detailed failure modes and effects analysis (FMEA) to ensure that all postulated failures result in a known safe state if the RPS experiences conditions such as disconnection of the system, loss of power, or exposure to a postulated adverse environment. Therefore, the staff could not evaluate conformance to GDC 23, "Protection System Failure Modes." This was Open Issue 3 in the DSER. Open Issue 3 has since been resolved by the FMEA submitted by GE and included in the SSAR, and additional studies performed by the staff. As a result of these studies, the staff determined that some aspects of this issue relate to the potential for common-mode failure of the I&C system. Consequently, the DSER issue concerning the lack of a FMEA was resolved, but the potential common-mode failure and the required redundancy and diversity issues were identified as DFSER Open Items 7.2.6-1 and 7.2.6-2. These open items are resolved in Section 7.2.6 of this report. The staff concludes that the RPS design meets the requirements of GDC 23, "Protection System Failure Modes."

When the a staff issued the DFSER, the figures (interface electrical diagrams (IEDs)) and interface block diagrams in SSAR Section 7.2 lacked many details to be submitted later. This was DFSER Confirmatory Item 7.2.1-4. The figures have been corrected in the SSAR and this item is, therefore, resolved.

7.2.2 Safety System Logic and Control and Specific Subsystem Descriptions

The SRP and the SSAR format are established along functional system boundaries. The ABWR I&C system is significantly different from the I&C system designs in plants when the SRP was last revised in 1984 in that the safety functions are combined the SSLC, into a common computer-based logic, control, display system. Consequently, the ABWR I&C system does not follow the discrete system boundaries assumed in the SRP. The SSLC serves both the RPS/MSIV and ESF systems. The following sections describe common elements of the SSLC which are used for the RPS. Although GE does not consider the EMS to be part of the SSLC, it is inherently interconnected with the SSLC and, therefore, is included in this section of this report.

The staff reviewed some of the I&C system documents referred to as master parts list (MPL) documents. These documents are referenced as supporting documents in the IED figures in the SSAR. The MPL documents reviewed and discussed throughout this report were prepared for the Tokyo Electric Power Company, Inc., Kashiwazaki Kariwa Nuclear Power Generation Station, Units Nos. 6 and 7. These documents list applicable Japanese laws, regulations and standards. In addition, the Japanese standards list the U.S. standards which are either specifically accepted in the MPL documents or are referenced as guidelines. For example, MPL C71-4010 (Rev. 1, July 2, 1990), "Reactor Protection System Design Specification," states acceptance of 10 CFR Part 50, Appendix A (GDC) but refers to IEEE 279, IEEE 603, and RG 1.152 as standards that form a part of the document or are listed as references without a specific statement of acceptance or conformance. Several items in the MPLs are specific to the Japanese design and, therefore, will require revisions to refer to U.S. criteria if these documents are used in the ABWR design. For example, the Japanese plant RPS is specified to use a 50 HZ, 120 Vac power supply instead of a 60 HZ supply as specified in the ABWR SSAR. The MPL document also includes the seismic scram function which is not included in the ABWR design. The staff has not listed all such discrepancies in this report. The verification process in the ITAAC will be used to confirm that the MPL design documents, when developed and used by the COL applicant, will be in accordance with the SSAR and CDM. Where necessary, the staff has requested GE to clarify the standards commitments in the SSAR for the ABWR so reliance in the SSAR is sufficient for the safety finding. Standards commitments are discussed throughout this report.

MPL ABBE-4080 (Rev. 0, April 25, 1990), "EMS/SSLC Interface Requirements, Requirements Specification," states that system timing shall be a synchronous between the EMS and the SSLC. The interface between any one channel of the EMS and the corresponding channel of the SSLC may be either by hardwired or by fiber optic cable. The SSLC will determine from which of the dual EMS channels (per division) to select the information. The selection will be based upon data quality checks. The data quality checks may include checksums, parity checks, rate limiting checks, and the de-bouncing of digital inputs. The DTM will perform these functions except for the display functions which are taken directly to the displays from the control room multiplexing unit (CMU). Typical data formats are presented with options from which to select, dependent on the final design input and output parameters. The above information is consistent with the SSAR description.

The staff reviewed MPL A32-4080 (Rev. 1, April 24, 1990), "Safety System Logic and Control Design Specification." The document is in general conformance with the SSAR and CDM but does not add a significant amount of design information about the SSLC. Many of the design procedures are described as items to be developed later. This document lists topics to be considered when the design is developed and is similar to the SSLC CDM. The MPL document is, therefore, appropriate for the intended purpose.

7.2.2.1 Essential Multiplexing System

As described in SSAR Chapter 7.2, the EMS portion of the RPS system transmits data to the SSLC from the sensors that are not hardwired or part of the NMS. SSAR Appendix 7A discusses the EMS system. Additional information is included in SSAR Appendix 19N, "Analysis of Common-Cause Failure of Multiplexing Equipment," SSAR Chapter 7 Appendix C, and the MPL documents discussed below.

Data multiplexing is an integral part of the ABWR I&C system design. Multiplexing systems will transfer data between sensors and actuation control devices distributed throughout the plant, and the logical processing units in the control room. The multiplexing system applications procedure (MPL Document No. A11-4120) and the SSAR define two multiplexor systems. The multiplexing system tasks are divided between the EMS and the Non-Essential Multiplexing System (NEMS).

The EMS receives inputs only from safety systems. Any outputs to non-safety systems are electrically isolated. The EMS is not listed as a separate safety system in GE Table 7.1-2 of the SSAR, but is considered part of each safety system for which it provides data communications. The EMS interfaces with the RPS/MSIV and ESF systems, and is qualified to the same quality standards as the RPS, ESF, and SSLC systems. The EMS is powered from the Class 1E 125 Vdc buses. The EMS transmits approximately 1500 signals either from the sensors to the SSLC or from the SSLC to the actuated equipment (for the ESF functions only). The EMS has a separate CDM description in addition to those aspects of the EMS included in the functional requirements of the RPS CDM and functional and design process requirements of the SSLC (I&C) CDM.

The SSAR states that the NEMS is used in the non-safety-related control systems and is not considered a safety system. The NEMS is discussed further in Section 7.7 of this report.

The EMS is comprised of four independent divisional multiplexing systems, each of which has redundant data links within the channel. In the DSER, the staff concluded that GE had not committed to appropriate industry standards for certain aspects of the EMS design, including the multiplexors themselves. GE has since committed to a deterministic, dual redundant, fiber optic ring structure which will follow the guidelines of industry accepted Standard American National Standards Institute (ANSI) ASC X3T9.5 (1988), "Fiber Distributed Data Interface (FDDI)." As noted in previous GE responses to staff requests for additional information, the design is not specifically constrained to this particular format and configuration. Specifically, the CDM does not require a specific design at this time. Rather, the CDM provides functional requirements with the SSAR providing more specific information at the functional level of detail. The SSAR also refers to IEEE 802.5 (1985), "Token Ring Access Method and Physical Layer Specifications," if a lower throughput of information is incorporated in the EMS design. Both options are consistent with International Standard Organization (ISO) 7498 (1984), "Open Systems Interconnection - Basic Reference Model," as the data link layer and physical layer. IEEE 802.2 (1985), "Standard for Local Area Networks: Logical Link Control," will define the protocols necessary to move data to the higher levels of the ISO model. Any changes to these standards commitments would involve an unreviewed safety question and, therefore, require NRC review and acceptance prior to implementation. Any requested changes to this commitment shall either be specifically described in the COL application or submitted for license amendment after COL issuance. For example, if a new ANSI FDDI standard is developed which uses a shorter wavelength than the current standard, the COL applicant or holder will need to verify to the staff that the dispersion will still be small enough to meet the FDDI's bit error rate.

The data link and physical layers are part of the general seven-layer hierarchical model of the open systems interconnection which the FDDI protocol conforms to. The seven layers are application, presentation, session, transport, network, datalink (including logical link control and media-access control), and physical (including physical and physical medium dependent). The FDDI standard (ANSI ASC X3T9.5) also requires a station management function at each station to supervise sub-layer and ring management operations. The descriptions that GE has presented, with the caveat that this is but one of a number of possible implementations, are in general conformance to the above standard to the level of detail provided. These standards are supported by the industry.

Vendors currently offer chip sets to implement the individual FDDI sublayers, as well as complete systems that conform with the above standards. FDDI controllers are available with several different computer buses. The staff concurs with GE that no additional prototyping is required to demonstrate the feasibility of the design because the EMS will use proven technology. The capacity of the FDDI should be capable of handling the quantity of inputs/outputs at the rates necessary for proper operation of the SSLC. This will be verified by testing during the ITAAC implementation. The prototype requirements described in Section 7.1 of this report still apply.

The staff concludes that the ABWR SSAR references and contains commitments to appropriate standards for the EMS design at this level of design detail and, therefore, this open issue (Open Issue 5 regarding adequate commitments to industry standards) from the DSER is resolved. The implementation of the design in accordance with the standards referenced above and other ABWR SSAR statements will be verified during the ITAAC. The multiplexor design will also be verified through a staged ITAAC as discussed in Section 7.1.3.3 of this report. When the final design development is initiated, the COL applicant will submit the design plans for staff review and for development of the ITAAC verification audit points.

As a data highway, the EMS multiplexes data from the sensors to the control room logic units, and multiplexes the commands from the control room computers to the appropriate actuation control devices (for the ESF actuation only). The RPS (and the MSIV initiation) does not use the EMS for the output of the SSLC to the scram pilot valve initiation. With the exception of hardwired input connections (turbine stop valve closed, turbine control valve fast closure, MSIV position, manual scram, and the neutron monitor systems input) all of the RPS (and most of the ESF) sensor inputs will be processed by the EMS.

The hardware architecture of the EMS described in the SSAR uses fiber optics for the communications medium and microprocessors for the node controllers. The SSAR also states that the ABWR EMS may use coaxial cable or twisted-pair connections. If copper wire is used in place of the fiber-optic cable, electrical isolation will be maintained and provided where necessary (and demonstrated by testing of the devices in accordance with the Licensing Review Bases (LBB), Appendix B) thus resulting in potentially additional isolation devices. Fiber-optic cable is an inherently excellent electrical isolator, and its use to the extent practical will reduce the need for additional isolation devices.

The major components of the EMS are the remote multiplexing units (RMU), the CMU and the fiber-optic (or copper) cables. The RMU and CMU will contain the following items: transmission line interface circuits, processors, memories, signal conditioning circuitry, analog to digital conversion, and watchdog timers. The specific number of RMUs has not been determined. The only specific EMS requirement identified at this time is that within each division of EMS, the wide-range and narrow-range reactor water level sensors will be processed by different RMUs. This will provide an additional level of defense-in-depth such that any single RMU failure will only disable one of the water level indications and SSLC inputs. All of the components of the EMS are identified as Class 1E. At the RMU, an input sensor interfacing is provided to switch contacts, 4-20 ma current loops, thermocouples, RTD devices, pulse inputs, and voltage inputs. The RMU output interface (which is not used for RPS functions) is to relays, solenoids, voltage to current converters, panel meters, and indicator lights. All equipment will be rack mounted and operated from the divisional 125 VDC Class 1E power sources.

The RMU will be located in mild environment areas throughout the plant near the sensors and the actuation control devices with which they interface. The RMU and CMU are connected to the fiber-optic cable, and together control transmission to assign which RMU/CMU transmits or receives signals. The RMU multiplexes and sends data messages to the control room via the fiber-optic cable. In the control room, the CMU demultiplexes the message and sends it to the appropriate DTM of the SSLC. The DTM will select which of the redundant CMU signals it will use. Control signals from the SLU for the ESF functions of the SSLC are sent to the CMU which in turn transmits these control signals to the appropriate RMU.

The EMS design concept identifies the RMU and CMU as sharing a modular, microprocessor-based, bus-oriented architecture, using similar modules. The RMU and CMU are configured somewhat differently with the CMU having additional communications modules instead of input and output modules. The following functional modules will be in a typical RMU/CMU:

a. Input - acquires analog and digital data from the sensors.

b. Output - transmits control signals to equipment actuation control devices (ESF function only).

c. CPU - processes the signal data, coordinates I/O and communications, and performs calibration and diagnostics.

d. Memory - contains the stored program executed by the microprocessor and stores intermediate data.

e. Communications - formats and multiplexes data that is sent through the serial optical link.

f. Front Panel Interface - permits technician access to calibration and diagnostic functions.

GE stated in SSAR Chapter 20, that the RMU failure mode was dependent on what sensors were connected to it, either fail-safe (negative, low) or fail-as-is (last reading). The particular failure state for the RPS will be fail-safe and will be verified during the ITAAC implementation.

GE stated in SSAR Chapter 20, that the EMS includes self-test software that detects malfunctions in the hardware modules and provides alarms in the control room. GE has not presented specific information on the types of malfunctions for which testing is being provided for or the EMS reaction to each malfunction (e.g., restart, alarm, fail-safe, or fail-as-is). However, GE has provided a listing of typical parameters that are monitored. These include the status of the central processing unit, parity checks, data plausibility checks, watchdog timer checks, memory checks, voltage level checks, and data range and boundary checks. The self tests also provide the capability for internal testing and error checking within the EMS. In addition, the EMS is subject to routine periodic operator initiated surveillance checks required by the TS. Also, as described in the simplified FMEA in Chapter 15.B.4 of the SSAR, the EMS will reject corrupted signals in the transmission. Further, the DTM, which receives inputs from the EMS, uses a data quality check to determine which of the CMUs it will accept data from.

The selection of the EMS hardware was not identified in the SSAR or supporting material. However, the top level specification documents referenced the design control and hardware reliability standards and criteria required to be met. In response to staff questions (Q) (Q420.92, Chapter 20 of the SSAR), GE committed to MIL-HDBK-251 and to MIL-STD-217E for EMS hardware reliability and thermal effects.

The staff identified an open issue in the DSER based on a concern that common- mode failure from effects such as electromagnetic interference (EMI/EMC) or design error could cause the loss of the EMS in more than one division. In response, GE performed an assessment of the effect on the ABWR I&C systems for a loss of all four divisions of the EMS. The GE study (SSAR Section 19n) concluded that under such a condition, the plant could be safely shut down from the remote shutdown system. This response was insufficient to fully resolve the common-mode failure concern, and further information was requested from GE. The concern about EMI/EMC, reliability and thermal effects was eventually resolved as discussed in Section 7.2.8 of this report. The concern with potential common-mode failure of the EMS (and other equipment) was also eventually resolved as addressed in Section 7.2.6 of this report.

The SSAR defines the top level hardware architecture for the EMS but presents limited information for the software that runs in the EMS microprocessors. GE stated the following high level design goals for the EMS software:

a. real time executive kernel
b. hierarchical task structure
c. simple modules
d. on-line calibration with bypass
e. self-test as background process
f. automatic recovery

The ABWR EMS uses microprocessors to control the movement of data from the sensors to the SSLC, between the different components in the EMS itself, and from the SSLC logic to the actuated devices (for the ESF systems). The high level block diagrams of the data signal paths are simple and direct. However, in a microprocessor-based system, the software implied in the blocks of the system diagram can mask much of the safety system's design complexity. In a microprocessor-based data transport system, the software is an essential line element in the execution of the safety system functions. For this reason, the staff concluded in the DSER that the design of the EMS did not provide the appropriate level of detail essential to the staff's review. This was identified as Open Issue 1 in the DSER. Since the DSER, GE has committed to the use of several additional software development standards as stated in the SSAR. This commitment, in conjunction with the design process, corresponding ITAAC steps, and staff audit verification, resolved this issue. The topic of hardware and software qualification is addressed further in Section 7.2.8 of this report.

The SSAR initially did not present an architecture for the software design. GE had not demonstrated how the decision logic, a parallel process in an analog system, would be implemented by the software, which is usually a serial process. The staff considered this an open issue in the DSER related to level of detail. GE has responded to this issue by providing the timing requirements for each stage of the control process from sensor to scram initiation, and demonstrating that the scram time requirements (50 msec) can be met with the EMS and the SSLC. The decision logic will be accomplished using a deterministic serial process. The timing requirements will also be demonstrated in the ITAAC implementation. This is acceptable to resolve the issue of serial vs. parallel processing.

The following is a list of software elements that process a signal as it moves from the sensor through the EMS to the SSLC control logic for the RPS.

a. Remote multiplexer unit - input

    1. operating system/executive
    2. signal acquisition
    3. signal conditioning
    4. analog to digital conversion
    5. digitized signal data preparation for transmission
    6. digitized data transmission on fiber optic link

b. Control room multiplexer unit - input

    1. operating system/executive
    2. digitized data acquisition from fiber optic link
    3. data preparation for SSLC/DTM
    4. data transmission to SSLC/DTM
    5. multiplexed data transfer control
    6. display information transmission

c. DTM processing by SSLC software modules.

A basic design parameter of a local area network is whether synchronous or asynchronous transmission is used. The staff identified concerns about this issue in the DSER, and in particular was concerned about the use of synchronization across channels which could violate the single-failure requirements of IEEE-279 and IEEE-603. GE responded by stating in the SSAR and CDM that the EMS will run asynchronously. There will be no common clock shared between channels. The inherent time skew in the logic processing that will result by not synchronizing the channels will be controlled by maintaining a significantly high sample rate and storing the last four validated variables in each TLU for each vote. There is also no common clock within a division, however, the RMUs will append timing information to the sensor information to be multiplexed to the CMUs. The staff finds this acceptable, and the DSER issue of synchronization is, therefore, resolved.

Although the detailed design of the EMS depends on the hardware that is selected, the functional requirements for the EMS as part of the ABWR safety systems are not hardware dependent. Because the detailed EMS hardware design was not provided, a concern regarding level of detail was identified as part of Open Item 1 in the DSER. In addition, Open Item 2 identified the need for design information about the isolation of corrupted data in multiplexors. In response to these issues, GE has defined the high level functional requirements of the EMS, the major parameters that define the data transmission attributes, and the criteria for selecting the data transmission hardware.

GE stated several functional requirements for the EMS as follows:

1. The EMS is to be implemented such that any single failure or any single channel removal within the system shall not prevent proper action of the RPS at the system level.

2. The EMS in each division will be independent, and physically and electrically separated from the other divisions to preclude failures in one EMS from propagating to another division.

3. Non-safety system failures are to be isolated from the safety systems.

4. The capability for testing during power operations is to be provided.

5. The EMS will satisfy all the environmental operability requirements (seismic, temperature, humidity, radiation level).

6. The EMS will have adequate capability to withstand surges in accordance with IEEE-472.

7. The EMS will have a designated temperature range of 10 to 40 °C. (50 to 104 °F), for relative humidity of 10 to 60 percent for the CMU and of 10 to 90 percent for the RMU.

The staff concludes that the combination of additional information on EMS functional requirements, commitment to appropriate additional standards and the ITAAC process provide the necessary information on the EMS design and, therefore, the related part of DSER (SECY-91-294) Open Items 1 and 2 are resolved.

The two loops of the network are designated "master" and "standby" by the receiving fiber optic interface. The designation of which loop is "master" is on the basis of transmission errors, checksum errors, and other tests in the self diagnostics. The diagrams that the staff has reviewed showed that each DTM in the SSLC has two fiber optic interfaces. The "master" designation is also applicable at the RMU level where ESF equipment actuation commands are received. It was not initially clear to the staff how these two fiber optic units arbitrate between themselves to determine which is to be the "master" loop. GE stated that the decision will be based upon a data validity check at the DTM. The specifics of what will be valid data for each parameter was not provided. This was Open Item 6 in the DSER. GE has committed to the FDDI standard for data validation guidance which the staff finds acceptable to resolve this issue. The correct implementation and demonstration by test of the master/slave switch will be included in the ITAAC for multiplexors. DSER Open Item 6 is, therefore, resolved.

MPL DMH-4270 (Rev. 2, February 3, 1989), "Essential Multiplexing System Design Specification," provides various EMS design details. It states, for example, that the bit error rate shall be under 10E-9 "theoretically." This is consistent with the FDDI guidelines. The CDM commitment is to a functional demonstration of capability without specifying particular throughput or error rates. The MPL document also states that the RMU and CMU shall have the capability to transmit the data at the rate of one megabit/sec (Mbps). The FDDI standard is written for essentially an 100 Mbps system. The difference in data transmission rate between the GE MPL and the FDDI standard is an example of an MPL revision that will be required and verified during the ITAAC implementation in order to ensure consistency with the CDM. The EMS will be capable of sampling a sensor with a 10-msec sampling rate, but the actual sampling rate for each sensor has not yet been determined. The sampling rate will be established as part of the detailed design by the COL applicant. The maximum transmission distance for any single multiplexing station is specified in the MPL as 500 m (approx. 1640 ft). A maximum distance verification was to be included in the ITAAC. This was DFSER Confirmatory Item 7.2.2.1-1. Although the CDM has not been revised to specifically reflect the distance verification, the functional requirements in the CDM and an SSAR change restriction are acceptable to the staff and this item is, therefore, resolved.

The need to incorporate consideration of instrument channel inaccuracies due to A/D converter in the setpoint methodology was DFSER Confirmatory Item 7.2.2.1-2. GE revised CDM Section 3.4, "Instrumentation and Controls" to include setpoint methodology and specifically includes consideration of the A/D converter accuracy. This item is, therefore, resolved.

MPL A32-4080 (Rev. 0, April 25, 1990), "EMS/SSLC Interface Requirements, Requirements Specification," specifies that the four channels of the EMS be asynchronous. This is consistent with the SSAR and the CDM. It also requires a fixed format for both the messages and the sequence of the messages. This is consistent with the general GE commitment to use a deterministic EMS. The MPL notes that the RMUs shall provide appropriate filtering without providing a specification of what is appropriate. Providing appropriate filters will be part of the design development and implementation process which is the responsibility of the COL applicant.

The EMS design of the ABWR uses microprocessors and related software throughout the safety systems. While promising improvements in performance and reliability, this advanced technology also can introduce problems and failure modes that had not been addressed before in reactor safety system design. Several of the communications control functions performed within the EMS are implemented by software. RG 1.152, which endorses ANSI/IEEE ANS Standard 7-4.3.2 (1982), has been promulgated as the standard for developing software for safety systems. However, GE's commitment to this standard was ambiguous. There was no evidence in the SSAR that the software development for the EMS would be in accordance with IEEE Standard 7-4.3.2. This was an open issue in the DSER (SECY-91-294). GE subsequently removed the ambiguity from the SSAR commitment to IEEE Std. 7-4.3.2 and added additional software requirements to the SSAR which are discussed in Section 7.2.8 of this report. This issue is, therefore, resolved.

In the DFSER, the staff stated that the sensor data transmitted through the EMS should be clarified in Chapter 7.2 of the SSAR. For example, the turbine stop valve closure input was described in Section 7.2.1.1.4.2 (6) as using the EMS, which was not consistent with other information provided by GE. This was DFSER Confirmatory Item 7.2.2.1-3. GE has revised the SSAR to specifically state which types of data are transmitted through the EMS and which types one not. This item is, therefore, resolved.

GE has committed in the SSAR that the EMS design will meet the relevant SRP criteria and will meet current industry standards for multiplexing. GE also committed to additional hardware and software criteria beyond those contained in the SRP. Based on the above, the staff concludes that the EMS will be capable of accomplishing the RPS performance requirements. Verification of the final design against the above criteria will be performed during the ITAAC implementation. Any changes to the commitments described above concerning the performance specifications and architecture of the EMS would involve an unreviewed safety question and, therefore, require NRC review and acceptance prior to implementation. Any requested changes to commitments involving the EMS performance specification or architecture shall either be specifically described in the COL application or submitted as a license amendment after COL issuance.

7.2.2.2 Digital Trip Module

The digital trip module (DTM) is a microprocessor-based unit which performs the basic comparison of the sensors to the preestablished setpoint, and provides a trip or no-trip output for that parameter. For the RPS, there is one DTM for each set of sensor channels (a division of sensors, refer to Figures 7.2-1 and 7.2-2 in this report). The DTMs for the ESF functions are separate and are not shared as was the case for the EMS. The DTM, as with all of the SSLC, is Class 1E. The hardware and software requirements for the DTM are addressed in Section 7.2.8 of this report. The RPS DTM is designed to be fail-safe. A loss of power to the DTM will result in a trip signal to each of the TLUs until that division is placed in bypass by the operator.

The DTM will receive input from both the EMS and directly hardwired sources. The hardwired sources include both the traditional hardwired inputs (discussed previously in this section) and the process PRM which provides the MSL high radiation input. The only RPS scram signals not processed through the DTM are from the manual operator controls and the NMS.

The DTM will send a trip or no-trip signal for each of the sensed parameters to each of the four TLUs. The signal to the TLUs which are not in the same electrical division will be electrically isolated and separated. The DTM will also provide data to the Class 1E divisional display units, the Class 1E mimic board, and isolated outputs to the plant computer and non-Class 1E alarms and annunciators.

GE described the following elements of the DTM:

1. Operating system/executive
2. Setpoint check
3. Trip decision transmission if setpoint exceeded
4. Serial data transmission to all divisions

For the items that are multiplexed, both multiplexor loops within a division and the identical sensor information contained on both is available to the DTM. As described in the previous section on the EMS design, the DTM will perform a data validity check and select from which CMU it will receive information.

As part of the data checking, the DTM will also verify that it is receiving the correct parameter and will ignore any other signals, such as those for the ESF DTMs. The DTM will also perform a data validity check on the non-EMS inputs and provide alarms if the checks fail.

Figure 7.A.2-1 in SSAR Appendix 7A shows all the sensor signals sent via the EMS. This drawing was listed in the DFSER as needing revision to be consistent with the distribution of the sensor inputs to the DTM as described above. This was DFSER Confirmatory Item 7.2.2.2.2-1. The SSAR has been revised to properly show the DTM sensor inputs, and this item is, therefore, resolved.

MPL A32-4080 (Rev. 0, April 25, 1990), "EMS/SSLC Interface Requirements, Requirements Specification," requires that the DTM permit manual trip of individual plant data inputs when an inoperable condition is detected in the incoming EMS data. This is consistent with the SRP criteria for a manual trip.

MPL C71-4010 (Rev. 0, May 18, 1990), "Reactor Protection System, Hardware/Software System Specification," lists the 11 sensor inputs and 34 outputs to the TLUs (if seismic inputs/outputs are not used) for each DTM. In addition to the sensor tripped/not-tripped status, the DTM output will include the mode switch MSL isolation trip bypass permissive to the TLU in its own channel. The DTMs are essentially identical between the four RPS divisions. There will be sensor identification and small timing differences that provide for some small differences, however, most of the device's hardware and software is identical between divisions. The DTMs are also similar to the DTMs of the ESF functions. The software algorithms will be significantly different between the RPS/MSIV and ESF functions but the operating system, the self diagnostics, and the input/output will be identical. The staff was concerned that the DTMs could be vulnerable to a potential common-mode failure of all divisions and between the RPS/MSIV and ESF due to a hardware or software design error. This was an open issue in the DFSER and the resolution is discussed in Section 7.2.8 of this report.

7.2.2.3 Trip Logic Unit

The trip logic unit (TLU) is a microprocessor-based unit which performs a two-out-of-four coincidence logic calculation based upon the signals it receives from the four DTMs and the four divisions of NMS. There is one TLU for each of the RPS logic channels and it does not share any functions with the ESF system with the exception of the ability to be bypassed by the operator. The TLU also receives the manual operator division scram inputs and the sensor channel bypass command. The primary TLU output is to the MSIV and RPS OLUs. The TLU also has outputs to the Class 1E displays and isolated outputs to the non-Class 1E displays and annunciators. The TLU is Class 1E.

MPL C71-4010 (Rev. 0, May 18, 1990), "Reactor Protection System, Hardware/Software System Specification," lists the approximately 106 interfaces (inputs and outputs) for each TLU.

MPL A32-4080 (Rev. 0, April 25, 1990), "EMS/SSLC Interface Requirements, Requirements Specification," requires that the TLU include a watchdog timer function on all outputs to ensure constant updating. Time-out shall cause outputs to assume predetermined safe states. These predetermined states were not stated in this MPL and, therefore, the DFSER stated that information on this matter is to be included in the ITAAC verifications. This was DFSER Open Item 7.2.2.3-1. In response to this issue, GE stated that tests have been added to the ITAAC to demonstrate that the failure state is fail-safe for the RPS. In addition, the MPL for the SSLC design specification provides the system failure modes and specifies that the RPS is fail-safe. Therefore, this open item is resolved.

GE identified the operating system/executive and the functional software as the two elements of the TLU software. The TLU configuration is characterized by the nine I/O interface boards connected to the central bus. There are a total of 21 input channels and nine output channels physically connected to each TLU. The details of the configuration are as follows:

a. There are four fiber optic communications interface boards that handle the following data channels:

   1. bypass control inputs - 4 channels
   2. DTM inputs (other divisions) - 3 channels
   3. NMS inputs (other divisions) - 3 channels
   4. recirculating pump trip output - 1 channel
   5. data output to plant computer - 1 channel

b. There are two signal interface boards that handle the following data channels:

   1. DTM input (own division)
   2. NMS input (own division)
   3. operator control input
   4. TLU control output (adjacent division)

c. There are two logic level interface boards that handle the following data channels:

   1. MSIV input - 4 channels
   2. auto reactor trip output - 1 channel
   3. auto MSIV closure output - 1 channel
   4. MSIV test close output - 4 channels

d. There is one contact input interface board that handles the following data channels from the SSLC panel switches:

   1. non-coincidental trip disable input
   2. main condenser vacuum bypass input
   3. auto trip test input
   4. auto isolation test input

The TLUs are essentially identical between the four RPS divisions. There will be identification and small timing differences in some TLUs, however, most of the devices are identical. The TLUs are also similar to the SLUs of the ESF functions. The TLU software algorithms will be significantly different but the operating system, the self diagnostics, and the input/output will be identical. The staff was concerned that the TLUs could be vulnerable to a potential common-mode failure similar to the DTMs. This was an open issue in the DFSER and the resolution is discussed in Section 7.2.8 of this report.

7.2.2.4 Output Logic Unit

MPL C71-4010 (Rev. 0, May 18, 1990), "Reactor Protection System, Hardware/Software System Specification," lists the 6 inputs and 12 outputs for each OLU. The OLU is a Class 1E solid state electronics device (non-microprocessor-based) which receives signals from the TLU within its own electrical division. There are separate units for the MSIV and RPS initiation functions. The OLU also receives the division bypass commands, the manual division trip input for the RPS OLU and manual division isolation input for the MSIV OLU. The output of the OLU units is to the MSIV and RPS load drivers which provide a division level two-out-of-four vote. This is in addition to the separate two-out-of-four sensor input vote which occurs in the TLU.

7.2.2.5 Self-Test System

The self-test system (STS) feature of the SSLC is described in Section 7.1.2.6 of the SSAR. The STS performs testing on both the RPS and ESF functions of the SSLC. The STS was originally presented in the SSAR as an overlay testing and surveillance system which continually and automatically performs end-to-end testing of all active circuitry in the SSLC using short test pulses similar to the Clinton nuclear system protection system design. GE has substantially revised the SSAR description since the DSER was issued to tailor the STS for a multiplexed microprocessor-based design.

GE originally classified the STS as safety-associated and stated that Class 1E equipment would be used wherever the STS interfaced with safety-related equipment. The staff was concerned that the test features were not classified as Class 1E and could potentially degrade the safety function. This issue was identified as an open item in the DSER (SECY-91-294). The staff was also concerned about the potential interaction between the STS and the master/slave configuration in the SSLC causing problems with the SSLC operation and possibly violating the separation requirements of the SRP. The design of the STS as described in the SSAR is now substantially integrated into the SSLC, and GE has committed that all STS equipment will be qualified as part of the SSLC (Class 1E). The revised description and Class 1E designation have also resolved the configuration and separation issues identified in the DSER. This is acceptable and the DSER open issue is, therefore, resolved.

The STS for safety-related systems and the SSLC are required to be Class 1E in accordance with the criteria of the SRP. This had not been explicitly stated in the SSAR and was identified as DFSER Confirmatory Item 7.2.2.5-1. The SSAR has been revised to clarify that the Class 1E requirement will apply for all safety-related self-test or self-diagnostic features and, therefore, this item is resolved. The Class 1E requirement applies to self-test features imbedded in the software of safety-related systems. However, it is acceptable to use non-Class 1E testing equipment in typical configurations where a safety-related I&C channel or system under test is taken out of service to perform manual tests.

The protection system in-service testability requirement comprises a set of six separate tests, which together constitute a complete system test:

1. A manual scram test will de-energize one set of scram pilot valve solenoids at a time. This test will also verify the indications in the main control room and the plant computer input.

2. A calibration check of the NMS will verify calibration of setpoints.

3. The single rod scram test will include a physics review performed before insertion of each rod.

4. A calibration check of the analog sensor inputs at the inputs to the RMUs will verify, by injecting calibrated sensor signals in place of the normal sensor inputs and monitoring the SSLC control room panels, linearity, accuracy, fault response, and downscale and upscale trip responses. This test is accomplished by placing a division of sensors in the bypass position.

5. A check of sensor operation will verify sensor inputs by cross comparison of other channels, by varying the monitored variable, or by substituting a test source. If the test requires disconnecting the sensor from the system, an out-of-service alarm will be given in the main control room.

6. A self-diagnostics test of the SSLC equipment will be run to detect and determine the location of a failure in the functional SSLC system. The self-test provision within each division of the SSLC system consists of an on-line, continuously operating, self-diagnostic monitoring network, and an off-line semi-automatic, end-to-end surveillance program. The self-diagnostic software within each logic processing unit will monitor critical circuit nodes and timing functions, as well as states of registers, memory locations, and program flow and timing. These monitoring functions are designed to detect both internal problems and certain external problems such as corrupted data input. The self-diagnostics will include hardware techniques such as watchdog timers.

The capability to perform on-line protection system testing is included in the CDM. The specific implementation is described in the SSAR. Specifics of the system design features to permit testing are not available. The ITAAC process will verify that the above tests can be accomplished and includes use of the test equipment indicated for the above tests itself to demonstrate some of the other I&C functional requirements in the ITAAC.

One significant feature of the testing that has been described by GE is the elimination of the need to lift leads and install jumpers to perform testing. Such actions have been a significant problem area at currently operating plants. The DFSER stated that this feature should be included as a CDM requirement, which was identified as DFSER Confirmatory Item 7.2.2.5-2. The SSAR has been revised to include the ABWR design testing features that eliminate jumpers and lifting leads, and the TS have been prepared based on these design features. The CDM has been revised to reflect the technician interfaces for testing, and the staff finds it to be acceptable. This confirmatory item is resolved.

Any changes to the design testing features and commitments identified in the SSAR would involve an unreviewed safety question and, therefore, require NRC review and acceptance prior to implementation. Any requested changes to these commitments shall either be specifically described in the COL application or submitted for license amendment after COL issuance.

7.2.3 Indication of Bypassed and Inoperable Status

MPL C71-5030 (Rev. 0, April 23, 1990), "Reactor Protection System Verification and Validation Criteria Design Specification," requires that only one sensor channel be bypassed at a time. However, the ABWR design does not include the capability to bypass an individual sensor. The sensor bypass referred to in this MPL is the same as that which the staff and GE have defined as a division of sensors bypass. The ITAAC in CDM 3.4, Instrumentation and Control, includes a verification that only one division of sensors may be bypassed at a time.

MPL C71-4010 (Rev. 1, July 2, 1990), "Reactor Protection System Design Specification," states that one sensor channel may be bypassed and the coincidence logic will go to a two-out-of-three vote at the TLU. This bypass is implemented at the input to the TLU. Bypass of one division of output logic at the OLU will result in a two-out-of-three coincidence logic. This bypass is implemented at the input to the OLU. The above MPL document also requires that bypass status be readily apparent and under direct control of the operator.

When any part of the RPS or supporting systems is bypassed or made inoperable, a continuously displayed status of this condition is required in the main control room. SSAR Section 7.2.2.2.1 states that this requirement is met with individual indicator lights grouped near the affected equipment. The following alarms and annunciators are provided for the RPS bypass and inoperable status and are qualified as Class 1E:

RPV level 3 scram bypassed
RPV pressure high scram bypassed
Drywell pressure high scram bypassed
Neutron flux high scram bypassed
MSIV closure scram bypassed
CRD charging water pressure low scram bypassed
MSL radiation high scram bypassed

The following alarm is provided for the RPS inoperable function as non-Class 1E:

Indicated RPV water level invalid

There are also additional commitments for all bypasses to be annunciated, however, the alarm and annunciator list from which this information was extracted did not list all of the non-Class 1E alarms and annunciators but only those specifically identified in the Emergency Planning Guidelines (EPGs). To address this issue, the staff reviewed the ABWR SSAR for commitment to RG 1.47, "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems," Position C.2, which requires that the bypassed and inoperable status of RPS auxiliary or supporting systems be automatically indicated in the main control room. The description in the SSAR indicates that adequate indication and annunciation for these systems will be provided in the main control room. The indication is automatic at the system level when the system loses power or when it is out of service. A switch will be provided for manual initiation of bypass indication for out-of-service conditions which could not be automatically annunciated. The staff concludes that GE has provided adequate commitment to RG 1.47. The DFSER listed the verification that all bypasses are appropriately annunciated as DFSER Confirmatory Item 7.2.3-1. Bypass testing has been included in the CDM, and this item is, therefore, resolved.

7.2.4 Alarms and Annunciators

For the ABWR, the most significant change in alarm and annunciator design from previous designs is that many of the alarms are now Class 1E and as such will be powered from the divisional Class 1E power sources. These Class 1E alarms are distributed between the fixed display (fixed alarm tiles above the mimic section) and the VDU. As a part of the ABWR design, the RPS channel trips will annunciate and identify the variable which caused the trip. Loss of annunciator events at currently operating plants have demonstrated that this is important information for the operator. SSAR Chapter 18 and the corresponding section of this report for the human factors review provide a discussion and evaluation of the displays.

The ABWR has committed to using non-Class 1E "uninterruptable power supplies" for the non-Class 1E annunciators. Details on the distribution of the non-Class 1E power loads were not provided by GE in the SSAR. However, independence of Class 1E and non-Class 1E power supplies and isolation of non-Class 1E alarms from Class 1E alarms will ensure that the Class 1E alarms would be available in the event of loss of the uninterruptable power supplies.

At the staff's request, GE provided an inventory of control room instrumentation, including a list of Class 1E alarms, based upon the EPGs. Because the inventory is emergency mitigation based, it does not include a list of the non-Class 1E alarms that are not required for entry conditions into the EPGs. Additional discussion on the inventory is provided in Section 18.0 of this report.

The number of fixed position alarm tiles in the ABWR control room has been significantly reduced compared to current operating plants by employing alarm prioritization and filtering. In the ABWR, the fixed alarm tiles are only used for important plant-level alarm conditions that potentially could affect plant availability and plant safety, or indicate the need for immediate operator action. Less critical alarms are presented on the operator console VDUs. In addition, the large control room overview display panel will include important safety-related (Class 1E) system-level alarms and some non-Class 1E system-level alarms.

The following is a list of Class 1E alarms for the ABWR. The fixed displays will be primarily alarm tiles near the large overhead mimic. The term "VDU" refers to the CRTs or plasma screens on the control room consoles. (There are several VDUs, and this list is not intended to imply that all the alarms are on one screen.)

The Class 1E alarms that indicate RPS information include:

(1) RPV water level (fixed)
(2) drywell pressure high (fixed)
(3) drywell pressure high (VDU)
(4) RPV water level 3 (fixed)
(5) RPV water level 3 (VDU)
(6) RPV pressure high (fixed)
(7) RPV pressure high (VDU)
(8) neutron flux high (fixed)
(9) neutron flux high (VDU)
(10) MSIV closure (fixed)
(11) MSIV closure (VDU)
(12) CRD charging water pressure low (fixed)
(13) CRD charging pressure low (VDU)
(14) main turbine stop valve closure (VDU)
(15) main turbine control valve fast closure (VDU)
(16) MSL radiation high (fixed)
(17) MSL radiation high (VDU)
(18) RPS Div I trip (VDU)
(19) RPS Div II trip (VDU)
(20) RPS Div III trip (VDU)
(21) RPS Div IV trip (VDU)
(22) RPS Div I manual trip (VDU)
(23) RPS Div II manual trip (VDU)
(24) RPS Div III manual trip (VDU)
(25) RPS Div IV manual trip (VDU)
(26) manual scram (A) initiated (VDU)
(27) manual scram (B) initiated (VDU)
(28) reactor scram (fixed)
(29) reactor period short (fixed)
(30) SRNM neutron flux upscale rod block (VDU)
(31) SRNM neutron flux upscale reactor trip (VDU)

The non-Class 1E alarms that annunciate RPS information include:

(1) drywell pressure high (fixed)
(2) RPV pressure high (fixed)

In addition to the above displays of alarm conditions, access to the plant computer will allow the operator in the control room to identify the specific sensor or sensors that caused a channel trip. The plant computer will also maintain the sequence of events log. This log will contain both nuclear steam supply system and balance of plant inputs. The plant computer is not part of the safety systems and no credit is taken for this system in meeting SRP acceptance criteria. The plant computer is described in Section 7.7.1.5 of this report.

The ABWR design provides for the distribution of alarms between the fixed displays, the VDUs, and the large overview panel; the independence of divisionalized Class 1E and uninterruptible non-Class 1E power supplies that provide power to the alarm systems; and the isolation of non-Class 1E alarms from the Class 1E alarms. These provisions in the design minimizes the potential for total loss of annunciators due to single failures as experienced by operating nuclear plants and meet the requirements of redundancy, independence, and separation of alarm systems of the Commission's position on control room annunciator reliability in SECY-93-087.

7.2.5 Support Systems

The support systems discussed in this section include the line power supplies and the HVAC systems.

SSAR Chapter 8 and the corresponding section of this report discuss the plant station power. For the SSLC, the power is supplied by four independent and separated Class 1E 120 Vac sources, each of which is backed up with a Class 1E 125 Vdc battery source through an invertor. Divisions I and III of the 120 Vac sources are each supplied power from the corresponding division 480 Vac power source; 120 Vac Divisions II and IV are supplied from the Division III 480 Vac source as indicated by SSAR Figure 8.3-3. SSAR Figure 7.2-1 initially incorrectly showed a fourth division of 480 Vac power and, therefore, needed to be revised to reflect the power supply design. This was DFSER Confirmatory Item 7.2.5-1. SSAR Figure 7.2-1 was subsequently corrected and this item is, therefore, resolved.

MPL C71-4010 (Rev. 1, July 2, 1990), "Reactor Protection System Design Specification," is consistent with the SSAR and the CDM in that it calls for four Class 1E 120 Vac systems. However, the specifications in the MPL identified only two 125 Vdc systems whereas the ABWR CDM and SSAR stated that there are four 125 Vdc systems provided. GE indicated in the SSAR that the four 125 Vdc system configuration is the one for the ABWR design, and this is acceptable to the staff.

The scram pilot valve solenoids are powered from the Divisions II and III 120 Vac SSLC buses.

A potential Division II 6.9 KV/480 Vac transformer or 480 Vac switchgear failure could disable the 480 Vac power sources to the inverters associated with Divisions II and IV 120 Vac SSLC buses. Such a failure could also affect the inverters and result in disabling the capability to use the backup 125 Vdc sources. The failure of the inverters would lead to degradation or loss of power to the two 120 Vac SSLC buses. This issue was addressed in two ways. The first is by incorporating a TS requirement that limits the bypass of SSLC channels. This was an open item in Chapter 16 (Item 16-2) of the DFSER and the resolution is discussed further in Section 7.11 of this report. The second way is by providing electrical protection assemblies (EPAs) for the SSLC buses in the ABWR electrical distribution system design. EPAs provide an additional level of protection to the SSLC power supplies. Each assembly consists of a circuit breaker with a trip coil driven by logic circuitry which senses line voltage and frequency, and trips the circuit breaker open on a condition of overvoltage, undervoltage, or underfrequency. The EPAs will detect a spectrum of degraded conditions of the SSLC bus power supply and open the power supply line in time to prevent serious damage to connected equipment. Another area of benefit by using EPAs is reduction in the possibility of the scram pilot solenoid valves sticking as a result of damage caused by insufficient voltage supply to the solenoid coils. The ABWR design requires the coils of both scram pilot solenoid valves of each CRD to disengage when power is removed, to initiate a scram. The EPAs will detect an undervoltage condition and remove power prior to possible damage to the coils. Normal voltage drop between the location of the EPAs and the solenoid coils will be included in the design considerations. The ITAAC for the EPAs will include a verification that the wiring to the solenoid valves is sized so that the voltage drop in the cables will not result in insufficient voltage being supplied to the solenoid coils. The ITAAC will also include a verification that the neutral leads of the scram pilot solenoid valve coil windings are configured such that credible faults (e.g., hot shorts) will not prevent the valves from performing their safety function. The above ITAAC considerations were identified as DFSER Confirmatory Item 7.2.5-2. GE has included this information in the SSAR and the staff finds it to be acceptable. Therefore, this item is resolved.

The four primary SSLC cabinets (or set of cabinets) are installed in the main control room. Portions of the RPS and SSLC (in particular, the RMUs) are located in other mild environment areas such as the control building equipment rooms. The RPS I&C equipment which is designed for harsh environments and will be required to meet the electric equipment environmental qualification requirements of 10 CFR Part 50.49 are certain sensors, sensor lines, transmitters, and associated cabling. None of the microprocessor-based equipment, multiplexor units, or fiber optic links will be in areas designated as a harsh environment area. All of the SSLC equipment is qualified to the environment of the room in which it is located. The support system is the HVAC system for the specific area. Section 7.2.8 of this report describes the hardware and software qualification.

7.2.6 Defense-in-Depth Analysis

The I&C systems for the ABWR help ensure that the plant operates safely and reliably by monitoring, controlling, and protecting critical plant equipment and processes. Both safety and non-safety I&C systems for the ABWR are primarily digital-based and differ significantly from the primarily analog systems used in currently licensed operating plants. The digital I&C system shares more data transmission functions and process equipment than was the practice with the analog systems. The ABWR I&C systems use the same software and processing equipment (hardware) across the safety divisions, and therefore, a hardware design error, a software design error, a software programming error, or a maintenance error may affect all of the I&C system divisions and result in a common-mode or common-cause failure of redundant equipment. The staff was concerned that the use of digital computer technology in I&C systems could result in safety-significant common-mode failures. Because of these concerns, which were expressed early in the preliminary reviews of the ABWR, the staff requested that GE prepare an analysis of ABWR I&C system defense-in-depth based upon NUREG-0493, "A Defense-in-Depth and Diversity Assessment of the RESAR-414 Integrated Protection System" (1979). The staff believes that a level of diversity is necessary to provide defense-in-depth against potential common-mode software errors so that necessary safety functions can be performed reliably. The staff considers common-mode software errors to be a special case of single failure and, therefore, protection against such errors is to be part of the design bases.

To complicate the concern with regard to common-mode failure, the initial SSAR did not include the design details that would allow the staff to independently assess the diversity and defense-in-depth of the design and conformance with the guidelines of NUREG-0493. Therefore, the staff concluded that the common-mode failure potential of software had not been adequately addressed, and this was identified as an open item in the DSER. Subsequently, GE provided the results of analyses done to address the defense-in-depth issue as amendments to the SSAR. Resolution of this item is discussed further in this section of the report.

BACKGROUND

The first design reviewed by the staff specifically to address defense against potential common-mode failures in digital systems was the Westinghouse RESAR-414 design. The results of this study were published in NUREG-0493, "A Defense-in-Depth and Diversity Assessment of the RESAR-414 Integrated Protection System," (March 1979). NUREG-0493 discussed common-mode failures and different types of diversity, and presented a method for assessing the defense-in-depth of the design.

The staff described concerns with common-mode failures and other digital system design issues in SECY-91-292, "Digital Computer Systems for Advanced Light Water Reactors." SECY-91-292 describes how common-mode failures could defeat not only the redundancy achieved by the hardware architectural structure, but also could result in the loss of more than one echelon of defense-in-depth provided by the monitoring, control, reactor protection, and engineered safety functions performed by the digital I&C systems. The two principle factors for defense against common-mode/common-cause failures are quality and diversity. Maintaining high quality will increase the reliability of both individual components and complete systems. Diversity in assigned functions for both equipment and human activities, equipment, hardware and software, can reduce the probability that a common-mode failure will propagate.

The modules in the ABWR SSLC are to be implemented by microprocessor-based designs with identical or similar hardware and software used in all four divisions. Because of this similarity the concerns expressed in NUREG-0493 and SECY-91-292 apply directly to the SSLC.

The staff reviewed the ABWR I&C system design and concludes that common-mode failure concerns were valid for several reasons.

1. The commonality of the timing between channels is such that an error in one channel is expected to occur in all identical channels and equipment within a few milliseconds of each other.

2. The possibility exists that an initiating transient or accident itself creates the set of circumstances that reveal a software error. The staff considers that the models of the systems used to develop the test sets may not contain sufficient inputs for all transient and accident situations, and therefore, certain situations may not be adequately tested. In most software applications, including the ABWR design, it is not possible to include a 100-percent test of all software inputs due to the very large number of possible input combinations.

3. There is presently no general consensus as to a method to demonstrate a quantitative measurement of software reliability. Some assumptions, such as the GE ABWR PRA assumption of 4.25×10^{-8} failures per demand for the EMS, cannot be demonstrated with the current software metrics or existing data bases, and are likely unrealistically low.

4. Redundancy in software does not necessarily increase the reliability and availability of the overall system to an acceptable level as it can in analog systems. Some of the failure modes of a software-based system (particularly the common-mode failures mentioned above) are fundamentally different from those of an analog system.

5. Self-diagnostics and periodic testing provide a significant safety improvement by reducing the possibility of undetected failures during plant operation, but they do not prevent the failures from occurring. The improved self-diagnostics of digital systems do not resolve the common-mode failure issue. The analysis described subsequently in this report explains the common-mode vulnerabilities of the ABWR design, and the attributes which mitigate those failures.

Several NRC regulations and industry standards address the need for defense against potential common-mode failures. GDC 22 requires that "design techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protection function." IEEE Standard 279-1971 requires that "equipment, not subject to failure caused by the same credible event, shall be provided to detect the event . . . ." IEEE Standard 603-1980 has the same requirement as IEEE-279. IEEE Standard 379-1968 states that "certain common-cause failures shall be treated as single failures when conducting the single failure analysis. Such failures can be in dissimilar components and can have dissimilar failure modes. Failures resulting from cascaded failures and from design bases events have already been discussed and are those which shall be included in the analysis. Common-cause failures not subject to single-failure analysis include those that can result from external environmental effects, design deficiencies, manufacturing errors, and operator errors. Design qualification and quality assurance programs are intended to afford protection from external environmental effects, design deficiencies, and manufacturing errors. Personnel training, proper control room design, and operating and maintenance procedures are intended to afford protection from maintenance and operator errors." Common-mode failure issues are also addressed in the requirements of 10 CFR 50.62 concerning mitigation of anticipated transients without scram (ATWS).

There are several different types of diversity, each of which offers certain protection against the common-mode failures. Various forms of diversity include signal diversity, equipment diversity, aspect diversity, and people diversity. Signal diversity includes the use of different signals (sensors) to initiate an action, such as neutron flux and reactor pressure as diverse signals for initiation of reactor scram. Equipment diversity includes using different kinds of equipment to perform a function. An example of equipment diversity described in NUREG-0493 is the use of relay vs solid-state logic in the I&C system. The ABWR design employs equipment diversity in that the remote shutdown station (RSS) is a hardwired analog system and is, therefore, diverse from the microprocessor-based SSLC. Included in the equipment diversity category is the use of different software languages. Aspect diversity involves using different logic levels. An example for the ABWR is the functional algorithm diversity between the DTM functions within a channel. Another example of aspect diversity in the ABWR design is the de-energize to actuate (fail-safe) aspect for the RPS actuation vs. the energize to actuate (fail as-is) aspect of the ARI. People diversity refers to using different groups of people to design, verify, validate, and maintain an I&C system. Examples of people diversity required in the ABWR design are the different groups performing the I&C system design vs. those performing the QA function, and the independence between the software verifier and the software designer. It is difficult to define how much improvement in safety results from a given kind or degree of diversity. For microprocessor design, this is especially difficult because there is no industry consensus on a method to quantify software reliability and/or availability.

DISCUSSION OF THE RESOLUTION

GE's initial analysis of common-mode failure was provided in Appendix 7A of the SSAR. In the SSAR, GE stated that the use of shared sensors in the design may increase the effects of potential common-mode failures. Therefore, the SSLC system architecture is designed to provide maximum segregation of system functions by using separate DTMs and TLUs within each of the four I&C divisions. In the analysis it was also noted that for the reactor shutdown function there are five different methods for controlling reactivity including the RPS hydraulic scram, the air header dump valves of the ARI system, the fine motion control rod drive (FMCRD) insert function of the ARI system, the SLCS, and the CRD system. The reactor core cooling function can be accomplished as described in the SSAR by four different systems including the motor driven (FDWC) system, the motor driven high-pressure core flooder system (HPCF), the turbine driven RCIC system, and the low-pressure mode of the RHR system. Appendix 7A also described the RSS which provides diverse (hardwired) core cooling control functions. GE concluded that the ABWR meets the intent of NUREG-0493 for I&C system diversity. The staff, however, concluded in the DSER that the GE analysis did not adequately address potential loss of safety functions due to postulated common-mode failures, and that this was an open issue.

The staff considers the two principle factors for defense against common-mode failures to be quality and diversity. The quality aspects of the ABWR I&C systems are addressed in other sections of this report. Quality, in part, is achieved by the use of quality design standards for the hardware and software, and the I&C system testing to be performed. With few exceptions, the staff concluded in the DFSER that the quality issues had been substantially resolved.

Though there were some studies performed by GE in the design process, no analyses were presented to the staff which adequately demonstrated how the SSLC design complied with NUREG-0493, and thereby provided adequate defense against potential common-mode failures. The staff concluded in the DFSER that this concern had not yet been addressed adequately by GE. The staff identified the defense-in-depth issue as DFSER Open Items 7.2.6-1, 7.2.6-2, and 7.2.6-3.

Because the staff determined that the ABWR SSAR and related documentation had not adequately addressed the common-mode failure and defense-in-depth concern, the staff performed its own common-mode failure assessment of the ABWR based upon the guidance of NUREG-0493. Some additional considerations were added to the original NUREG-0493 approach, such as evaluation of information available to the operator, common-mode failures during accidents as well as transients, the time available for systems actuation, and the use of non-Class 1E systems to provide a diverse means of accomplishing safety functions if the first-line safety systems failed. The staff's defense-in-depth and diversity assessment of the ABWR design was performed by the Lawrence Livermore National Laboratory (LLNL) (under contract to the staff - referred to as the LLNL diversity study). The results were made available to GE for determination of any factual errors in the design assessment. GE informed the staff of some errors in the study which have been corrected.

The LLNL diversity study evaluated I&C system function for all of the SSAR Chapter 15 events. This set of events was judged by the staff to be sufficiently complete to bound those situations requiring initiation of safety systems based on the NUREG-0493 methodology. The study evaluated each event in conjunction with a set of postulated common-mode failures. Two specific events were selected for detailed study in the preliminary stages of the review. Those events were generator load rejection with normal bypass, and steam system piping break outside containment. Assumptions made by LLNL in the analysis during the review were documented in the LLNL diversity study. One aspect that was not specifically evaluated in the study was anticipated operator actions. As described later in this report, operator actions have now been considered.

In the LLNL diversity study several areas of concern were identified. The use of the EMS for the RPS, ESF and information to the operator was a particular vulnerability, along with the common elements shared by the DTMs and TLUs. In the study it was concluded that, in general, there was information and system controls to mitigate each of the transients investigated, however, there may not be sufficient time and information to complete all necessary activities manually to maintain safety functions, especially if the actions required the use of the RSS.

Due to the significance of the EMS to the proper functioning of both the RPS and ESF, a common-mode failure of the EMS was a significant concern of the staff in the resolution of the defense-in-depth and diversity issue. The modules identified in the EMS are to be implemented by microprocessor based designs with similar hardware in all four divisions. MPL A11-4121 (Rev. a (preliminary), January 19, 1988), "Multiplexing System Application Procedure, Design Procedure," specifically states that the individual multiplexing systems shall be designed with a high degree of standardization. The DTM and TLU/SLU components of the SSLC also share common software design features which would result in a failure of all four channels if there is a software error.

In reviewing the results of the study, the staff found diverse I&C system features in the ABWR design, several of which GE had previously presented as solutions to the potential common-mode software error concern. In the study, credit was given to non-Class 1E systems to mitigate an event if the non-safety systems were reasonably expected to be available. Thus, from the study the staff determined that the ABWR design has a number of attributes which provide defense-in-depth and protection against a potential software error as follows:

1. The turbine inputs to the RPS are hardwired (do not use the EMS). Further, the NMS and the PRM system RPS inputs use microprocessors but are directly wired to the SSLC and do not use the EMS. Therefore, an EMS common-mode failure would not disable these inputs.

2. Manual scram functions and manual MSIV actuation are hardwired from the control room, do not use the EMS, and are not dependent upon microprocessors.

3. The ARI function, that is part of the ATWS system, is independent of the EMS. The ARI system is a two-out-of-three logic initiation non-Class 1E system that is separate and independent from the RPS. By letter dated June 2, 1993 (on important features identified by the ABWR PRA), GE identified this feature as providing a significant factor in the reduction in the probability of an ATWS.

4. The NEMS is diverse in both hardware and software from the EMS. The NEMS is a non-Class 1E system.

5. The RSS is conventionally hardwired from the station itself to the actuation devices (does not use the EMS) and does not use microprocessors. The RSS is outside of the main control room as required for its primary design function of shutting down the reactor upon abandoning the main control room. The RSS is a two channel control and indication station which contains most of the ESF capabilities.

6. Final display information for the operator is provided by a separate, diverse means -- the Class 1E fixed mimic display and the Class 1E divisional VDUs. Alarms and parameter information are also available on the non-Class 1E VDUs and from the plant computer.

7. The NMS bypasses the DTM and is input directly to the TLU. A common DTM failure will not fail the NMS scram function. The ARI system controls also bypass the DTM because the ARI system is not part of the SSLC.

8. The DTMs and TLU/SLUs have a significant level of functional diversity between the RPS and ESF functions and between portions of the ESF. As shown on Figures 7.3-1 and 7.3-2 of this report, there are three DTMs, one TLU and two SLUs per channel of the SSLC. The software algorithms within the SSLC are functionally diverse between the RPS and ESF functions as a result of the equipment to be tripped/actuated, i.e., deenergize-to-operate function for RPS and MSIV, and energize-to-operate function for ESF.

9. GE stated in the SSAR that the SSLC software will be relatively simple which will result in a high degree of assurance that the required testing will reveal virtually all of the software errors. The EMS software is even simpler. However, no supporting analysis has been provided to support these conclusions. All the safety-related software will be verified and validated. The staff concludes that software, even that which has been verified and validated with a high-quality program may still have undetected errors. This was DFSER Open Item 7.2.6-4. In a letter dated April 30, 1993, GE provided a description of the software development program, a commitment to provide a set of hardwired backups, and a revised common-mode failure analysis. In the letter, GE concluded that a software simplicity analysis was no longer relevant since the quality and diversity issues for software will address the staff's concerns regarding software reliability and digital system defense-in-depth. The staff agrees that this analysis is no longer necessary as the basis for resolving the defense-in-depth questions. These concerns are addressed in the computer ITAAC and overall digital system diversity, and, therefore, Open Item 7.2.6-4 is resolved.

Based on the two examples reviewed in detail in the LLNL diversity study, the staff and GE concluded that some postulated failures in the SSLC (and EMS) would disable the displays and controls of the ESF systems in the main control room. Some postulated failures would also result in the loss of a significant amount of information to the operator in the control room. Another conclusion was that it would be necessary to access the RSS in order to initiate the ESF equipment and mitigate the events under consideration. For the events reviewed, there was display and mitigation capability available for all postulated failures. The mitigation control function was occasionally at the RSS or was provided by a non-Class 1E system. Normally available control systems, such as the FDWC system, were credited in the analysis, while systems not normally used, such as the fire water system for reactor water injection, were not.

The results of the LLNL diversity study were presented to GE. The staff identified the following four primary concerns arising out of the study:

1. The response to the initiating event needed to be confined to the main control room.

2. Consideration of the time available for manual operator actions was incomplete in the analysis.

3. System level actuation capability from the control room for the ESF functions was lacking.

4. Display of the necessary Class 1E variables in the main control room was lacking.

The staff requested GE to complete its review of the LLNL diversity study and respond to the above concerns. The above results were also provided to the Commission for approval in SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs," April 2, 1993, as the staff's generic position for digital system defense against common-mode failures for advanced light-water reactors.

The staff position on I&C system diversity for ALWRs as approved by the Commission in a SRM dated July 21, 1993, is as follows:

1. The applicant shall assess the defense-in-depth and diversity of the proposed I&C system to demonstrate that vulnerabilities to common-mode failures have been adequately addressed.

2. In performing the assessment, the vendor or applicant shall analyze each postulated common-mode failure for each event that is evaluated in the accident analysis section of the safety analysis report (SAR) using best-estimate methods. The vendor or applicant shall demonstrate adequate diversity within the design for each of these events.

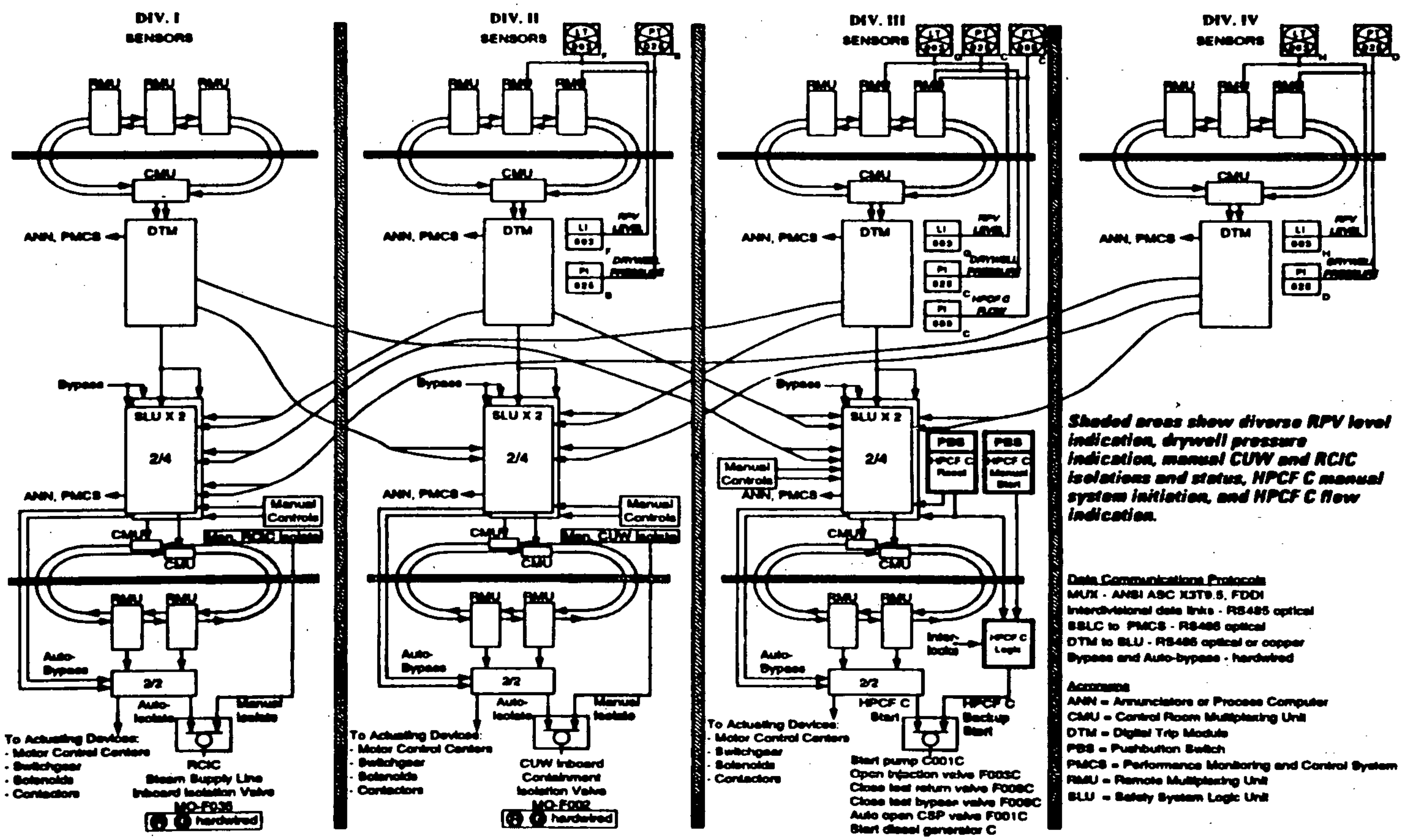
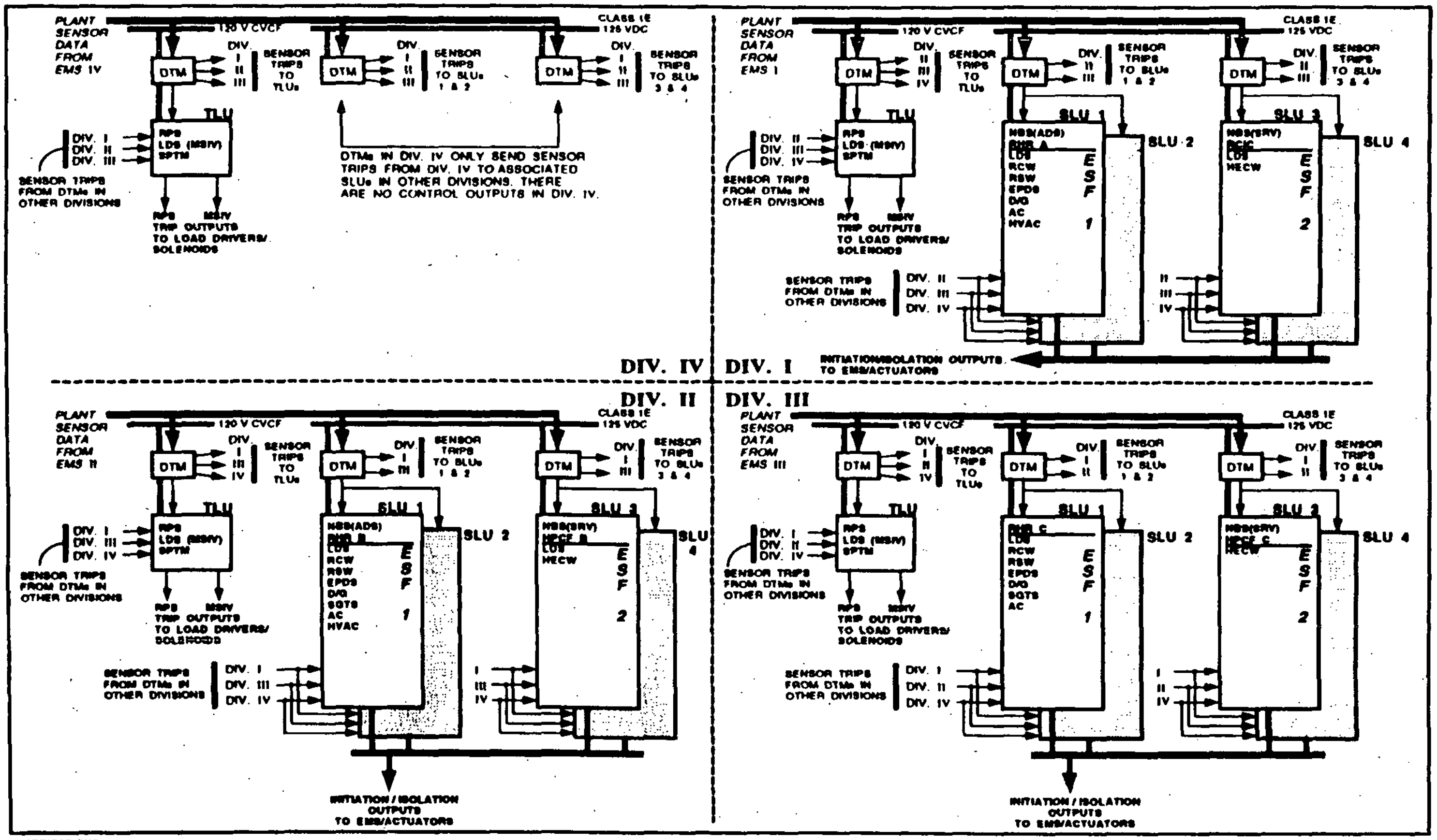


Figure 7.3-1 Safety system logic and control communications path for engineered safety systems



ABBREVIATIONS:

DTM = DIGITAL TRIP MODULE
EMS = ESSENTIAL MULTIPLEXING SYSTEM
SLU = SAFETY SYSTEM LOGIC UNIT
TLU = TRIP LOGIC UNIT

AC = ATMOSPHERIC CONTROL
D/G = DIESEL GENERATOR
EPDS = ELECTRICAL POWER DISTRIBUTION SYSTEM
ESF = ENGINEERED SAFETY FEATURES
HECW = HVAC EMERGENCY COOLING WATER
HVAC = HEATING, VENTILATING & AIR CONDITIONING
LDS = LEAK DETECTION & ISOLATION SYSTEM
MSIV = MAIN STEAM ISOLATION VALVE
NBS = NUCLEAR BOILER SYSTEM

NMS = NEUTRON MONITORING SYSTEM
PRRM = PROCESS RADIATION MONITORING
RCIC = REACTOR CORE ISOLATION COOLING
RCW = REACTOR BUILDING CLOSED COOLING WATER
RHR = RESIDUAL HEAT REMOVAL
RPS = REACTOR PROTECTION SYSTEM
RSW = REACTOR SERVICE WATER
SGTS = STANDBY GAS TREATMENT SYSTEM
SPTM = SUPPRESSION POOL TEMPERATURE MONITORING

NOTES:

1. NMS AND PRRM (NOT SHOWN) ARE STANDALONE SYSTEMS WITH TRIP OUTPUTS TO RPS AND ESF CONTROLLERS OF SSLC.
2. POWER SOURCES (PER DIVISION)
   EMS: CLASS 1E, 125 VDC
   ESF 1/ESF 2: CLASS 1E, 125 VDC
   RPS/MSIV: CLASS 1E, 120 V CVCF
   NMS/PRRM CLASS 1E, 120 V CVCF

Figure 7.3-2 Assignment of interfacing safety system logic to safety system logic and control controllers

3. If a postulated common-mode failure could disable a safety function, then a diverse means, with a documented bases that the diverse means is unlikely to be subject to the same common-mode failure, shall be required to perform either the same function or a different function that provides adequate protection. The diverse or different function may be performed by a non-safety system if the system is of sufficient quality to perform the necessary function under the associated event conditions.

4. A set of displays and controls located in the main control room shall be provided for manual system-level actuation of critical safety functions and monitoring of parameters that support the safety functions. The displays and controls shall be independent and diverse from the safety computer system identified in Items 1 and 3 above.

The staff's proposed applicable regulation for digital system defense against common-mode failures for the ABWR design is, as follows:

Digital instrumentation and control systems provided for the standard design must include:

(1) an assessment of the defense-in-depth and diversity of instrumentation and control systems,

(2) a demonstration of adequate defense against common-mode failures, and

(3) provision for independent backup manual controls and displays for critical safety functions in the control room.

GE completed its review using events that GE believes envelope the Chapter 15 events. GE disagreed with the staff's position and indicated that adequate defense-in-depth and diversity should consider the likelihood of the postulated events in conjunction with the assumed common-mode failures. Under these circumstances, credit for the RSS to mitigate the event should be permitted.

In response to the GE position, the staff prepared a list of the set of equipment which it believed was necessary to bring the ABWR design into compliance with the proposed staff position and which were necessary for a demonstration of appropriate diversity. The list was based on a functional, symptom-based approach to accident mitigation to assure that adequate reactivity control, core cooling, reactor coolant system integrity, and primary containment integrity are maintained for all events. This list was also based upon a review of the control functions identified in the EPGs and the post-accident monitoring indications identified in RG 1.97.

GE completed three studies related to furthering the issue of defense against common-mode/common-cause failures in the ABWR I&C systems. The first was an analysis of the common-mode failure of ABWR multiplex equipment (SSAR Appendix 19N, Amendment 22). This study identified the following potential common-mode failure mechanisms:

(1) Earthquake
(2) Loss of dc power
(3) Loss of cooling
(4) Sensor miscalibration
(5) RMU miscalibration
(6) Set point drift
(7) Maintenance/test error
(8) Manufacturing error
(9) EMI
(10) Fire
(11) Software fault

In the study, GE addressed each of the above issues. Most were evaluated as not being credible causes due to the qualification of the I&C equipment, physical separation, or administrative controls. The study contained the conclusion that common-mode software fault is a credible, although unlikely, possibility. GE committed to administrative controls to minimize errors, TS requirements to assure failure detection, and symptom-based procedures to assure that adequate core cooling is maintained in the event of a common-mode EMS failure.

The second GE study relating to the issue of defense against common-mode failures was the preparation of an inventory of controls, displays, and alarms relied upon for accident mitigation based on the EPGs, including their locations. This information was submitted as SSAR Appendix 18F. Based on the information provided in the SSAR, GE demonstrated that the displays will have separation and some diversity. The diversity for the safety-related displays is primarily provided between the fixed mimic panel and the safety channel VDUs.

The third study prepared by GE was an event-based common-mode failure evaluation. (The result of the study was presented to the staff at a meeting held at the GE offices in San Jose, California on August 26, 1992, and was documented as part of the minutes of that meeting). In performing the study, GE evaluated 14 events from the SSAR Chapter 15 transients and accidents, considering the emergency operating procedure entry conditions and a postulated common-mode failure in the ABWR I&C system. The study addressed the automatic actions that would occur following each event coincident with the postulated common-mode failure. The common-mode failure postulated for each of the events was an undiscovered fault in the EMS resulting in all valid and correct control and monitoring data transmissions being lost. The emergency operating procedure (EOP) entry conditions were then considered. In several events, more than one monitored parameter is expected to indicate the need to enter the EOPs. Indication of reactor pressure vessel water level low was the entry condition for many of the events; drywell pressure high indication was the entry condition for several others. The study considered operator actions described in the EOPs, with the equipment which was not disabled by the postulated common-mode failure. The reactor and containment response was evaluated for each event and was compared to the SSAR Chapter 15 analysis (which does not assume common-mode failures of I&C equipment). A summary of each event was provided which concluded that, for one hour or more, sufficient water was available for decay heat removal.

A significant factor considered in this analysis was the function of the feedwater system during the postulated transient or accident. For each of the events (except for feedwater line breaks or failures) the feedwater system was assumed to remain operational. The feedwater system was also assumed to function properly in response to the specific event scenario. For example, a loss-of-coolant accident (LOCA) inside containment, a loss of condenser vacuum, or a loss of the auxiliary power transformer requires a reduction in feedwater flow to the reactor vessel. GE stated that reduced flow was a normal expected function of the feedwater system under the above event condition. If the feedwater system continues to run at 100 percent flow, the available water inventory would be exhausted in approximately 4 minutes. If the feedwater system responds as assumed in this study, water inventory would be available for over an hour. Based on this study, GE concluded that there was adequate capability in the ABWR design to mitigate each event for sufficient time until the RSS capabilities could be used.

The staff reviewed the GE study as documented in the SSAR and agreed with the conclusions. However, the staff believes that GE relied too heavily upon the feedwater system as a backup to safety-related RPV inventory means; therefore, additional backup capability should be provided in the main control room. This was discussed further in the DFSER.

CONCLUSION

The staff concludes in the DFSER that the following additions to the ABWR design are needed in order to provide acceptable defense-in-depth against potential common-mode failures of the I&C systems:

1. The following control capability will be added in the control room:

   a. clean up water line isolation valve (inboard) manual initiation. This is required by both Items 3 and 4 of the above staff diversity position.

   b. RCIC steamline isolation valve (inboard) manual initiation. This item is required by both Items 3 and 4 of the staff position listed above.

   c. HPCF system manual initiation. In the diversity analyses, many of the events rely on the feedwater system to mitigate the event in combination with the postulated common-mode failure. Experience with feedwater systems at operating plants raises a concern that they may not be reliable enough to satisfy the Item 3 requirement that the backup system be of sufficient quality to perform the necessary function under the associated event conditions. The ABWR feedwater system uses a triplicated I&C system and electric motor-driven pumps which should provide a higher degree of reliability than in currently operating plants. However, the reliability of the I&C system, especially responding to transients and accidents for which it was not specifically designed, cannot be determined. By letter dated June 2, 1993, GE provided a description of the important features identified by the ABWR PRA. This report identified the HPCF logic and control as an important item and stated that "although the probability of a common cause failure of the SSLC is very low, an independent and diverse means of HPCF operation further reduces the risk associated with system operation through the multiplexed digital SSLC." Therefore, the staff concluded that it is prudent to require HPCF backup capability to provide water to the reactor vessel. HPCF backup control in the main control room provides additional assurance that long term core cooling will continue to permit additional time for implementation of the emergency action plan, manning of the RSS, or other long term actions. The staff concluded that only one channel of HPCF capability was needed, and that system-level actuation be provided at the lowest level in the safety computer system architecture. The controls may be hardwired either to analog components or to simple, dedicated, and diverse software-based digital equipment that performs the system-level actuation logic. The above staff conclusions are consistent with the Commission approved diversity position.

2. The following display capability will be added in the control room:

   a. RPV water level
   b. RPV water level (level 3) alarm
   c. Drywell pressure
   d. Drywell pressure (high) alarm
   e. CUW isolation valve status
   f. RCIC steamline isolation valve status
   g. HPCF flow.

The displays may be analog components or simple, dedicated, and diverse software-based digital displays.

3. The RSS displays will be operable during normal operations. This will permit an operator to assess the status of the displayed parameters without transferring control from the main control room.

4. The FDWC system will be designed and tested to verify its availability consistent with the event analyses described above. Testing is required to provide an appropriate level of assurance that the feedwater system can respond to the conditions assumed in the analyses which are beyond the system's normal design basis. The inclusion of only a single I&C channel of high-pressure water injection capability in the main control room is predicated upon satisfactory operation of the feedwater system during the analyzed events as demonstrated by testing. For each event analyses of SSAR Chapter 15 (the GE and LLNL analyses referenced above) which shows that feedwater provides mitigation following the postulated common-mode failure of the safety-related I&C systems function, the FDWC system shall be tested using simulated inputs to demonstrate that it will perform as assumed in the analyses. This requirement is added to the ITAAC.

The above staff conclusions were provided to GE. GE has completed the last of the analyses demonstrating adequate core cooling. The analysis technique was recently revised when GE discovered that the code being used to model the events was not adequate for the steam cooling mode. Using a model that provided steam cooling analysis, GE concluded that the CRD pumps did not provide adequate inventory to mitigate a feedwater line break and, therefore, GE agreed to add hardwired HPCF pump initiation control and flow indication as diverse backup capability. This was the last remaining area of disagreement between the staff and GE, and this issue is now resolved. GE has provided an Appendix C to SSAR Chapter 7 which includes the above commitments to provide adequate defense against potential common-mode failures. Appendix C addresses the issues identified above and specifies the diversity and set of equipment that will not be subject to potential common-mode failures. Therefore, DFSER Open Items 7.2.6-1, 7.2.6-2, and 7.2.6-3 are resolved and the design meets the staff's proposed applicable regulation for digital instrumentation and control systems.

REACTOR VESSEL WATER LEVEL INSTRUMENTATION

One issue associated with defense-in-depth and the diversity of design required to provide adequate protection against common-mode failures remains open. This issue concerns the RPV water level measurement. The issue was DFSER Open Item 20.3.8. The primary issue concerns the use of identical measurement techniques using condensing chambers and differential pressure transmitters to measure the water level and provide signals to I&C logic. Resolution of this item is discussed in Chapter 20 of this report.

7.2.7 Setpoints

The RPS setpoints will be listed in the RPS TS. The actual setpoints will not be established at this time because the design has not been completed, and the equipment has not been selected. The COL applicant will provide the specific setpoints based on the I&C system design and equipment prior to fuel load. The general requirement is that the setpoints be established high enough to preclude inadvertent actuation, but low enough to assure that proper margin is maintained in the setpoint determination. The TS (SSAR Chapter 16) have been submitted for review and are addressed in Section 16 of this report. Additional TS discussion is also included in Section 7.11 of this report.

GE has committed in the SSAR to meet the guidelines of RG 1.105, which govern instrument setpoints. RG 1.105 endorses ISA S67.04-1982, "Setpoints for Nuclear Safety-Related Instrumentation Used In Power Plants," with some exceptions. This is an acceptable commitment since ISA S67.04 defines a structured analysis acceptable to the staff to determine specific setpoints that adequately considers inaccuracies. This standard is currently undergoing revision and is expected to be issued in the near future. This is an example of a SSAR commitment which may be changed at some later time. The ITAAC for setpoints (which are included in the I&C CDM) requires a plant specific setpoint analysis which details the procedure for establishing specific setpoints. This plant specific analysis will be audited by the NRC during ITAAC implementation. Any changes to the setpoint commitments in the SSAR would involve an unreviewed safety question and, therefore, require NRC review and acceptance prior to implementation. Any requested changes to this commitment shall either be specifically described in the

COL application or submitted for license amendment after COL issuance.

7.2.8 Hardware and Software Qualification

The SSLC will be qualified as a Class 1E system. All SSLC components associated with protection systems are Class 1E and will be qualified to the same standards as the protection systems. All programmable digital equipment used for safety-related functions will be qualified in accordance with the safety system design basis with which they interface. This includes environmental and seismic qualification. The SSAR describes commitments to the SRP qualification criteria. The staff reviewed the commitments and has concluded that, for the topics that are addressed by the SRP, the commitments are adequate. The staff identified three additional areas of qualification that are not fully addressed in the SRP criteria. These issues are discussed in SSAR Appendix 7A, and are software qualification, electro-magnetic susceptibility, and mild environmental qualification. These issues were identified in the DSER as open issues.

The ABWR SSLC is dependent upon the proper functioning of the software to perform its safety functions. The standard for software which has been formally endorsed by the staff to specifically address software qualification is ANSI/IEEE ANS-7-4.3.2 (1982), "Application Criteria for Programmable Digital Computer Systems in Safety Systems of Nuclear Power Generating Stations," which was endorsed by RG 1.152 in 1985. The ABWR SSAR stated that, software development will, in general follow this standard. The staff considered this statement unclear and requested GE to provide a clear commitment to the software development process to be followed for the ABWR I&C system. This issue is also related to the GDC 1 issue previously discussed in this report concerning commitments to industry standards, and the issue concerning level of detail identified in the DSER. Because the software products and the software development plans for the SSLC have not been developed yet, there was no method available to the staff to independently verify that the software was in conformance with ANSI/IEEE ANS 7-4.3.2.

ANSI/IEEE ANS 7-4.3.2 is high level description of a software V&V process. Since it was published in 1982 there have been several other standards issued which provide additional guidance for V&V. In addition, there are other aspects to software qualification in addition to V&V. ASNI/IEEE ANS 7-4.3.2 has undergone significant revision and was reissued in November 1993.

To resolve the above issue on software qualification, GE has committed in the SSAR to software development, documentation and verification, in accordance with the 10 CFR Part 50, Appendix B requirements for quality assurance for safety-related systems and the following standards:

1. ASME NQA2a, Part 2.7, "Quality Assurance Requirements of Computer Software for Nuclear Facility Applications."

2. ANSI/IEEE ANS 7-4.3.2-1982, "Application Criteria for Digital Computers in Safety Systems for Nuclear Facilities." GE has also committed to the revised version of this standard.

3. IEEE 730-1984, "IEEE Standard for Software Quality Assurance Plans."

4. IEEE 828-1983, "IEEE Standard for Software Configuration Management Plans."

5. IEEE 829-1983, "IEEE Standard for Software Test Documentation."

6. IEEE 830-1984, "IEEE Standard for Software Requirements Specifications."

7. IEC 880-1986, "Software for Computers in the Safety Systems of Nuclear Power Stations."

8. IEEE 1012-1986, "IEEE Standard for Software Verification and Validation Plans."

9. IEEE 1033-1985, "IEEE Recommended Practice of Application of IEEE Standard 828 to Nuclear Power Generation Stations."

10. IEEE 1228 (draft), "Standard for Software Safety Plans."

11. IEEE 1042-1987, "Guide to Software Configuration Management."

12. Standard 2167A-1988, "Defense System Software Development."

GE has provided a discussion of software development in Table 7B.1 of the SSAR. In addition to a description of the process, and the commitment to standards, GE has added the following caveat:

> Note that the documents listed above may differ regarding specific methods and criteria applicable to the SMP. In situations where such differences exist, all of the methods and criteria presented within those documents are considered to be equally

appropriate and valid and, therefore, any of the above listed documents may be selected as the basis for elements of the SMP.

The staff finds the list of standards acceptable. The software development for the ABWR is included in the I&C CDM. The CDM outlines a software development plan which follows the guidance of the standards included in the SSAR. The standards listed in the SSAR are expected to change over the lifetime of the design certification. The staff expects the COL applicant to use the current version of standards which are available when the software implementation is started. Section 7.1 of this report provides a discussion of the approach to implementation of the ITAAC process for the I&C systems. The staff considers the first step in the staged ITAAC involving development of software design plans to be critical to successful completion of the ABWR I&C system because of the rapidly changing technology in digital I&C systems, and the many different vendors that may be involved in implementation of the ABWR design.

GE has committed to several additional software development methods and features as discussed below. GE committed to use formal methods for the SSLC when formal methods are developed sufficiently. Formal methods are typically defined as entailing a mathematical or occasionally a graphical software specification rather than natural language specification. The outputs would be able to be verified mathematically. Use of formal methods is in a relatively early stage of development at this time. The staff agrees with GE that it is premature to specifically commit to a particular formal method in the SSAR and, therefore, the staff finds the commitment by GE to use the accepted industry software design practices at the time that the software is developed to be appropriate.

GE has committed to provide a safety and hazards analysis, a sneak circuit analysis, and a timing analysis for the digital I&C systems. GE was requested to provide a description of the specifics of the program in these areas. This was DFSER Open Item 7.2.8-1. GE has provided additional details on these items in the CDM and the SSAR, and this item is, therefore, resolved.

GE has committed to the use of software metrics to track error rates during software development. GE has not specified a particular software metric as this will probably be selected by the final software vendor, and is an area that may change over the lifetime of the design certification. This was DFSER Confirmatory Item 7.2.8-1. The SSAR and the CDM have been revised to include the selection of software metrics during the development process. This item is, therefore, resolved. The staff also identified COL Action Item 7.2.8-1 for this issue since a possible resolution considered when the DFSER was written was for the COL applicant to select the software metrics. The selection of software metrics is now included in the CDM, SSAR and NRC audit of the staged ITAAC. Therefore, this COL action item is no longer applicable and is resolved.

GE stated that "proven technology" will be employed in the design and development of the ABWR I&C systems. The staff requested GE to clarify what was meant by proven technology. In response, GE stated that the definition of proven technology provided in the EPRI RD was appropriate for the ABWR. This EPRI requirement states that three years of successful experience in an application (nuclear or non-nuclear) very similar to the nuclear power plant application is adequate to demonstrate proven technology. The staff agrees with this definition and finds this acceptable. The stated goal, to which both the staff and GE agrees, is to use the best available technology without using unproven designs.

GE has committed to simple modular software programs that follow the guidance of DOD-STD-2167. The staff agrees that safety systems should have simple modular programs and finds this acceptable. GE has not specifically committed to follow DOD-STD-2167 for any other design requirements, though there are similar requirements for a structured design process between DOD-STD-2167 and the GE ABWR SSAR commitments.

At the time of issuance of the DFSER, the issue of commercial dedication of software for use in safety systems had not been adequately addressed by GE. GE subsequently made several commitments regarding commercial dedication of software, and the specific wording to be included in the SSAR. These commitments were identified as DFSER Confirmatory Item 7.2.8-2.

The first aspect of the commercial dedication issue is the use of well- developed operating systems in the development of a plant specific digital system, such as the SSLC. The staff agrees with GE that it is not necessary for the SSLC developer to perform a formal V&V of the operating system. However, it is essential that the SSLC developer assure that the operating system was developed under strict guidelines and has the quality necessary for a safety system.

The second aspect of the commercial dedication issue is the use of a complete component, such as a programmable logic controller, where most of the software has been developed prior to the decision to use it in a nuclear application. As with the operating systems described above, it is necessary for the developer to verify that the equipment selected is of sufficiently high quality for use in a safety system. It is not necessary for the final developer to repeat the V&V activities, but it is necessary for the developer to verify that the original equipment designer has followed equivalent criteria. This concern was resolved with SSAR commitments which were identified as DFSER Confirmatory Item 7.2.8-2.

Included in the commercial dedication issue is the qualification of the automated tools and design support software. It is necessary for the I&C system developer to verify that the tools are accurate. The staff expects the developer to verify the quality of the tools used in the design.

Also related to the issue of commercial dedication is the staff concern regarding communication by the suppliers of errors discovered in the suppliers' tools or software to the end user. This is similar to the 10 CFR Part 21 defect reporting required for Class 1E vendors. This was identified as DFSER Confirmatory Item 7.2.8-3. The CDM and the SSAR have been revised to include the selection criteria for commercial software, accuracy of tools, and notification of the end user by the developer of changes. Therefore, DFSER Confirmatory Items 7.2.8-2 and 7.2.8-3 are resolved.

Any changes to the hardware and software development commitments described in this section of this report would involve an unreviewed safety question and, therefore, require NRC review and acceptance prior to implementation. Any requested changes to this commitment shall be either specifically described in the COL application or submitted for license amendment after COL issuance.

The MPL documents do not add substantially to the information concerning hardware and software qualification. MPL C71-4010 (Rev. 0, May 18, 1990), "Reactor Protection System, Hardware/Software System Specification," lists the inputs and outputs for the DTM, TLU, and OLU but does not provide any additional hardware or software descriptions or specifications.

The second area of digital system qualification not addressed in the SRP criteria and which was an open issue in the DSER, concerns the qualification of the RPS and the other digital I&C equipment for the electromagnetic environment to which it will be exposed. This issue includes EMI, surge withstand capability, electrostatic discharge (ESD), radio frequency interference, and EMC. GE noted in the SSAR that one of the effective means of protection against EMI effects is the redundancy and separation of the divisions of the SSLC. GE committed in the SSAR to the following standards for electromagnetic environmental considerations:

1. ANSI IEEE C63.12-1987, "American National Standard for Electromagnetic Comparability Limits - Recommended Practice."

2. ANSI/IEEE C37.90.2-1987, "IEEE Trial - Use Standard, Withstand Capability of Relay Systems to Radiated Electromagnetic Interference from Transceivers."

3. ANSI/IEEE C62.41-1980, "Guide for Surge Voltages in Low-Voltage AC Power Circuits."

4. ANSI/IEEE C62.45-1987, "Guide on Surge Testing for Equipment Connected to Low-Voltage AC Power Circuits."

5. MIL-STD 461C-1987, "Electromagnetic Emission and Susceptibility Requirements for the Control of Electromagnetic Interference."

6. MIL-STD 462-1987, "Measurement of Electromagnetic Interference Characteristics."

7. IEC 801-2, "Electromagnetic Comparability for Industrial-Process Measurement and Control Equipment, Part 2: Electrostatic Discharge Requirements."

8. IEEE 518-1982, "Guide for the Installation of Electrical Equipment to Minimize Electrical Noise Inputs to Controllers from External Sources."

Commitment to the above standards resolved the DSER open issue. The standards listed above require the selection of specific test categories. Verification of the appropriate selection will be performed during the ITAAC implementation. The selection at the planning stages includes consideration of installation techniques (such as indicated in IEEE-1050 for shielding and grounding), and verification at the site that the installed condition is enveloped by the qualification testing.

EMI protection is included in the I&Cs CDM. One specific feature that the staff requested be included in the CDM is that the digital equipment be tested for the low range of the EMI spectrum as well as the mid to upper ranges. This was DFSER Confirmatory Item 7.2.8-4. The CDM was revised to include the selection of the EMI testing ranges when the equipment has been selected. This is acceptable to the staff and, therefore, this item is resolved. Any changes to these EMI commitments would involve an unreviewed safety question and, therefore, require NRC review and acceptance prior to implementation. Any requested changes to this commitment shall either be specifically described in the COL application or submitted for license amendment after COL issuance.

The third area of digital system qualification concerns the qualification of the SSLC equipment for the mild environment temperature profiles to which the equipment could be subjected during plant operation. One issue in particular concerned the possibility of local hot spots as a result of higher current densities when using digital chip designs. GE committed to qualification of the SSLC internal panel components to a temperature rise of 15 °C (27 °F) above the normal ambient operating conditions. The electronic equipment panel cooling to maintain qualification is achieved by natural convection of the room air in the panels. Fans may be used to improve long term reliability of electronic equipment, but no credit is taken for forced air circulation for thermal qualification purposes. MPL DMH-4270 (Rev. 2, February 3, 1989), "Essential Multiplexing System Design Specification," lists the CMU and RMU environmental qualification requirements as 10 - 40 °C (50 - 104 °F) temperature, 10 - 60 percent relative humidity, and seismic Category I.

GE stated in the SSAR that the SSLC will be constructed from electronic components purchased to military specifications to the extent practical, and the components will be qualified by testing to higher temperatures than specified in the SSAR for a given room environment. The staff agrees that it is desirable to have this additional margin built into the design. This was DFSER Confirmatory Item 7.2.8-5. The SSAR has been revised to include this information and, therefore, this item is resolved.

Based on the above evaluation of the RPS qualification, the staff concludes that the commitments in the SSAR meet the requirements of IEEE-279 for equipment qualification and quality of components and modules.

7.2.9 RPS Findings and Conclusions

The design description of the RPS in the SSAR was evaluated to confirm commitments to the SRP and the applicable regulatory guides and industry codes and standards. This review was concerned with the trip parameter sensors, EMS, SSLC and the protection actuation circuits. Based on staff review of the information provided for the sensor and protection actuation circuits, the staff concludes that the SSAR provides acceptable commitments to the appropriate SRP criteria.

The RPS includes systems and components that GE has committed to be designed to survive the effects of earthquakes, other natural phenomena, abnormal environments and missiles. The staff, therefore, concludes that the GE commitments meet the requirements of GDC 2, "Design Bases for Protection Against Natural Phenomena," and GDC 4, "Environmental and Missile Design Bases," for the RPS.

Based on the review, the staff concludes that the design and the design process for the RPS as described by GE in the SSAR and CDM provides instrumentation to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems. It appears that appropriate controls have ben provided to maintain the variables and systems within prescribed operating ranges. Therefore, the staff finds that the RPS design satisfies the requirements of GDC 13.

The staff also concludes, as discussed in Section 7.2.1 of this report, that the GE commitments to the requirements of 10 CFR 50.55a(h) (IEEE-279) and the requirements of GDC 20, "Protection System Functions," for the functional requirements of the RPS are acceptable.

The staff concludes that periodic testing of the RPS as described in the SSAR, and as discussed in Section 7.2.1 of this report, conforms with the criteria of RG 1.22 and IEEE Standard 338 as supplemented by RG 1.118 and is, therefore, acceptable. The staff further concludes that GE commitments to IEEE-279 with regard to system reliability and testability are consistent with the requirements of GDC 21, "Protection System Reliability and Testability," and are acceptable.

The staff concludes, as discussed in Section 7.2.1 of this report, that the RPS as defined by GE meets the criteria of IEEE-384 as supplemented by RG 1.75 for protection system independence with the exceptions noted in SSAR Table 1.8-7, "Summary of Differences from SRP Section 7," and discussed in the SSAR. The staff finds the identified exceptions acceptable, and therefore, the staff concludes that the RPS meets the requirements of GDC 22, "Protection System Independence."

Based on the staff review of results of the FMEA of the RPS in conjunction with the results of the studies of the RPS design for defense against common-mode failures, the staff concludes, as discussed in Section 7.2.1 of this report, that the RPS design adequately meets the requirements of GDC 23, "Protection System Failure Modes."

Based on the review of the reactor protection system design, the staff finds that the system is designed to meet the requirements of IEEE-279 regarding control and protection system interaction meets the requirements of GDC 24, "Separation of Protection and Control Systems."

Based on the review of the reactor protection system, the staff concludes that the system satisfies the protection system requirements for malfunctions of the reactivity control system such as accidental withdrawal of control rods. Therefore, the staff finds that the RPS satisfies the requirements of GDC 25.

Based on the review of the protection and reactivity control systems, the staff finds that these systems are designed to assure a high probability of accomplishing their safety functions in the event of anticipated operational occurrences. The staff, therefore, concludes that the design meets the requirements of GDC 29.

The staff review also included consideration of the RPS quality and diversity, as discussed previously in Section 7.2.1 of this report. Based on this review, the staff concludes that GE has specified the appropriate quality requirements, and has provided adequate defense-in-depth and diversity for postulated common-mode failures. The staff has determined that changes to commitments involving (1) computer hardware and software development and quality standards, (2) essential multiplexor design and standards criteria, (3) electromagnetic environment protection criteria and standards, (4) design features and commitments of the SSLC self-test system, and (5) setpoint methodology, would involve an unreviewed safety question and, therefore, require NRC review and acceptance prior to implementation. Any requested changes to these commitments shall either be specifically described in the COL application or submitted for license amendment after COL issuance.

The staff also concludes that the staged audit approach of the ITAAC implementation discussed in Section 7.1.3.3 of this report is an important aspect in the final acceptance of the RPS.

Based on the above discussions and findings, the staff concludes that the design and the design process of the RPS as described by GE in the SSAR and CDM meets the requirements of GDC 2, 4, 13, 20, 21, 22, 23, 24, 25, and 29 and 10 CFR 50.55a(h) (IEEE-279). The RPS design is, therefore, acceptable.

7.3 Engineered Safety Features Systems

7.3.1 System Description

This section describes the I&Cs for equipment in the various ESF systems. ESF system descriptions are provided in SSAR Chapter 6. The ESF systems for the ABWR utilize the SSLC system which is shared with the RPS. Figures 7.3-1 and 7.3-2 of this report provide an overview of the ESF implementation with the SSLC. Section 7.2 of this report discusses the components of the SSLC. Differences between the RPS design implementation and the ESF design are discussed in the following section.

This section describes application to the ESF system of the design bases information identified in IEEE-279, and the various new technology criteria identified by RG 1.152, NUREG-0493 and the ABWR CDM. The additional RPS criteria discussed in Section 7.2 for software, EMI and mild environment qualification also apply to the ESF systems.

The ESF systems are:

(1) emergency core cooling systems
(2) LDS
(3) wetwell and drywell spray mode of RHR
(4) suppression pool cooling mode of RHR
(5) SGTS
(6) emergency diesel generator support systems
(7) reactor building cooling water system
(8) essential HVAC emergency cooling water system (HECW)
(9) high-pressure nitrogen gas supply system

The systems that provide the ESF functions for the ABWR are similar to those of operating BWR designs previously reviewed by the staff. As with the RPS, the primary differences are in the method of implementation of the I&C aspects of the design. The focus of this review was on the use of the EMS and the SSLC system in place of the relay, solid state logic, and copper wire cable of previous I&C system designs.

7.3.1.1 Emergency Core Cooling Systems

The ECCS consists of the HPCF system, the automatic depressurization (ADS), including the safety/relief valve (SRV) electrical actuation logic system, RCIC system, and low-pressure flooder (LPFL) mode of the RHR system. The ECCS I&C (ESF actuation) systems sense the need for ECCS action and initiate appropriate equipment as required. Though the ESF systems share the SSLC and EMS with the RPS, the ESF systems are described in the

SSAR and CDM as separate I&C systems which use the SSLC and EMS.

7.3.1.1.1 High-Pressure Core Flooder System

Four reactor water level and drywell pressure transmitters (one for each division) provide inputs to the SSLC logic, either of which logic signal activates the HPCF system upon a two-out-of-four coincidence of the parameter setpoint. The logic arrangement permits on-line testing of the electronics. After activation, the two HPCF pumps reach rated flow within 36 seconds. The HPCF pumps automatically stop on reactor high water level, or can be manually shutdown. The HPCF pumps are interlocked to prevent starting if an open suction path from the condensate storage tank or suppression pool is not available.

Separation within the HPCF actuation logic system is such that no single failure can prevent system actuation. The logic system is also designed so that no single failure results in a spurious actuation. While the initiating logic is a four-channel system, the actuated system is a two-channel system. The HPCF control is powered by Divisions II and III of the SSLC as shown in Figure 7.3-2 of this report. Divisional separation is maintained between the four sensor inputs (Divisions I, II, III, and IV sensor channels) to the logics, the Divisions II and III logic channels, and the Divisions II and III output controls (output channels).

7.3.1.1.2 Automatic Depressurization System

The MSL inside the drywell have a total of 18 SRV which discharge to the suppression pool. Eight of these valves are designated for use as the ADS. ADS consists of redundant trip channels in two separate logics that control two separate solenoid-operated pilot valves on each ADS valve. Either pilot valve can operate its associated ADS valve. ADS initiation signal is either reactor low water level (L1) and high drywell pressure or a sustained reactor low water level (sustained for 8 minutes). Both parameter setpoints must be reached before ADS is initiated. The valves are interlocked with the HPCF and RHR pump discharge pressure sensors to assure that an HPCF or RHR pump is running prior to depressurization. There is also a time delay (29 seconds) between the completion of the logic voting and the initiation signal. This time delay allows the HPCF or RCIC systems to restore reactor vessel water level if they are available before depressurizing the reactor by actuating ADS. Manual initiation of ADS is also available.

Sensors provide inputs to the four-division SSLC. The ADS actuation output signal to the valves is via Divisions I and II output channels (electrical divisions) of the SSLC. The SSLC uses eight reactor vessel water level sensors; four different sensors for Divisions I and II. Both electrical divisions are routed to each of the eight ADS valves. The ADS Division I actuation output energizes a solenoid pilot valve on each ADS valve. Similarly, the ADS Division II actuation output energizes the second pilot valve on each ADS valve. Actuation of either solenoid pilot valve opens the ADS valve.

7.3.1.1.3 Reactor Core Isolation Cooling

The RCIC system is a high-pressure injection system which uses a steam turbine-driven pump. The actuated equipment is a single train system. This system is initiated when either high drywell pressure or low reactor water level (L2) setpoints are met. Each parameter has four sensors which provide input via the EMS to the SSLC. The output is via electrical Division I of the SSLC and the EMS.

The RCIC pump turbine is automatically shutdown on turbine overspeed, high turbine exhaust pressure, RCIC auto-isolation signal, low pump suction pressure, reactor water level high (L8), or manual trip if the initiation signal is not present. The RCIC system fails-as-is upon loss of power to the SSLC or loss of input signals. Automatic and manual isolation capability for the RCIC system is provided as part of the leak detection and isolation system.

The RCIC system itself is not redundant because the HPCF system provides functionally redundant capability. Some RCIC system valves are Division II components and appropriate separation of these devices and circuits from the Division I equipment is maintained. System tests are accomplished with a division-of-sensors bypass in place, as discussed in Section 7.2 of this report. On-line signal verification is accomplished by the SSLC. System status annunciation and performance indicators are provided in the control room.

7.3.1.1.4 RHR/Low-Pressure Flooder

The LPFL is an operating mode of the RHR system which is designed to provide water to the reactor vessel following a design basis LOCA. The LPFL is initiated automatically on reactor low water level (L1) signals from the eight water level transmitters of the NBS. The LPFL injection valve actuation logic requires a reactor low-pressure permissive for automatic actuation. These transmitters are separated into four divisions, as discussed in Section 7.2 of this report. Four transmitters provide signals (one from each division) to RHR Divisions I and III, while the other four supply similar signals to RHR Division II. The LPFL system is also initiated on high drywell pressure as sensed by four transmitters (one from each division) of the NBS. The signals from each parameter are combined, through fiber optic isolators, in two-out-of four logic for each division of LPFL in order to meet the single failure criterion. The RHR/LPFL flow paths are redundant with functional diversity provided by the HPCF and RCIC systems. The SSLC incorporates automatic testing of the instrumentation and verification of the output signals. System status annunciation and performance indicators are provided in the control room. The equipment will be environmentally qualified for the location in which it is to be installed.

7.3.1.2 Leak Detection and Isolation System

The LDS I&C consist of temperature, pressure, radiation, and flow sensors to detect, indicate, and alarm leakage from the reactor primary pressure boundary, and, in certain cases, also close isolation valves to shut off leakage outside the containment. Manual control is provided in the control room for system level isolation of leakage. Each power-operated isolation valve is also provided with a separate manual control switch in the control room independent of the automatic and system level manual logic. All LDS isolation valves are actuated with de-energize to isolate logic.

The LDS system has several isolation capabilities. Containment isolation is initiated on high drywell pressure, low reactor water level (L1, L1.5, L2, L3), manual operator action, and high radiation from the PRM system. Direct operator action, via manual logic reset control, is required to reset the trip condition, provided the initiating signal is cleared. Reactor water cleanup system isolation is initiated on high differential flow and high equipment area temperature. RHR system shutdown cooling suction lines are isolated on low reactor water level (L3) and high ambient temperature. RCIC is isolated on high ambient temperature and high turbine exhaust pressure. The MSL is isolated on low reactor water level (L1.5), high MSL differential pressure, high MSL radiation, high MSL tunnel ambient temperature, high MSL tunnel area temperature in the turbine building, low main condenser vacuum, and low MSL pressure.

7.3.1.3 RHR/Wetwell and Drywell Spray Cooling

The wetwell and drywell spray cooling is an operational mode of the RHR system. This mode uses RHR pumps B and C. The wetwell and drywell spray cooling is manually initiated from the control room, with drywell pressure providing permissive interlocks from NBS sensors and EMS and SSLC system functions for the drywell cooling mode. The sensor circuits and logic are provided with separation, redundancy and testability consistent with other ESF circuits described in this section and in Section 7.2 of this report. The manual initiating sequence begins with an LPFL initiation signal (low reactor water level). If high drywell pressure and/or high wetwell pressure is also present, the operator will manually close the reactor injection valves, and manually open the spray valves. If low water level occurs again, the system will automatically realign to the reactor injection mode.

7.3.1.4 RHR/Suppression Pool Cooling Mode

The suppression pool cooling mode of RHR uses the same I&C as the LPFL previously described. Redundancy is provided by three separate logic divisions. However, unlike the LPFL, no functional or equipment diversity is identified. This system is automatically initiated upon receipt of a high temperature signal from the SPTM system. The suppression pool cooling mode is also initiated manually. Annunciators and indicators of the RHR systems' operation status in the suppression pool cooling mode are non-safety and available in the control room.

7.3.1.5 Standby Gas Treatment System

The SGTS is initiated automatically upon the detection of high drywell pressure, low reactor water level, high radiation in the fuel handling area or secondary containment HVAC exhaust air. Manual initiation is also available. Two logic divisions are powered from separate ESF buses. The SGTS I&C are supplied power from the Divisions II and III emergency power supplies. Both electrical isolation and physical separation of the divisions are maintained. The system electronics are tested by signal insertion. The SSAR states that SGTS electrical equipment is conformed to the environmental conditions for the area in which it is to be installed. Non-safety-related system status indicators and annunciators are provided in the control room.

7.3.1.6 Emergency Diesel Generator Support Systems

The three emergency diesel generators provide power to and are controlled by Divisions I, II, and III of the Class 1E power supplies. The EDG support systems are described in SSAR Chapter 9 and the associated section of this report. The support systems include the jacket water system, the starting air system, the lube oil system, and the fuel transfer system. Though not specifically mentioned in the SSAR, the support systems also include the EDG HVAC system. The I&C for the support systems are designed to the same criteria as the primary system.

7.3.1.7 Reactor Building Cooling Water System

The I&C system for the reactor building cooling water system consists of two-out-of-four logic with system initiation occurring on low reactor water level or high drywell pressure signals. The instrumentation system's output to the actuated equipment is separated into three divisions such that no single failure can disable this system. Annunciators and indicators of system status are non-safety-related.

7.3.1.8 Essential HVAC Emergency Cooling Water System

The HECW system supplies demineralized chilled water to the cooling coils of the control building safety-related electrical equipment rooms and main control room coolers, and the diesel generator zone air conditioning systems. The HECW system is composed of three divisions, each of two divisions containing two 50-percent capacity refrigerators and chilled water pumps and one division containing one refrigerator and chilled water pump. The systems' I&C output to the actuated equipment are supplied from Divisions I, II, and III power buses.

The HECW system divisions are mechanically and electrically separate. The system is designed to operate during both accident conditions and normal plant operation and during all modes of operation for the control building and diesel generator zone cooling systems. The HECW system operation is initiated automatically when the controls in the main control room are set for automatic operation and any of the HVAC systems located in the control building or diesel generator areas are started. The HECW system can also be started manually from the control room. The HECW system I&C will be tested manually.

The HECW system I&C equipment will be qualified for the particular environment in the area in which it is located as described in Section 7.2 of this report. The environmental qualification of the electrical equipment is also verified via the ITAAC. DFSER COL Item 7.3.1.11-1 discussed the testing and temperature verification that the COL applicant is to include in its pre-operational test procedures in order to confirm electrical equipment environmental qualification. GE revised SSAR Section 7.3.3.1 to add the requirement that the COL applicant include temperature profiles for racks containing Class 1E microprocessor equipment (for various loss of HVAC conditions) in the pre-operational test procedures. This is acceptable to the staff. The COL applicant items associated with I&C equipment cooling are discussed in Section 7.8 of this report.

7.3.1.9 High-Pressure Nitrogen Gas Supply System

The high-pressure nitrogen gas supply system provides compressed nitrogen to the ADS SRV, the MSIVs (for testing only), and other instruments and valves. This system supports both safety- and non-safety-related portions of the plant. The safety-related portion of the system consists of two redundant banks of high-pressure nitrogen bottles and associated piping, valves, and controls powered from separate essential power supplies (Divisions I and II). Upon detection of low nitrogen pressure to the ADS accumulators, this system will automatically isolate the safety-related portion of the system from the non-safety-related portion by isolation valves which automatically terminate the normal nitrogen supply and open the emergency nitrogen gas bottle supply to the ADS accumulators.

7.3.2 Safety System Logic and Control and Specific Subsystem Descriptions

The SSLC is discussed in detail in Section 7.2 of this report. This section discusses the differences between the RPS and ESF portions of the SSLC. The primary difference is the use of safety SLUs in place of the RPS TLUs. As shown in Figure 7.3-2, there are a total of 12 SLUs with four in each of the three SSLC divisions. The SLUs are contained in the portion of the SSLC designated as the logic channel in the TS. The actuated ESF equipment controlled from the SLUs may be in single, redundant or triplicated system trains as described in the system descriptions above. The ESF equipment is divided between the SLUs 1&2 and SLUs 3&4 logic functions to provide some diversity in case of a failure. For example, the HPCF and RCIC high-pressure reactor injection functions are provided by different SLUs than the RHR low- pressure reactor makeup function. The full distribution of systems on the various output channels is presented in the MPL design specifications.

The SLU architecture is arranged so that each ESF logic function has two SLUs performing the same function. Both SLUs (1&2 or 3&4) receive the same input from the DTM, manual controls and bypasses, and in some cases they receive the same direct sensor inputs for interlock protection. The logic in both SLUs must agree before the initiation signal is processed via the output channel to the actuated equipment. This two-out-of-two voting arrangement occurs at the remote multiplexing unit. A single failure of an SLU or EMS channel (one of the two links within an electrical division) will not initiate an ESF function. With the exception of the containment isolation signals which are fail-safe, the logic for the ESF systems is designed to a fail-as-is condition.

MPL A32-4080 (Rev. 0, April 25, 1990), "EMS/SSLC Interface Requirements, Requirements Specification," Figure 1, "EMS/SSLC Interface Block Diagram," uses the term "Auxiliary Supporting Features Logic" (ALU) for part of the SSLC while the SSAR and the CDM use the term "Safety System Logic Unit" for the same device. This inconsistency was identified as DFSER Confirmatory Item 7.3.2-1. GE committed to revise the MPLs which use the ALU terminology to be consistent with the SLU terminology as used in the CDM and SSAR. The ITAAC will verify that these documents are consistent when the design is implemented. Therefore, this item is resolved.

7.3.3 Findings and Conclusions

The general ESF design and arrangements are in accordance with the requirements of the SRP to the extent information was available for review. Many of the issues that are identified in Section 7.2 concerning the RPS also apply to the ESF systems. The level of detail for the ESF systems was an open issue in the DSER. The staff found the level of detail available for review inadequate in the DFSER. However, because these systems will be included in the digital I&C design development and ITAAC process, the DSER open issue is resolved. The potential for common-mode software problems may also exist with the ESF systems. This was an open issue in the DFSER. The issue of defense against common-mode failures was resolved as discussed in Section 7.2 of this report.

GE had not provided a detailed FMEA for I&C of the ESF systems as required to demonstrate conformance with the requirements of IEEE-279 and the guidelines of NUREG-0493 regarding defense-in-depth analysis. This was identified as an open issue in the DSER and was part of the common-mode failure discussion (DFSER Open Item 7.2.6-1). This item is resolved as discussed in Section 7.2 of this report.

The design description of the ESF I&C systems in the SSAR was evaluated to confirm commitments to the SRP and the applicable regulatory guides and industry codes and standards. This review was concerned with the EMS, SSLC, and the ESF system initiation and actuation circuits. Based on staff review of the information provided for the initiation and actuation circuits, the staff concludes that the SSAR provides acceptable commitments to the appropriate SRP criteria.

The ESF actuation system includes systems and components that GE has committed to be designed to survive the effects of earthquakes, other natural phenomena, abnormal environments, and missiles. The buildings containing ESF systems and components will be designed to meet and withstand the probable maximum flood at the site, and meteorological events. The structures containing the ESF components and the ESF instrumentation and electrical equipment will be seismically qualified. To protect the ESF systems in the event of a postulated fire, the redundant portions of the systems will be separated by fire barriers. The ESF system instrument taps and sensing lines located inside the drywell will be qualified to remain functional during and following a LOCA. The staff, therefore, concludes that the GE commitments meet the requirements of GDC 2, "Design Bases for Protection Against Natural Phenomena," and GDC 4, "Environmental and Missile Design Bases," for the ESF systems.

As discussed above, in Sections 7.3.1 and 7.3.2 of this report, GE has committed that all components of the ESF systems are qualified for the environments in which they are located. Separation and isolation will be preserved, both mechanically and electrically, in accordance with IEEE-279 and RG 1.75. Commitments to other requirements of IEEE-279, such as testing, bypasses, and manual initiation, and corresponding provisions in the design are also described in the SSAR and discussed above. The staff concludes that the GE commitments to the design basis requirements of IEEE-279 and the requirements of GDC 20, "Protection System Functions," for the functional requirements of the ESF actuation systems are acceptable.

In conjunction with SSLC discussed in Section 7.2 of this report, ESF system logic and component testing capabilities will be provided to fully test ESF systems during reactor operation. The staff concludes that periodic testing of the ESF I&C system as described in the SSAR conforms with the criteria of RG 1.22 and IEEE-338 as supplemented by RG 1.118 and is, therefore, acceptable. The staff further concludes that GE commitments to IEEE-279 with regard to system reliability and testability are consistent with the requirements of GDC 21, "Protection System Reliability and Testability," and are acceptable.

Divisional separation of sensor inputs and output channels that will be provided in the design of various ESF systems are discussed above. The evaluation of the SSLC regarding channel separation and electrical isolation is discussed in Section 7.2 of this report. The staff concludes that the ESF actuation systems as defined by GE meet the criteria of IEEE-384 as supplemented by RG 1.75 for protection system independence with the exceptions noted in SSAR Table 1.8-7, "Summary of Differences from SRP Section 7," and discussed in the SSAR. The staff finds the identified exceptions acceptable, and the staff concludes that the ESF actuation systems meet the requirements of GDC 22, "Protection System Independence."

Based on the staff's review of results of the FMEA of the ESF I&C systems in conjunction with the results of the studies of the digital I&C system design for defense against common-mode failures, the staff concludes that the design of the I&C of ESF systems adequately meets the requirements of GDC 23, "Protection System Failure Modes." The ESF systems and components fail in the as-is position in that they require power to operate (i.e., energize to operate). Electrical power is required to perform the emergency functions of the ESF systems and components. The redundancy provided in the design of the ESF I&C, as discussed above, assure that no single failure can cause ESF system failure when required, or inadvertent initiation. The evaluation of defense against common-mode failure in the I&C systems is discussed in Section 7.2 of this report.

The I&C system for ESF has no control function of non-safety systems. However, it does provide isolation signals and inputs to non-safety portions of cooling systems, and annunciators and computers through appropriate isolation devices. Such circuits will be treated as associated circuits or non-Class 1E circuits. Associated circuits will be in accordance with Class 1E circuit requirements up to and including the isolation devices. Non-Class 1E circuits will be separated and isolated from Class 1E circuits or be treated as associated circuits. The staff also concludes that the GE commitments to the design basis requirements of IEEE-279 regarding control and protection system interaction meets the requirements of GDC 24, "Separation of Protection and Control Systems."

The staff's review also included consideration of the ESF system quality and diversity as discussed previously in Section 7.2 of this report. The staff also concludes that the staged audit approach of the ITAAC implementation discussed in Section 7.1 of this report is an important aspect in the final acceptance of the ESF I&C systems.

Based on the above discussions and findings, the staff concludes that the design and the design process of the ESF I&C systems as described by GE in the SSAR and CDM meets the relevant requirements of GDC 2, 4, 20, 21, 22, 23, and 24, and 10 CFR 50.55a(h) (IEEE-279). The design of the ESF I&C systems is, therefore, acceptable.

7.4 Systems Required for Safe Shutdown

7.4.1 System Description

The following systems are identified in the SSAR as required for safe shutdown of the reactor:

(1) alternate rod insertion function
(2) standby liquid control system
(3) reactor shutdown cooling mode of the RHR system
(4) remote shutdown system

This section provides a discussion of the I&C aspects of these systems, with a review of the interface effects between these systems and the RPS, ESF, EMS, and the SSLC. The four systems addressed in this section have a corresponding CDM section. The review of the CDM for these systems had not been completed at the time the DFSER was issued, and this was identified as DFSER Open Item 7.4.1-1. GE provided the CDM for these systems. The adequacy and acceptability of the CDM is evaluated in Section 14.3 of this report. On the basis of this evaluation, this item is resolved.

7.4.1.1 Alternate Rod Insertion System

The ARI function is accomplished by the rod control and information system (RC&IS), the reactor flow control (RFC) system, and the FMCRD. The ARI system provides the capability for automatic insertion of all rods by an alternate and diverse method from the RPS as necessary for mitigation of an anticipated transient without scram on receipt of high reactor dome pressure and low reactor water level (Level 2) signals. The RC&IS, including the portion for ARI actuation, is not classified as a safety-related system, but is single-failure proof and incorporates features in its design for high reliability and availability. (RC&IS is discussed further in SSAR Section 7.7 and Section 7.7 of this report.) The Level 2 low reactor vessel water level signal is provided via the SSLC (ESF portion) and, therefore, the sensors for this input are Class 1E. In the ARI SSAR description, GE did not indicate whether the RPV water Level 2 inputs to the SSLC are hardwired and, therefore, would not share common equipment with the RPS input. The SSAR needed to clearly state the design for this feature. This was DFSER Confirmatory Item 7.4.1.1-1. GE subsequently revised the SSAR to specify which SSLC signals are multiplexed and which are hardwired. Therefore, this item is resolved.

The requirement for a reactor shutdown system for operational transients is identified in 10 CFR 50.62, "Requirements for Reduction of Risk from Anticipated Transients Without Scram Events for All LWR Designs." Topical Report NEDE-31096-A was submitted by GE to address ATWS events for currently operating BWRs and was approved by the staff. GE indicated, in response to staff questions, their intent to fully conform to NEDE-31096-A for the ABWR. This was identified as DFSER Confirmatory Item 7.4.1.1-2. GE revised the SSAR to include a commitment to the topical report.

Therefore, this item is resolved. The staff concludes that the ARI design meets the requirements of 10 CFR 50.62.

The ARI sensor trip setpoints are set above the corresponding RPS settings to allow the RPS trip to occur first. The RPS reactor water level trip is at Level 3 compared to Level 2 for the ARI. The reactor dome pressure trip setpoint is higher than the RPS reactor pressure trip setpoint. Manual actuation of rod insertion requires two manual actions to be taken in order to avoid inadvertent actuation of the rod trips. The logic for the ARI function as described in the SSAR is designed such that no single failure results in the failure to insert more than one operable control rod when the ARI is actuated.

The RPV Level 2 signal from the SSLC is provided to each of the triple redundant recirculation flow control (RFC) system controllers where a two-out-of-four vote is taken in each controller. The reactor dome pressure signals from the steam bypass and pressure control (SB&PC) system are provided to each of the RFC controllers where a two-out-of-three vote is taken. Manual ARI initiation is provided to the same controllers where a two-out-of-two vote is taken. Any of these three RFC controller output signals will initiate the ARI function. The output (trip or no-trip) from each of the RFC controllers is provided as input to the redundant RC&IS controllers where a two-out-of-three vote is taken to initiate FMCRD run-in. (Scram-follow input signals are also provided to the RC&IS.) The output of the two channels of the RC&IS are provided, along with the output of the FMCRD emergency insertion logic channel, to the FMCRD inverter controllers where they are combined in a three-out-of-three vote logic to initiate rod insertion. The output from the RFC controllers also provides input to a two-out-of-three vote logic for the ARI function performed by the redundant scram air header exhaust valves. The ARI system also initiates a recirculation pump trip as described in Section 7.7 of this report.

7.4.1.2 Standby Liquid Control System

The SLCS I&C are designed to initiate the injection of liquid neutron absorber (borated water) into the reactor. The SLCS is a two-train system with one pump for each train. The I&C system is designed to withstand seismic Category I earthquake loads and I&C equipment is mounted in seismically qualified panels. Power for the I&C is provided from the Class 1E instrument bus. The system is designed to be highly reliable with many safety-related system features but it is not classified as a safety-related system. In response to Q420.125 (SSAR Chapter 20), GE stated that the SLCS is hardwired and does not interface with the EMS. Therefore, the SLCS I&C system does not share any components with the RPS.

Reactor pressure and SLCS borated water storage tank level sensing equipment are used to determine that the liquid neutron absorber is being pumped into the reactor. The system is capable of being tested while the plant is operational by using the SLCS pumps to inject demineralized water into the reactor vessel. Indications, controls and annunciators for SLCS status and control are located in the control room. SLCS status indicators are also provided at the SLCS local control panel to indicate system operating conditions.

The SLCS was initially described in the ABWR design as a manually-initiated system with no capability for automatic initiation. By letter dated June 2, 1993, (regarding important features identified by the ABWR PRA) GE stated that the SLCS would be automatically initiated in order to avoid the potential for operator error and further reduce the probability of adverse consequences due to an ATWS. This change is consistent with the requirements of 50.62(c)(4) for the SLCS automatic initiation. The two SLCS pumps and associated valves will be initiated upon an ATWS initiation signal derived from high RPV pressure and SRNM ATWS permissive for 3 minutes, or low RPV level and SRNM ATWS permissive for 3 minutes. If the control rods have been inserted by the RPS or ARI (automatically or manually) the APRM should indicate downscale before 3 minutes and, therefore, the SLCS would not initiate. The staff finds the SLCS design modification to be in conformance with the requirements of 50.62(c)(4), therefore, acceptable.

7.4.1.3 Reactor Shutdown Cooling Mode of the RHR System

The reactor shutdown cooling mode of the RHR system is initiated by manual operator action with interlocks on the RHR valves to ensure correct cooling mode alignment. The RHR reactor shutdown mode is entered during both normal and emergency shutdown. Normal shutdown is accomplished with all three of the RHR trains in operation and brings the reactor to approximately 51.7 °C (125 °F) within 20 hours following a reactor scram. Emergency shutdown operation brings the reactor to cold shutdown (less than 100 °C (212 °F)) with two RHR pumps operating within 36 hours after control rod insertion. The RHR equipment is Class 1E and redundant, and is seismically and environmentally qualified for the installed location. RHR system controls are located in the control room. The staff finds the reactor shutdown cooling mode I&C design acceptable.

7.4.1.4 Remote Shutdown System

The remote shutdown system provides a means to accomplish reactor shutdown functions (controls and indications) from outside the main control room and bring the reactor to cold shutdown. By letter dated June 2, 1993, (regarding important features identified by the ABWR PRA) GE identified the RSS as an important feature for reducing ABWR core damage frequency due to a control room fire.

The RSS does not include reactor scram capability or complete control of ESF systems. The RSS design assumes that the operator scrams the reactor from the main control room prior to going to the RSS, and that there is no coincident design-basis accident. Two divisionalized RSS panels are provided at separate locations. Their operation is administratively and procedurally controlled. The remote shutdown controls include manual transfer switches to transfer control functions from the control room to the RSS. In addition, on transfer of controls to the RSS, an alarm actuates in the control room.

The staff requested (Q420.15) additional clarification of the intended use of the RSS and the degree of isolation and independence of the RSS from the SSLC and EMS. In the response, GE stated that the RSS is totally separate and independent from the SSLC and EMS because it is hardwired from the sensors to the RSS panels and from the RSS panels to the actuated devices, and does not use multiplexed signal interfaces. Inclusion of this clarification into the SSAR was DFSER Confirmatory Item 7.4.1.4-1. GE revised the SSAR to incorporate the above clarification and, therefore, this item is resolved. The RSS capability is a consideration in the staff's resolution of the common-mode failure issue as discussed in Section 7.2.6 of this report.

The two RSS panels are powered by separate Divisions I and II Class 1E power. Equipment controlled from these panels is powered from the same divisions as when normally controlled from the SSLC. The RSS includes controls for one train of HPCF, two trains of RHR, two trains of RCW, two trains of RSW system, two trains of electrical power distribution system, and the flammability control system. Transfer switches that transfer controls from the control room for these trains of equipment and the emergency diesel generators, are located in the RSS. Indication is also provided to monitor shutdown functions.

7.4.2 Specific Findings

The ARI function and the SLCS instrumentation are part of the resolution of the issue of potential common-mode failure of the EMS and SSLC components. The analysis to resolve this issue also considered the function of the systems discussed above for mitigation of SSAR Chapter 15 events in combination with postulated common-mode failure in the safety-related digital I&C system, and how operation of these shutdown systems may need to be reconsidered. Specifically, the RSS operation may require reconsideration of the equipment required or the time available to the operator to achieve shutdown using the RSS. This was part of the I&C system common-mode failure analysis which was an open issue in the DSER (SECY-91-294) and DFSER Open Item 7.4.2-1. The use of the equipment described in this section to assist in mitigation of the consequences of a common-mode failure is addressed in Section 7.2 of this report. This open item is, therefore, resolved.

The SSAR did not initially describe how the transfer of sensor outputs from the control room to the RSS would occur without the loss of the calibration data updates stored in the SSLC system microprocessors. The information required to address this issue was part of Open Issue 1 in the DSER (SECY-91-294). GE subsequently revised the SSAR to state that when transfer is made to the RSS, the 4-20ma outputs are routed directly to the RSS, and the automatic calibration function in the RMUs is no longer part of the input signal to the RSS. During the use of the RSS, the automatic calibration function will not be available. Conventional, manual calibration of I&C equipment is available if the length of time of operation of the RSS requires recalibration. This is acceptable to the staff, and this DSER (SECY-91-294) open issue is, therefore, resolved.

7.4.3 Evaluation Findings

The review of the system interfaces for the systems required for safe shutdown included the sensors, circuitry, redundancy features and the actuated devices that provide the I&C functions to prevent the reactor from returning to criticality and provide a means for adequate residual heat removal. The review also addressed the interfaces between the safe shutdown systems and the RPS. The primary characteristics of these systems and requirements are included and verified in the CDM.

The staff concludes that the systems required for safe shutdown are acceptable and meet the relevant requirements of General Design Criteria (GDC) 2, 4, 13, 19, 34, 35, 38, and 44 and the applicable standards and regulatory guides. This conclusion is based on discussions above and summarized as follows:

The staff examined the information submitted for this design to determine its conformance to the GDC, standards and guidelines identified in the SSAR Section 7.1 and the

SRP. The staff finds that there is reasonable assurance that systems conform fully to the guidelines applicable to these systems.

The staff's review has included the identification of those systems and components required for safe shutdown which are designed to survive the effects of earthquakes, other natural phenomena, abnormal environments and missiles. Based upon our review we conclude that the design of those systems and components is consistent with the design bases. Additional evaluation is provided in Section 7.1 of this report. Therefore, the staff finds that the design of these systems and components satisfies this aspect of the GDC 2 and 4.

Based on the review, the staff concludes that instrumentation and controls have been provided to maintain variables and systems which can affect this fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems within prescribed operating ranges during plant shutdown. Therefore, the staff finds that the systems required for safe shutdown satisfy the requirements of GDC 13.

Instrumentation and controls have been provided within the control room to allow actions to be taken to maintain the nuclear power unit in a safe condition during shutdown including a shutdown following an accident. Equipment at appropriate locations outside the control room have been provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown, and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures. Therefore, the staff concludes that the systems required for safe shutdown satisfy the requirements of GDC 19.

The staff review of I&C required for safe shutdown systems included determination of conformance to the requirements for testability, operability with onsite and offsite electrical power, and single failure. The staff concludes that these systems as described above incorporate provisions for testability and operability with onsite and offsite power, and to cope in the event of a single failure where applicable and, therefore, meet the relevant requirements of GDC 34, 35, 38, and 44.

7.5 Safety-Related Display Instrumentation and Information Systems Important to Safety

7.5.1 System Description

Safety-related display systems are those which provide information (1) for manually initiated and manually controlled safety functions, (2) to indicate that the plant safety functions are being accomplished, and (3) to provide information from which appropriate action can be taken to mitigate the consequences of anticipated operational occurrences and accidents.

The information systems important to safety provide the operator with the status of the plant to allow manual safety actions to be performed when necessary. The following systems are identified in the SSAR as information systems important to safety:

a. Safety parameter display system (SPDS)

b. Information systems associated with emergency response facilities

c. Nuclear data link

The plant site emergency response center and communications links with the NRC emergency response center are conditions of the COL and are not addressed in this section.

This report evaluates the instrumentation aspects of the information systems with emphasis on interfaces between these systems and the SSLC and EMS as well as the application of advanced technology to processing and display of data important to safety. GE presented in the SSAR a comprehensive list of variables that were considered essential for providing safety-related information to the operators. Tables of conformance and specific exceptions to the guidelines of RG 1.97 were provided in the SSAR, and functional requirements for display of data were provided in the SSAR process system descriptions. One difference between the ABWR and currently operating BWRs is the incorporation of a Class 1E NMS that is in conformance with RG 1.97 Category I instrumentation criteria.

The manner in which the required data is processed and displayed, and dependencies on supporting hardware and software were not described in the SSAR. This was identified as part of the open item in the DSER (SECY-91-294) concerning level of detail (Open Item 1). The details of the displays will be determined during the implementation of the human factor CDM process described in Section 18 of this report. GE has committed to provide the displays for Category 1 RG 1.97 parameters on the fixed mimic panel. These parameters are also available at the operators display console. SSAR Chapter 18, Table 18F, provides a listing of the parameters and the general location of the associated display equipment. The supporting I&C display system equipment is described in Section 7.2 of this report with a similar emphasis on the use of the ITAAC process to implement the I&C design.

The accuracy of the RG 1.97 displays is not specified in the RG 1.97 guidelines or in the SSAR. However, the Emergency Procedure Guidelines imply an accuracy in the displays. For example, the Primary Containment Control Guideline (Ref. SSAR 18A.5) specifies an entry condition when the suppression pool water level is above 7.1 meters or below 7.0 meters. The TS also have similar accuracy requirements. The staff finds that the accuracy requirements implied will be met with the expected equipment. However, because the equipment has not been selected, accuracy will be confirmed in the ITAAC process. The final TS prepared by the COL applicant prior to fuel load will also include the specific setpoints and accuracy for the selected equipment.

NUREG-0737 Item I.D.2 requires that each applicant install a SPDS that will display to operating personnel a minimum set of parameters which define the safety status of the plant. This can be attained through continuous indication of direct and derived variables as necessary to assess plant safety status. Operating reactors have implemented the SPDS with a stand-alone display design. The ABWR is significantly different in that the SPDS parameters are integrated into the total main control room information display design as are the RG 1.97 parameter displays. The staff finds this acceptable. Additional discussion on the SPDS design is provided in Section 18 of this report.

7.5.2 Findings and Conclusions

The scope of the staff's review included an assessment of the proposed application and design of the EMS and SSLC to support operator displays important to safety. Other documentation normally reviewed for information system design such as component states, functional control diagrams, electrical and physical layout drawings, and descriptive information were not available and will be part of the ITAAC. The staff has considered applicable criteria, guidelines, and design bases, including those for indication of bypassed or inoperable safety systems, in the review discussed in this section.

The staff evaluated the information submitted for the ABWR information system design to determine its conformance to the guidelines identified in the SSAR Section 7.1 and the SRP. The review was concerned with the interfaces between the safety-related information system and the safety-related I&C for systems such as the RPS, LDS, and ECCS.

Table 7.5-7 of the SSAR lists drywell pressure as one of the variables required for indication of manual actions necessary for reactor shutdown from outside the control room. However, the parameters listed in SSAR Section 7.4 for display on the remote shutdown panel do not include this parameter. GE stated that this parameter is not required for shutdown using the RSS in the absence of a postulated design-basis event and should, therefore, not be listed in Table 7.5-7. Revision of this table was identified as DFSER Confirmatory Item 7.5.2-1. Table 7.5-7 was revised to remove the extraneous reference to drywell pressure and, therefore, this item is resolved.

The safety-related display information provided to the operator is derived from the SSLC and EMS. Additional display information is provided from the non-essential systems. The display information was considered in the resolution of the I&C system common-mode failure issue addressed in Section 7.2 of this report because the analysis of a common-mode failure in conjunction with an event included the information available to the operator.

Based on the above discussions and findings, the staff concludes that the ABWR design includes the necessary operator display information and, therefore, meets the requirements of RG 1.97 for post-accident monitoring instrumentation and TMI Action Plan Item I.D.2 for the SPDS, and is acceptable. The detailed information on the safety-related display instrumentation will be reviewed during the implementation of the ITAAC.

The staff concludes that the safety-related display instrumentation and information systems important to safety are acceptable and meet the requirements of GDC 2, 4, 13 and 19. This conclusion is based on discussions above and summarized as follows:

The staff concluded in Section 7.1 of this report that GE has identified in the SSAR the I&C systems which are important to safety. Additional evaluation is provided in Section 7.1 of this report.

The staff has reviewed the design of those systems and components which are required to survive the effects of earthquakes, other natural phenomena, abnormal environments and missiles. Based on the staff's review, the staff concludes that the design of those systems and components is consistent with the design bases. Therefore, the staff finds that the design of these systems and components satisfies this aspect of the GDC 2 and 4.

The staff concludes that the display and information systems to safety include appropriate variables and that their range and accuracy are consistent with the plant safety analysis. Therefore, we find that the information systems satisfy the requirements of GDC 13 and the applicable guidelines satisfy the requirements of GDC 19 with respect to information to operate the unit safely under normal conditions and to maintain it in a safe condition under accident conditions.

7.6 All Other Instrumentation Systems Required for Safety

The instrumentation systems included in this section are those required for safety but not previously discussed in other sections of this report, although some aspects of these systems are included in previous sections of this report.

7.6.1 System Description

The following systems are described in this section:

(1) neutron monitoring system
(2) PRM system
(3) high-pressure/low-pressure interlocks
(4) drywell vacuum relief system
(5) containment atmosphere monitoring (CAM) system
(6) SPTM System

All of these systems are included in the CDM. The CDM had not been reviewed at the time of issuance of the DFSER, and this was identified as DFSER Open Item 7.6.1.1. Descriptions of the above systems have been included in the CDM and this open item is, therefore, resolved. Section 14.3 of this report discusses the acceptability of the CDM.

7.6.1.1 Neutron Monitoring System

The safety-related subsystems of the NMS consist of the SRNM, the LPRM, and the APRM subsystems. The LPRM and the APRM together are referred to as the PRNM. The non-safety-related portions of the NMS (the automatic traversing in-core probe (ATIP) and multi-channel rod block monitor (MRBM)) are discussed in Section 7.7 of this report.

The SRNM monitors neutron flux from the source range (1.E+3nv) to 15 percent of rated power. The SRNM subsystem has 10 SRNM channels with each channel having one fixed in-core regenerative fission chamber sensor. The SRNM preamplifier signals are transmitted to the SRNM digital measurement and control (DMC) units in the main control room. The DMC units contain the software algorithms for signal processing, neutron flux, and power calculations. The SRNM was described by GE as being functionally the same as the wide range NMS in currently operating BWR plants. The SRNM provides trip signals to the RPS, and rod block signals to the rod controls. Each of the four trip channels receives input signals from a different set of SRNM channels. Unlike the other sensor inputs to the SSLC, the NMS provides a trip/non-trip decision directly to the TLU without use of the DTM to process the sensor data.

Three SRNM channels provide input to each of SSLC Divisions I and III, and two SRNM channels provide input to each of SSLC Divisions II and IV. The 10 SRNM channels are divided into three bypass groups. A total of three SRNM channels can be bypassed with no more than one SRNM bypassed per SSLC channel. No additional divisional bypass is allowed. If two (in Divisions II and IV) or three (in Divisions I and III) of the SRNMs are out of service, one channel of the RPS will be tripped.

The PRNM consists of the LPRM and APRM subsystems. The LPRM monitors power in the power range. The LPRM provides signals to the APRM and to the plant computer. The LPRM consists of 52 detector assemblies, each with four fission chamber detectors. The LPRM channels provide trip signals when an LPRM is upscale, downscale, or bypassed. The APRMs consist of four DMC APRM channels, each of which receives the 52 LPRM signals as inputs. The APRM DMC units average the inputs to provide a core average neutron flux which corresponds to the core average power. Each APRM channel is associated with a single RPS trip channel. The APRM also provides rod block functions.

GDC 12 requires that the reactor core and associated coolant, control, and protection systems be designed to assure that power oscillations which result in conditions exceeding specified acceptable fuel design limits are not possible, or can be reliably and readily detected and suppressed. BWR licensees were requested in NRC Bulletin No. 88-07 to take actions to prevent the occurrence of uncontrolled power oscillations during all modes of operation. The NRC also requested that the BWROG perform generic evaluations of the BWR plant response to core thermal hydraulic instabilities, and provide long-term solutions to prevent or quickly mitigate oscillations or operation in potentially unstable power/flow regions. The BWROG committee on thermal hydraulic stability has developed a long-term stability solution. A licensing Topical Report, NEDO-31960, "BWR Owner's Group Long-Term Stability Solutions Licensing Methodology," has been submitted for NRC review and approval. This topical report describes several options for BWR licensees to choose to deal with stability concerns.

In response to NRC Q440.187, GE stated that in order to meet the stability design requirements specified in the ALWR Utility Requirements Document, Option III, the OPRM system, a microprocessor-based protection system, will be implemented in the ABWR design. The OPRM system uses microprocessors to monitor groups of APRM signals. Upon detection of neutron flux oscillations characteristic of a thermal-hydraulic instability, the system will initiate an automatic suppression function (ASF) to suppress oscillations prior to exceeding safety limits.

Licensing Topical Report NEDO-31960 states that the OPRM which also uses the LPRM input is a Class 1E protection system and conforms to all applicable requirements of IEEE-279-1971. There are four OPRM channels, each of which provides inputs to trip logics which initiate an ASF. The OPRM function is in parallel with, and independent of, the existing Class 1E and non-Class 1E functions of the power range NMS. The OPRM does not affect the design bases for the existing power range monitoring components, their calibration, or their separation.

The OPRM function provides inputs to the ASF for the purpose of suppressing oscillations prior to exceeding the plant minimum critical power ratio safety limit. The OPRM is installed and maintained as an RPS protection function or for a select rod insert (SRI) function. The SRI function is intended to reduce core power to less than the turbine bypass capacity, so that the unit avoids a scram during a load rejection event. For implementation as an RPS function, four OPRM channels provide four separate inputs, one for each RPS trip channel. Any two channels in trip will result in a reactor scram.

GE has indicated and the staff concurs that Option III, LPRM-based, OPRM system is the preferred method of addressing stability. The ABWR OPRM system meets the requirements specified in the EPRI URD for ALWR and is consistent with the guidelines in Topical Report NEDO-31960. The staff, therefore, finds the OPRM system in the ABWR design to be acceptable. This was identified as DFSER Confirmatory Item 7.2.1-2. The SSAR and the CDM have been revised to include the OPRM. This item is, therefore, resolved.

7.6.1.2 Process Radiation Monitoring System

Radiation monitoring is provided on a number of process lines, HVAC ducts, and vents. The following radioactive material discharge routes are monitored for radiation:

(1) MSL tunnel area
(2) reactor building ventilation exhaust (including fuel exchange area)
(3) radwaste liquid discharge
(4) off-gas discharge
(5) gland steam condenser and mechanical vacuum pump off-gas discharge
(6) stack discharge
(7) turbine building vent exhaust
(8) standby gas treatment ventilation exhaust
(9) drywell sump liquid discharge
(10) control building air intake supply
(11) radwaste building ventilation exhaust

The four MSL tunnel monitors are Class 1E inputs to the RPS and are input to the DTMs for their respective channels.

The reactor building ventilation system radiation monitoring is Class 1E and includes the four exhaust air radiation monitors, the four fuel handling area exhaust radiation monitors, and the eight control building air intake radiation monitors. The exhaust air and the fuel area provide inputs to the leak detection system, and trip the ventilation systems on indication of high radiation. The control building air intake monitors isolate the HVAC for the control room upon indication of high radiation.

7.6.1.3 High-Pressure/Low-Pressure Systems Interlock Protection Functions

The only high/low-pressure interfaces for the reactor vessel involve the low- pressure modes of the RHR system. The logic for the pressure and level sensor inputs which provide the RHR system isolation valve closure signals is a two-out-of-four high reactor pressure or low RPV water level signal.

The inboard and outboard containment/pressure isolation valves for each of the three trains of the RHR system are powered from separate electrical divisions. The valves have permissive logics which prevent them from being opened when reactor pressure is greater than RHR system design pressure or when reactor water level is less than Level 3. These valve closure signals are provided by four divisionalized sensors in a two-out-of-four logic. An additional interlock is provided for the RHR system isolation valves from the RHR system area ambient temperature. The SSAR description of the valve interlocks originally contained an inconsistency concerning the RPV water level at which isolation occurs. This was DFSER Confirmatory Item 7.6.1.3-1. GE subsequently revised the SSAR to resolve the inconsistency. Therefore, this item is resolved.

7.6.1.4 Fuel Pool Cooling and Cleanup System

The fuel pool cooling and cleanup system is classified as a non-safety-related system and is discussed in Section 7.7 of this report.

7.6.1.5 Wetwell-to-Drywell Vacuum Breaker System

Direct control of the wetwell-to-drywell vacuum breakers is not provided. However, the open/close position of these vacuum breaker valves is monitored. The related instrumentation which indicates proper function of the vacuum breakers is described in Section 6.2.1.7 of the SSAR. This instrumentation includes the open/close position indicators for the wetwell-to-drywell vacuum breaker valves and indications of wetwell-to-drywell differential pressure.

7.6.1.6 Containment Atmosphere Monitoring System

The CAM system is a two-train Class 1E system. Each CAM system division monitors the total gamma-ray dose rate and concentration of hydrogen and oxygen in the drywell and/or the suppression chamber. The CAM system is a microprocessor-based system which provides measurement, recording, and alarms in the control room for operator information. Each divisional gamma radiation monitoring channel can be energized manually by the operator or automatically by the LOCA signal. Each divisional hydrogen/oxygen monitoring subsystem is powered continuously during plant operation.

7.6.1.7 Suppression Pool Temperature Monitoring System

The two-train SPTM system consists of eight sensor locations around the circumference of the pool, each of which has a group of four sensors for the two trains. The signal processing for the SPTM system is performed by microprocessors. The I&C of the SPTM system are powered by four divisionally separated electrical buses. The SPTM system initiates RHR suppression pool cooling, RCW load shedding, and RPS trip signaling. It also provides information for the operator in the control room and the remote shutdown panel.

7.6.2 Specific Findings and Evaluation

The SSAR includes an analysis of the safety-related portions of all instrumentation systems required for safety discussed above. The analysis is to show conformance to general functional requirements and specific regulatory requirements. The staff reviewed the information submitted for these systems to determine their conformance to the general design criteria, standards, and guidelines identified in the SSAR Section 7.1 and the SRP. Based on the review, the staff concludes that the design of the instrumentation for the systems in Section 7.6 of the SSAR and discussed in this section meet the requirements of GDC 2, 4, 10, 12, 13, 19, 23, 24, 28, 33 and 44, and the guidelines of applicable standards, regulatory guides, and branch technical positions, as applicable and are, therefore, acceptable. The additional requirements listed in Sections 7.1 and 7.2 of this report for SSLC qualification also apply to the safety-related portions of these systems. These systems are included in the CDM.

7.7 Control Systems

7.7.1 System Description

This section discusses control systems which are considered by GE to be not essential for the safety of the plant. These systems primarily use microprocessor-based equipment and transmit information via non-essential multiplexors. These systems include:

(1) NBS reactor vessel instrumentation
(2) RC&IS
(3) RFC system
(4) FDWC system
(5) process computer system and power generation control system (PGCS)
(6) NMS, ATIP and MRBM subsystems
(7) automatic power regulator (APR) system
(8) SB&PC system
(9) non-essential multiplexing system
(10) fire protection system
(11) drywell cooling system
(12) instrument air systems
(13) makeup water system
(14) atmospheric control system
(15) fuel pool cooling and cleanup system
(16) communications system

Although these systems are not directly needed for the performance of safety functions, their operation is important to the reliability of the plant. The non-safety systems are designed such that their failure will not prevent the proper operation of the safety systems. These systems are also designed to be of high quality to minimize the challenges to safety system functions. Because the CDM for these systems had not been reviewed when the DFSER was written, the staff identified the incomplete review of the CDM and ITAAC as DFSER Open Item 7.7.1-1. The key features of these systems are now included in the CDM and this item is, therefore, resolved. The codes and standards required for the safety systems as discussed in Section 7.1 of this report do not apply to these systems unless otherwise specified. The digital system design process for the safety systems are similar for these non-safety systems, except that documentation requirements are not as stringent.

7.7.1.1 Nuclear Boiler System Reactor Vessel Instrumentation

Only the non-safety portion of the NBS is included in this section. This part of the NBS provides monitoring and control input of variables during normal plant operations. The variables monitored include:

(1) reactor vessel temperature,

(2) reactor vessel water level (shutdown, narrow, wide, and fuel zone ranges),

(3) reactor core differential pressure,

(4) reactor vessel pressure,

(5) SRV seal leak detection,

(6) feedwater temperature

Sensors for the above variables share sensing lines with the safety system sensors. Separation and isolation is maintained between the safety and non-safety portions of the system.

7.7.1.2 Rod Control and Information System

The RCIS is a non-safety-related system which provides the operator with the information necessary to make changes in nuclear reactivity so that reactor power level and power distribution can be controlled by manipulating the control rods. This system includes those interlocks which inhibit rod movement (rod block) under certain conditions. The RCIS is also used to implement the ARI control rod insertion function, and a backup scram follow function. Upon an RPS scram, this system is initiated and starts the motor-driven FMCRD, to follow the RPS hydraulic scram of the rods.

The RCIS consists of two independent channels for monitoring and control rod positioning during normal operations. Disagreement between the two channels results in a rod block. The RCIS is designed to be a single-failure proof system. The CRD components which are required for shutdown of the plant and/or whose failure can result in gross fuel damage, are designed to meet the requirements of a safety-related system.

The two RCIS cabinets contain the rod worth minimizer, automated thermal limit monitor, and rod block functions. The RCIS receives the scram follow command from the RPS. It also receives the selected control rod run-in signal from the RFC system and the APR system. The RCIS also provides the signals to the RFC system to reduce flow when the RCIS fully inserts the rods as the result of an ARI initiation or scram follow.

The RCIS allows the operator to completely bypass up to eight control rods by declaring them inoperable and placing them in bypass. The operator can substitute a position for the bypassed rod into the RCIS. The RCIS has a dedicated control interface in the main control room.

7.7.1.3 Recirculation Flow Control System

The RFCS provides each of the two channels of the RCIS with separate isolated trip signals indicating the need for automatic selected control rod run-in upon trip or run-back of the recirculation pumps. This system receives reference power level signals from the NMS and compares the reference power level signals with the power level setpoint. The signals provided represent the validated total core flow. The primary purpose of this system is to control the speed of the 10 reactor internal pumps.

In the DSER (SECY-91-294), the staff expressed concern regarding GE's request to change the normal classification of a loss of all forced circulation from a moderate frequency event to an event not expected to occur in the lifetime of the plant. The staff presented a discussion of the reasons that it did not agree with GE's position. Included in that discussion was the concern that a potential common-mode software error could result in a loss of all forced circulation. In response, GE provided information concerning potential common-mode software failures and the reliability of the RFCS.

The failures addressed by GE included those of sensors, control modules, power supplies, multiplexor links, and speed controllers. Other failure mechanisms such as incorrect operator action, or maintenance and common-mode software errors, were also addressed with means indicated to reduce their likelihood, but such mechanisms are still possible causes of simultaneous reactor internal pump trip. The staff concluded that improvements were made in the reactor recirculation system of the ABWR when compared to currently operating plants; however, the staff considers the simultaneous trip of all of the recirculation pumps due to a common-mode failure to be a credible event that is expected to occur during the lifetime of the plant. Therefore, measures are required to mitigate such an event, or analysis provided to verify that the consequences are acceptable. This event is evaluated in Chapter 15 of this report.

7.7.1.4 Feedwater Control System

The FDWC system is a three-train system that controls the flow of feedwater into the RPV. The I&C system for feedwater operation is a triplicated system which is single-failure proof. The feedwater system can be controlled manually or automatically. Normal automatic control is based on reactor water level (when steam flow is very low) or on reactor water level, main feedwater line flow, and feedpump suction flow measurements during power operation.

7.7.1.5 Process Computer System and Power Generator Control System

The process computer is intended to provide a determination of core thermal performance, and to improve data reduction, accounting, and logging functions. The PGCS, a separate function of the process computer system, monitors overall plant conditions, issues control commands, and adjusts setpoints of lower level controllers to support automation of the normal plant startup, shutdown, and power range operations. The PGCS issues command signals to the turbine master controller.

The staff has reviewed MPL C71-4010 (Rev. 1, July 2, 1990), "Reactor Protection System Design Specification," which requires that both the tripped and reset conditions of the RPS-related sensor instrument channels and the RPS automatic or manual trip systems be logged by the process computer. For all conditions that cause reactor trip, the computer shall identify the specific trip variable, the divisional channel identity and the specific automatic or manual trip system. The staff identified the inputs to the plant computer as DFSER Confirmatory Item 7.7.1.5-1. The plant process computer and its inputs to the trip scram data logger are included in the CDM and the SSAR. This item is resolved.

GE indicated that the plant computer will have the capability to trend performance of all safety-related sensors. This will provide the capability for detection of problems which have occurred previously regarding loss of oil in oil-filled transmitters at operating plants. This resolved the operating experience concern discussed in NRC Bulletin 90-01, and Supplement 1 to Bulletin 90-01.

7.7.1.6 Neutron Monitoring System, ATIP and MRBM Subsystems

The ATIP subsystem of the NMS is comprised of three TIP machines, each with a neutron sensor attached to a flexible cable. The system includes the associated drive mechanisms and guide tubes, and is used to obtain flux readings along the axial length of the core. The MRBM subsystem logic issues a rod block signal to the RCIS. This microprocessor-based system receives input neutron flux signals from the LPRMs and APRMs, core flow data from the NMS, and control rod status to determine when rod block signals are required.

7.7.1.7 Automatic Power Regulator System

The APR system controls reactor power by providing commands to rod position or reactor RFC instrumentation. The APR receives input from the plant process computer, the PGCS, the SB&PC system, and the operator's control console. The output demand signals from the APR are to the RCIS and the RFC and SB&PC systems. The APR logic is performed by redundant microprocessors.

7.7.1.8 Steam Bypass and Pressure Control System

The SB&PC system controls the reactor system pressure during normal operation. The system regulates the position of the turbine control and/or steam bypass valves.

7.7.1.9 Non-Essential Multiplexing System

The NEMS is separate from the EMS but is similar in function. The NEMS supports communication between the non-safety I&C systems. The NEMS will be diverse from the EMS (hardware and software).

7.7.1.10 Fire Protection System

The I&C aspects of the fire protection system consist of the detection and suppression portions. This system is automatically actuated by smoke, infrared, or temperature detectors when fire is indicated.

7.7.1.11 Drywell Cooling System

The drywell cooling system is used to limit the temperature of the various drywell zones within ranges dictated by the equipment requirements.

7.7.1.12 Instrument Air System

The instrument air system is discussed in Section 9.3.6 of the SSAR and is not discussed in this section.

7.7.1.13 Makeup Water System

The makeup water system is discussed in Section 9.2.3 of the SSAR and is not discussed in this section.

7.7.1.14 Atmospheric Control System

The atmospheric control system is discussed in Section 6.2.5 of the SSAR and is not discussed in this section.

7.7.1.15 Fuel Pool Cooling and Cleanup System

The fuel pool cooling and cleanup system was reclassified as a completely non- Class 1E system since the DFSER was written. It is an independent system which monitors and controls the fuel pool temperature and maintains the water quality of the pool. This system consists of redundant trains with power supplies that are backed by the combustion turbine generator.

7.7.1.16 Communications System

The paging system is designed to provide facilities for mutual communication and simultaneous broadcasting within the plant. The sound-powered telephone system provides communications primarily for fuel transfer, testing, calibration, and maintenance.

The communication systems consist of a power-actuated paging facility and a separate network of cables and jacks to facilitate the use of sound-powered telephones for maintenance and repair. The paging system is primarily used for intraplant communication during plant operation. Handsets and speakers are installed in the rooms indicated below:

(1) main control room
(2) electrical equipment room
(3) fuel replacement area
(4) turbine operation area
(5) periphery of control rod hydraulic units area
(6) feedwater pump room
(7) elevators
(8) exteriors of plant buildings

In addition to the basic paging function, the paging equipment can be used for automatic surveillance of the main amplifier and manual switching to a spare amplifier as necessary. The paging equipment produces an emergency signal (siren) upon actuation of an emergency pushbutton. The circuits from the main paging equipment to each junction box are wired in separate routes.

A separate telephone communication system using portable sound-powered telephones is provided, but is outside the scope of the ABWR design. The system provides communication between boards in the main control room, between the main control room and field stations, and between field stations during testing or inspections.

The communication systems do not have a safety-related function. However, they are used during emergencies and because the communication system is required when plant control is at the remote shutdown station, its availability must be demonstrated assuming a main control room fire. This was DFSER Open Item 7.7.1.15-1. GE subsequently revised SSAR Chapter 9.5.2 to described the survivability of the sound-powered phones in the event of a control room fire. This item is, therefore, resolved.

No identification of the EMI radiation levels or frequency range is identified for the communication transmitter/receivers to be installed in the plant. In addition, the sensitivity of the safety computer systems to the electromagnetic fields is undefined. Therefore, a test program with field measurements and operational descriptions is required if spurious effects upon safety-related I&C equipment due to communications is to be avoided. This was identified as DFSER Open Item 7.7.15-2. This will be verified as part of the EMI ITAAC that is included in the instrumentation and controls CDM. Therefore, this item is resolved.

Based on the above, the staff concludes that the design of the paging communication system ensures its availability as required and is, therefore, acceptable.

7.7.2 Specific Findings and Evaluations

In the SSAR, GE provided an analysis to demonstrate that the above non-safety-related I&C systems are not required for any plant safety function, and that the plant protection systems are capable of coping with all failure modes of these I&C systems. The analysis shows how the design of the above I&C systems conforms to general functional requirements and specific regulatory requirements. The staff reviewed the information submitted for these systems to determine their conformance to the GDC, standards, and guidelines identified in the SSAR Section 7.1 and the SRP. Based on the review, the staff concludes that the design of the I&C systems in Section 7.7 of the SSAR and discussed in this section meet the requirements of GDC 13 and 19, and the guidelines of applicable standards and regulatory guides and is, therefore, acceptable.

7.8 COL License Information

Section 7.8 of the SSAR provides a discussion of topics that are not specifically addressed in the SRP but are important to the design and operation of safety-related equipment. These topics will be addressed by the COL applicant in its application. These topics include: (1) effects of station blackout on HVAC systems and the subsequent effects on plant electronics; (2) effects of ESD on exposed electrical and electronic equipment components; (3) effects of localized high heat spots in semiconductor materials for computing devices; (4) criteria for interfaces between the ABWR I&C systems and systems outside of the scope of the ABWR design certification; and (5) comprehensive functional tests necessary to meet the plant TS.

The staff requested (Q420.014) that GE address the effects of station blackout on that portion of the HVAC system which is required to maintain the function of plant electronics. GE responded that this issue will be addressed as a COL applicant requirement by performance of a temperature/heat rise analysis for the station blackout (no HVAC) scenario using the resulting environmental temperatures for the specific plant location as a basis for confirming appropriate electronic equipment performance. The heat rise analysis will verify that the required electronic equipment has been qualified to the highest expected temperature assuming station blackout. This is acceptable and will be verified as part of the ITAAC process by the COL applicant.

The staff requested (Q420.90) that GE address the possible effects of ESD on the proper performance of keyboards, keyed switches, and other exposed electrical equipment components. GE's response described the damage to components and system upsets that ESD can cause. The response described the steps used in modern equipment design to protect the equipment from ESD and precautions that should be used in their installation. The equipment design standards, installation and maintenance procedures are included as part of the EMC ITAAC. The COL applicant will verify by the EMC ITAAC that the GE recommendations for grounding and shielding are followed or provide an acceptable alternative.

The staff requested (Q420.92, Q420.94, Q420.95) that GE address possible localized hot spots in semiconductor materials due to high current densities and their effects on equipment reliability. GE's response described the requirement for a thermal analysis which will follow the methods of MIL-HDBK-217 and MIL-HDBK-251. Because the worst case for potential hot spots occurs during a postulated station blackout scenario (no HVAC) and, therefore, will require plant-specific information to be incorporated in the thermal analysis, the staff agrees that this issue will be addressed as part of the ITAAC process. The COL applicant will verify that adequate compensation is provided for internal heat rise to ensure proper electronic equipment performance.

GE performed an interface study of each of the I&C systems included in Chapter 7 of the SSAR, and determined that there are no safety-related electrical signal interfaces between safety systems and systems that are site specific that have not already been included in the requirements of the previous sections. Therefore, GE indicated that there are no interface requirements necessary to ensure safety-related system performance. However, the SSAR did not specifically address non-safety I&C system interfaces with safety-related systems outside the ABWR scope. If there are any such interfaces with equipment outside of the scope of the ABWR SSAR, the existing requirements for safety/non-safety I&C system isolation will apply and will be verified during the ITAAC phase. The COL applicant will verify that any safety/non-safety I&C interfaces are adequately separated and isolated.

The TS contain several different surveillance test requirements. One of those tests is a comprehensive functional test of the safety-related I&C systems that will be performed during each refueling outage. The TS require the tests, and the bases for the TS provide a short description of the testing that is to be done. GE has also included a detailed description of the comprehensive functional test in the SSAR. This description will be included by the COL applicant in its maintenance program.

7.9 Appendix 7A - Design Response to Appendix B of ABWR Licensing Review Bases

Appendix B to the GE ABWR LRB, dated August 1987, noted that the SRP did not provide standards and criteria for the review of state-of-the-art fiber optics, multiplexing, and computer controls. LRB Appendix B contains questions from the staff regarding digital I&C systems. The questions were intended to solicit additional information from the applicant beyond that normally provided in SAR because of the use of digital technology in the design. GE provided their responses to these questions in Appendix 7A of the ABWR SSAR. Since the DFSER was issued, GE also added SSAR Chapter 7 Appendices B and C to further address digital I&C system design issues, including defense-in-depth.

The responses in Appendices 7A, 7B, and 7C provide a level of design information similar to that provided in the I&C systems sections of the SSAR and in responses to other staff requests for additional information. The information provided was not sufficiently detailed in and of itself to permit the staff to reach its safety conclusions. An example is the GE statement in the response to NRC Request 11 (SSAR Appendix 7A) that "software development will, in general, follow RG 1.152." The staff concludes that the Appendix 7A response provides a commitment to address the issues involving digital system design as identified in Appendix B to the LRB, but does not present sufficient description of the design to demonstrate how the commitments are to be met. Appendix 7A was revised following the DSER (SECY-91-294) to include a commitment that the ABWR digital systems will meet the criteria of the additional standards listed in Sections 7.1 and 7.2 of this report. Verification that the I&C system design conforms with this commitment will be accomplished during the ITAAC phase.

In addition to the hardware and software aspects of the safety-related I&C system design that have been discussed previously in this report, SSAR Appendix 7A provides a commitment to the guidelines of RG 1.153 which endorses IEEE-603. For the I&C systems of the ABWR design, IEEE-603 provides guidance which is similar to that of IEEE-279. GE stated that the ABWR safety-related I&C system design be in conformance with IEEE-603, and the implementation of the I&C design will be verified during the ITAAC. This is acceptable to the staff.

Based on the review of the information provided in the SSAR (Appendices 7A, 7B, and 7C) and in related CDM, the staff considers that issues regarding standards and criteria for digital equipment, raised by the LRB Appendix B questions, have been adequately addressed.

7.10 Unresolved Safety Issues, Generic Safety Issues, and Operating Experience

7.10.1 Unresolved Safety Issues and Generic Safety Issues

USIs and GSIs are discussed in Chapter 20 of this report.

7.10.2 Operating Experience

The EPRI ALWR Utility Requirements Document, Volume II, for evolutionary reactor design provides a general description of the operating experience that is necessary before equipment should be considered for use in future nuclear power plants. The general guidelines specify that approximately 3 years of successful experience in applications similar (but not necessarily nuclear plant-related) to the intended nuclear power plant installation is appropriate. Failure to meet this EPRI requirement results in increased prototyping as specified in the EPRI requirements. GE has committed to the EPRI requirements and the staff finds this acceptable. This was DFSER Confirmatory Item 7.10.1-1. The SSAR has been revised to address operating experience as indicated above, and this is acceptable. DFSER Confirmatory Item 7.10.2-1 is, therefore, resolved.

In addition, as part of the operating experience review, the staff reviewed the following NRC bulletins and generic letters (GLs) which have been issued since 1980. The primary intent of this review effort was to assure that the operating experience gained as described in these documents is incorporated into the design and operating features for the ABWR.

The generic communications reviewed by the staff for operating experience in the I&C systems area and a discussion of the review follows:

<u>NRC Bulletins</u>

a. Bulletin 80-01 (January 11, 1980), "Operability of ADS Valve Pneumatic Supply."

This item is not applicable to the SSAR Chapter 7 I&C system review.

b. Bulletin 80-06 (March 13, 1980), "Engineered Safety Feature Reset Controls."

Bulletin 80-06 listed three actions to be taken by the licensee. The first was to review the I&C system schematics and verify that upon reset of an ESF actuation signal, the safety-related equipment remains in its emergency mode. For the ABWR, the ESF systems are reset individually and manually, and will remain in their emergency mode. This is acceptable.

The second item requires verification that the as-built I&C system configuration is in conformance with the schematics. This will be verified by the COL applicant during the ITAAC phase, and is acceptable.

The third item pertains to plant-specific corrective actions by operating plants and does not affect the ABWR. The concerns of Bulletin 80-06 have been adequately addressed in the ABWR design.

c. Bulletin 80-20 (July 31, 1980), "Failures of Westinghouse Type W-2 Spring Return to Neutral Control Switches."

Based on the information provided by GE, this type of switch will not be used in the ABWR design and, therefore, Bulletin 80-06 is not applicable.

d. Bulletin 81-02 (April 9, 1981), "Failure of Gate Type Valves to Close Against Differential Pressure."

This item is not applicable to the SSAR Chapter 7 I&C system review.

e. Bulletin 82-04 (December 3, 1982), "Deficiencies in Primary Containment Electrical Penetration Assemblies."

This item is not applicable to the SSAR Chapter 7 I&C system review.

f. Bulletin 88-07 (June 15, 1988), "Power Oscillations in Boiling Water Reactors."

The measures provided in the ABWR I&C system design to deal with power oscillations are addressed in Sections 7.2 and 7.6 of this report.

g. Bulletin 90-01 (March 9, 1990), "Loss of Fill Oil in Transmitters Manufactured by Rosemount."

GE committed for the ABWR I&C system design not to use the transmitters of concern identified in the bulletin which were built before July 1989. In addition, the plant computer for the ABWR will have the capability to trend the operational data of all safety-related transmitters. This trending is capable of detecting the type of problems described in Bulletin 90-01. This is, therefore, acceptable.

NRC Generic Letters

a. GL 80-03 (April 10, 1980), "Clarification of the Term "Operable" as it Applies to Single Failure Criterion for Safety Systems Required by TS."

This item is not applicable to the SSAR Chapter 7 I&C system review.

b. GL 83-27 (July 6, 1983), "Surveillance Intervals in Standard Technical Specifications."

This item is not applicable to the SSAR Chapter 7 I&C system review.

c. GL 88-02 (January 20, 1988), "Integrated Safety Assessment Program II (ISAP II)."

This item is not applicable to the SSAR Chapter 7 I&C system review.

d. GL 88-20 (November 23, 1988), "Individual Plant Examination of Severe Accident Vulnerabilities."

This item is not applicable to the SSAR Chapter 7 I&C system review.

e. GL 89-14 (August 21, 1989), "Line-Item Improvements in Technical Specifications - Removal of 3.25 Limit on Extending Surveillance Intervals."

Issues in GL 89-14 related to I&C system TS are discussed in Section 7.11 of this report.

f. GL 91-04 (April 2, 1991), "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle."

Issues in GL 91-04 related to I&C system TS are discussed in Section 7.11 of this report.

g. GL 91-09 (June 27, 1991), "Modification of Surveillance Interval for Electrical Protection Assemblies in Power Supplies for the Reactor Protection Systems."

Issues in GL 91-09 related to I&C system TS are discussed in Section 7.11 of this report.

One issue associated with operating experience concerns an aspect of the adequacy of the defense-in-depth and the diversity of the instrumentation design required to provide defense against common-mode failures remains open. Specifically, the issue concerns the need for diverse reactor pressure vessel water level measurement instrumentation. This issue was Open Item 20.3.8 in the DFSER and is also addressed in Section 7.2 of this report. The primary concern is in identical measurement instrument techniques used (condensing chambers and differential pressure transmitters) to measure the RPV water level and provide signals to the I&C logic.

Recent anomalies have been observed in RPV level instrumentation (and discussed in GL 92-04, "Resolution of the Issues Related to Reactor Vessel Water Level Instrumentation in BWRs Pursuant to 10 CFR 50.54(f)," Information Notice 92-54, "Level Instrumentation Inaccuracies Caused by Rapid Depressurization," and Bulletin 93-03, "Resolution of Issues Related to Reactor Vessel Water Level Instrumentation in BWRs") that were caused by the effects on non-condensible gas in the condensing chamber and reference leg of the water level instrumentation. GE committed to a modification of the design for the ABWR water level instrumentation by revising the sensing line connections and adding a water backfill capability to minimize the possibility of entrapping non-condensible gas in the reference legs and condensing chambers. The proposed modifications will be tested to validate the appropriateness of the modification. This issue is discussed in more detail in Chapter 20 of this report.

Operating experience at Grand Gulf Station (found during testing) has demonstrated that the solenoid coils on scram pilot valves can be damaged by undervoltage and thereby prevent the scram function. During the previous licensing review of Hatch 2, this issue was also raised and included questions of possible damage due to undervoltage to the RPS equipment in general, not just to the scram pilot valve solenoid coils. Resolution of this concern is addressed in Section 7.2.5 of this report.

A significant feature of the I&C system testing capability for the ABWR described by GE in the SSAR is the elimination of the need to lift leads and install jumpers when performing tests. Lifting leads and installing jumpers has been a source of problems at operating plants. This issue was part of DFSER Confirmatory Item 7.2.2.5-2. GE provided more detailed descriptions in the SSAR of the I&C system self-diagnostics and testing equipment to be used for surveillance and maintenance. The ABWR design has eliminated the necessity to lift leads or add jumpers for normally scheduled surveillance and maintenance. This item is, therefore, resolved.

Based on the above, the staff concludes that GE had adequately addressed and incorporated into the ABWR design for the I&C systems, features which deal with concerns identified from past operating experience. Therefore, operating experience issues are resolved.

7.11 Technical Specifications

The staff reviewed the draft ABWR TS provided prior to the issuance of the DFSER. Substantial questions were identified as discussed below, which required resolution prior to the staff finding the ABWR TS acceptable and, therefore, the TS were an open issue in the DFSER.

The design of the ABWR I&C systems is substantially different from that used in currently operating BWR plants and, therefore, the current BWR standard TS sections for the I&C systems were not readily applicable to the ABWR I&C systems. GE has substantially revised the ABWR TS to reflect the specific design of the ABWR.

Because the proposed ABWR I&C systems have not yet been in operation, there is no specific equipment history that can be used to assess the equipment surveillance intervals. The specific I&C equipment and vendors have not been selected, and are not required to be selected until after the COL is issued. The specifics of the I&C system equipment self-diagnostics are to be considered in assessing the TS surveillance intervals.

Prior to the issuance of the DFSER, GE had not claimed credit for the self- diagnostics to meet any particular TS requirements. The self-diagnostics are an integral part of the design, and form a significant basis for the establishment of surveillance intervals for equipment beyond the self-diagnostics. The inclusion of the self-diagnostics in the design is necessary to the acceptance of the TS as discussed in Section 16 of this report.

MPL C71-5030 (Rev. 0, April 23, 1990), "Reactor Protection System Verification and Validation Criteria Design Specification," states that field installation and validation tests for the RPS will be performed while the multiplexing loops for each RPS division are operational. The DFSER stated that the limiting conditions for plant operation must address the possibility of a multiplexor loop being out of service. The ABWR TS incorporate limiting conditions for this situation, and therefore, this item is resolved.

MPL C71-4010 (Rev. 1, July 2, 1990), "Reactor Protection System Design Specification," states that one sensor channel may be bypassed and the system actuation will be based on a two-out-of-three logic. This bypass is implemented at the input to the TLU. Bypass of one division of output logic will also result in a two-out-of-three coincidence logic. This bypass is implemented at the input to the OLU. There is no indication in the MPL document if both bypasses are allowed at the same time. The initial draft ABWR RPS TS referred only to the traditional channel check calibration and functional tests. In the DFSER, the staff stated that a description of how these I&C channel and division bypasses are to be implemented on the ABWR was required. A functional channel test should be defined. Subsequent versions of the ABWR TS added the definitions of sensor, logic, and output channels as well as a discussion of the various surveillance requirements including bypass conditions. This item is, therefore, resolved.

Plant operation in the event of power supply failures needed to be considered in the I&C system TS. A loss of one channel of SSLC power not only disables one channel of sensors and one channel of coincidence logic, but also disables 1/4 of the inputs to the other three coincidence logics. The DFSER stated that the TS needed to consider each of the credible equipment failures and the specific TS action that it invokes, if any. The TS were revised to consider credible failures and the appropriate statements of action have been incorporated. This item is, therefore, resolved.

The draft ABWR TS initially indicated that an indefinite bypass (no repair required until the next refueling outage) was appropriate for a channel out of service. No bases were provided to justify this condition. The staff noted that there is a potential single failure of the Division II, 6.9 KV/480 Vac transformer or 480 Vac switchgear which could disable both the Divisions II and IV SSLC, and at the same time not cause a reactor scram. If a Division I or III SSLC channel is in bypass at the same time, the result could be a loss of RPS function.

By letter dated June 2, 1993, on important features identified by the ABWR PRA, GE identified that the four divisions of the SSLC are designed to be highly reliable with features that reduce the possibility of inadvertent actuations. The PRA assumed a self-diagnostic fault detection rate for the SSLC of 0.95 with the remainder of the faults expected to be found during the quarterly SSLC surveillance required by TS. As discussed earlier, there is a lack of operational data for the SSLC equipment to support the PRA number.

Because of the above reasons, GE has revised the ABWR TS to remove the indefinite bypass of an SSLC channel. This is acceptable to the staff.

In the DFSER, the staff noted that specific operability aspects of the ABWR I&C system digital technology needed to be further evaluated when developing the TS. For example, if a failure of a safety-related I&C system (channel or division) during surveillance can be attributed to a software error, the appropriate TS operability requirements for other systems which may also be subject to failure as a result of the same software error are needed to be established. The TS were revised to include limits on continued operation and programmatic consideration (including reporting to the NRC) of software errors when they are discovered. This is acceptable to the staff.

The TS for design certification include I&C system setpoints that will be established by the COL applicant during the I&C system detailed design development. The setpoint methodology is included in the CDM and its implementation is included in the ITAAC.

Based on the above, the staff concludes that the ABWR TS for design certification incorporate the necessary operability and surveillance requirements for digital I&C systems and are, therefore, acceptable.

8 ELECTRIC POWER SYSTEMS

8.1 Introduction

The staff reviewed the advanced boiling water reactor (ABWR) design descriptions and commitments documented in Chapter 8 of GE Nuclear Energy's (GE's) standard safety analysis report (SSAR) Amendment 32. The bases for evaluating the adequacy of ABWR electric power systems presented in SSAR Chapter 8 were the acceptance criteria and guidelines for electric power systems contained in standard review plan (SRP) Chapter 8 and Regulatory Guides (RGs) 1.153 (Rev. 0), "Criteria for Power, Instrumentation, and Control Portions of Safety Systems," and 1.155 (Rev. 0), "Station Blackout." The Nuclear Regulatory Commission (NRC) approved these regulatory guides following the issuance of SRP Revision 3 (July 1983), and they apply to the ABWR electric power system design.

The staff's initial findings were included in the draft safety evaluation report (DSER), SECY-91-355. In addition, the staff's interim findings were included in the draft final safety evaluation report (DFSER), SECY-92-349. Open, confirmatory, and COL action items from these documents have been referenced in the discussions below.

8.2 Offsite Electric Power System

The offsite electric power system is commonly called the "preferred" power system. The staff's evaluation of this system focused on the system's importance as the preferred supplier of electric power for the onsite power system (that is, the Class 1E ac-distribution system), which supplies power to safety systems.

For the ABWR, the preferred power system comprises the following circuits:

- Normal preferred power circuit - a back-feed circuit from the transmission network to the input terminals of each of the three redundant, onsite Class 1E ac-distribution systems through the main transformer and three unit auxiliary transformers.

- Alternate preferred power circuit - from the transmission network through one reserve auxiliary transformer to the input terminals of each of the three redundant, onsite Class 1E ac-distribution systems.

Because GE shares the ABWR design responsibility for this system with the combined license (COL) applicants, those parts that are outside the scope of design of the ABWR standard plant, and those parts that are within the scope of design of the ABWR standard plant are described and evaluated as follows.

8.2.1 Preferred Offsite Circuits Outside the ABWR Scope of Design

The following portions of the preferred power circuits are outside the scope of design of the ABWR standard plant:

- Normal preferred power circuit from the transmission network through the main power transformer to the low-voltage terminals of the main transformer.

- Alternate preferred power circuit from the transmission network through the reserve auxiliary transformer to the low-voltage terminals of the reserve auxiliary transformer.

8.2.1.1 Scope of GE Design of Offsite Preferred Circuits

Section 3.1.2.2.8.2.2 and Sections 8.2.1, 8.2.2, and 8.2.3 of SSAR Amendments 7 and 10 were inconsistent with regard to which parts of the offsite system are within (or outside) the ABWR standard design scope (DSER (SECY-91-355) Open Item 22). Subsequently, GE committed to revise Section 3.1.2.2.8.2.2 of SSAR Amendment 7 so that it would be consistent with the above defined scope of design.

Verification that GE included appropriate changes in a future SSAR amendment to reflect the above information was DFSER (SECY-92-349) Confirmatory Item 8.2.1.1-1. GE included this information, in Sections 3.1.2.2.8.2.2 and 8.2.1 of SSAR Amendment 32, which is acceptable.

8.2.1.2 Definition of Offsite System

GE's draft SSAR submittal of April 3, 1992, indicated that the offsite power system begins at the terminals on the transmission network side of the circuit breakers connecting the switching stations to the offsite transmission network. That draft also indicated that the offsite power system ends at the terminals of the plant's main generator and at the circuit breaker input terminals of the medium-voltage (6.9 kV) switchgear. This description is not consistent with the NRC SRP definition of an offsite system DSER (SECY-91-355) Open Item 27. Specifically, GE's definition appeared to exclude the transmission network, as well as the plant's main generator and gas turbine generator. GE committed to revise the SSAR in a future amendment to be consistent with the NRC SRP definition of an offsite system.

Verification that GE included appropriate changes in a future SSAR amendment to reflect the above information was DFSER (SECY-92-349) Confirmatory Item 8.2.1.2-1. GE included this information, in Sections 8.1.2.1 and

8.2.1.1 of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

8.2.1.3 Offsite Power System Interfaces

This section addresses the staff's evaluation of GE's response to DSER SECY-91-355 Open Item 24.

In the draft SSAR submittal dated April 3, 1992, GE responded to Open Item 24 from the DSER (SECY-91-355) by defining interface requirements for the offsite circuits outside the scope of design of the ABWR standard plant. GE committed to document similar interfaces in the SSAR in a future amendment. Verification that GE included appropriate changes in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.2.1.3-1. GE included the appropriate changes in Section 8.2.3 of SSAR Amendment 33, which is acceptable. Therefore, this item is resolved.

Section 8.2.3.1 of GE's draft SSAR submittal of April 3, 1992, indicated that a COL applicant "should" meet the interface requirements defined in Section 8.2.3 of the ABWR SSAR. Applicants who reference the ABWR design will be required to address all interface requirements in its design scope. GE committed to revise the SSAR in a future amendment to indicate that interface requirements "shall" be met by the applicant. Verification that GE included this change in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.2.1.3-2. GE included this change in Section 8.2.3 of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

GE also committed to revise in a future SSAR amendment the section on interface requirements to include a listing of the regulatory requirements and associated regulatory and industry guidance, which a COL applicant must address in its design scope as part of the COL application. For the offsite system, these should include, as a minimum, the requirements of General Design Criterion (GDC) 17 and 18 of 10 CFR Part 50, Appendix A, as well as the guidelines of IEEE 765-1983. Verification that GE provided the above design information in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.2.1.3-3.

Subsequently, during a November 18, 1992 meeting, GE indicated that SSAR interface requirements were not meant to define the totality of the requirements (including regulatory requirements) which a COL applicant's interfacing design must ultimately meet. The design interface requirements need only include those requirements that are needed by the ABWR design to assure that the design as completed by the COL applicant meets the in scope design basis requirements in the SSAR. The staff agrees with this assessment. COL applicant design will be required to meet all applicable regulatory requirements and associated regulatory and industry guidance as part of its COL application. The applicable regulatory requirements and associated regulatory and industry guidance are delineated in the code of federal regulations, the NRC SRP, and industry standards and codes. These requirements and guidance do not need to be repeated as interface requirements. DFSER (SECY-92-349) Confirmatory Item 8.2.1.3-3 is therefore resolved. The staff's evaluation of design interface requirements is also addressed in Section 8.2.1.3.1 of this report.

In interface requirement (2), Section 8.2.3.1 of the draft SSAR submittal dated April 3, 1992, indicated that a COL applicant who references the ABWR design is expected to establish the size of the unit auxiliary transformers to ensure a voltage dip of no more than 20 percent during motor starting. It was the staff's understanding, based on discussions with GE, that the sizing of the unit auxiliary transformers is within the ABWR scope of design responsibility. GE committed to revise the SSAR in a future amendment to reflect the above as part of GE's scope. Verification that GE included this change in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.2.1.3-4. GE included this change in Section 8.2.1.2 of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

In interface requirement (4), Section 8.2.3.1 of the draft SSAR submittal dated April 3, 1992, GE indicated that it is acceptable and recommended to normally power all three divisions of the Class 1E ac-distribution system from the normal preferred power source. This interface requirement is not consistent with design commitments that are currently documented in other sections of the SSAR (see Section 8.2.3.5 of this report). GE committed to delete this requirement (specifically, the second sentence of interface requirement (4) in Section 8.2.3.1 of the draft SSAR revision dated April 3, 1992) when the SSAR was revised in the future. Verification that GE included this change in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.2.1.3-5. GE included this change in Section 8.2.3 of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

In interface requirement (5), Section 8.2.3.1 of the draft SSAR submittal dated April 3, 1992, GE indicated that the two offsite circuits will be connected to different transmission systems. It was the staff's understanding, based on discussions with GE, that there is only one transmission system. GE committed to clarify the interface requirement in a future amendment to indicate that the main and reserve offsite power circuits will be connected to different transmission circuits or lines (rather than systems) and that the transmission circuits or lines will be independent and separate. Verification that GE provided the above design information in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.2.1.3-6. GE included this information in Section 8.2.3(5) of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

Section 8.2.3.1 of GE's draft SSAR submittal dated April 3, 1992, did not include additional interface requirements defining what is required of the COL applicant in order to have independent and separate transmission circuits and switching stations. GE committed to include in a future SSAR amendment explicit interface requirements defining independence and separation of transmission lines and switching stations or switchyards. Verification that GE revised the SSAR to reflect as a minimum the following design commitments in a future amendment was DFSER (SECY-92-349) Confirmatory Item 8.2.1.3-7:

- The two designated preferred power circuits will not have a common take-off structure or use common structures for support.

- The two lines from the transmission network that are designated as the preferred power circuits will be designed to minimize their simultaneous loss as a result of failure of any transmission tower or failure from crossing lines.

- The preferred power circuits originating from the transmission network will be designed to minimize their simultaneous failure as a result of failure of a single breaker, switchyard bus, switchgear bus, or cable.

- System studies will be performed to demonstrate that the preferred power supply will not degrade below a level consistent with the availability goals of the plant as a result of contingencies such as loss of any of the following elements

  - nuclear power generating unit
  - largest generating unit
  - most critical transmission circuit or intertie
  - largest load

- The interconnection between the transmission network and the switchyard will consist of a minimum of two transmission lines that are designated as the preferred power supply circuits. Where more than two lines are available from the transmission network, any combination of two lines may be designated and used as the preferred power supply circuits, provided that each combination of two circuits meets the offsite system design requirements.

- Switchyard equipment will be designed to adequately withstand stresses from the worst-case faults.

- The physical design of the switchyards will minimize the probability that a single equipment failure will cause the simultaneous or subsequent loss of both preferred power supply circuits.

GE indicated in response to this item that SSAR interface requirements were not meant to define the totality of the requirements (including the requirements needed to assure independence and separation between transmission circuits) which a COL applicant's interfacing design may ultimately have to meet. The staff agrees with this assessment. Under 10 CFR Part 52, the site-specific portion of a COL applicant's design will be required to meet all applicable regulatory requirements and associated regulatory and industry guidance as part of their COL application. The applicable regulatory requirements and associated regulatory and industry guidelines are delineated in the code of federal regulations, the NRC SRP, and the industry standards and codes. These requirements and guidelines, therefore, do not need to be repeated as interface requirements in ABWR SSAR. In addition, GE indicated that the above listed items that were identified in the DFSER (SECY-92-349) to assure independence between offsite circuits would be considered as part of their proposed conceptual design description to be included in the site-specific part of the SAR for a facility referencing the certified design. The staff concludes that interface requirements defining what is required of the COL applicant in order to have independent and separate transmission circuits and switching stations are not required to be included as interface requirements in the ABWR SSAR. DFSER (SECY-92-349) Confirmatory Item 8.2.1.3-7 is therefore resolved. The staff's evaluation of interface requirements relating to capacity, capability, and independence of offsite circuits is in Section 8.2.2 of this report. The staff's evaluation of GE's conceptual design for the offsite system is in Section 8.2.1.3.2 of this report.

In interface requirement (4), Section 8.2.3.1 of the draft SSAR submittal of April 3, 1992, GE indicated that the COL applicant will analyze incoming transmission lines to ensure that their expected availability is as good as assumed in performing the plant's probability risk analysis (PRA). Based on discussions with GE, the staff contended that the assumptions made in performing the plant's PRA are within the scope of the ABWR design. GE committed to explicitly state as part of this interface requirement, the expected availability of incoming transmission lines, in a future SSAR amendment. Verification that GE provided the above design information in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.2.1.3-8. GE included this information in Section 8.2.3(4) of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

Design commitments documented in the SSAR indicated that the conceptual design of the ABWR offsite preferred power system will include two separate and independent switching stations (or switchyards). Given that there will be two separate and independent switchyards, the interface requirements within individual switchyards presented in GE's draft submittal of April 3, 1992, go beyond industry-recommended practice for offsite preferred circuits. The staff understood that GE will revise the SSAR to describe the parts of the offsite system switchyard not subject to interface requirements specified in Section 8.2.3 of the draft submittal of April 3, 1992. Verification that GE provided the above design information in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.2.1.3-9. GE indicated that they would re-evaluate their proposed interface requirements and conceptual design for the portion of the offsite system that is outside of their scope of supply. Interface requirements and the conceptual design are evaluated in Sections 8.2.1.3.1 and 8.2.1.3.2 of this report respectively. On the basis of these evaluations, this item is resolved.

8.2.1.3.1 Interface Requirements

This section addresses, in part, the staff's evaluation of GE's response to DSER (SECY-91-355) Open Item 24 and DFSER (SECY-92-349) Confirmatory Items 8.2.1.3-3 and 8.2.1.3-9 and complements the resolution of these items.

GE revised their interface requirements included in Section 8.2.3.1 of GE's draft SSAR submittal of April 3, 1992, and moved interface requirements that were included in Section 8.2.4 of SSAR Amendment 21 to Section 8.2.3 of SSAR Amendment 33. In addition, GE provided a representative conceptual design for the portion of the offsite system outside of their scope of supply (see Section 8.2.1.3.2 of this report).

Section 52.47(a)(1)(vii) of 10 CFR dictates that interface requirements be provided (for those portions of the offsite power system design for which GE's application does not seek certification) that are sufficiently detailed to allow completion of the final safety analysis. The staff concludes that GE has provided interface requirements in Section 8.2.3 of SSAR Amendment 33. The design, therefore, meets the above defined requirements of Section 52.47(a)(1)(vii) of 10 CFR and is acceptable.

The staff's assessment of the adequacy of these interface requirements is addressed in Section 8.2.2 of this report.

8.2.1.3.2 Conceptual Design

This section addresses, in part, the staff's evaluation of GE's response to DSER (SECY-91-355) Open Item 24 and DFSER (SECY-92-349) Confirmatory Items 8.2.1.3-7 and 8.2.1.3-9 and complements the resolution of these items.

GE added a Section 8.2.5 to SSAR Amendment 33 to address their conceptual design for the offsite system outside of its design scope. The conceptual design will consist of two independent offsite circuits from the transmission network to the electrical systems for which GE is seeking certification. These offsite circuits, including their associated I&C circuits, will be independent. Each circuit will have sufficient capacity and capability. Each circuit will have redundant I&C circuits including their ac and dc power supplies. In addition, the total offsite power system will have an expected availability that is as good as the assumptions made in performing the plant PRA.

To aid the staff in its review of the SSAR and to permit assessment of the adequacy of the interface requirements, Section 52.47(a)(1)(ix) of 10 CFR dictates that a representative conceptual design be provided in the ABWR SSAR for those portions of the plant's offsite power system design for which the GE's application does not seek certification. The staff concludes that GE has provided a representative conceptual design described above. The design, therefore, meets the above defined requirements of Section 52.47(a)(1)(xi) of 10 CFR and is acceptable. GE has included the above described conceptual design in Section 8.3.5 of SSAR Amendment 33 which is acceptable.

The staff's assessment of the adequacy of the interface requirements with the aid of the above described conceptual design is addressed in Section 8.2.2 of this report.

Certified Design Material

The certified design material (CDM) for the GE ABWR Design Stage 2 Submittal, transmitted by letter dated April 6, 1992, did not specify the appropriate interface-related ITAAC material for the offsite systems outside the ABWR scope of design. In the ITAAC, (GE) must address this interface area. This was DFSER Open Item 8.2.1.4-1. GE provided a revised set of design descriptions and ITAAC. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. On the basis of this evaluation, this item is resolved.

8.2.2 Preferred Offsite Circuits Within the ABWR Scope of Design

This section addresses, in part, the staff's evaluation of GE's response to DSER (SECY-91-355) Open Item 24 and DSER (SECY-92-349) Confirmatory Item 8.2.1.3-7 and complements the resolution of these items.

The following portions of the preferred power circuits are within the scope of design of the ABWR standard plant:

- Preferred power circuit from the low-voltage terminals of the main transformer through the three unit auxiliary transformers to the input terminals of each of the three redundant, onsite Class 1E ac-distribution systems.

- Alternate preferred power circuit from the low-voltage terminals of the reserve auxiliary transformer to the input terminals of each of the three redundant, onsite Class 1E ac-distribution systems.

8.2.2.1 Physical Separation (Transformers and Circuits)

1. Physical Separation of Transformers

By its draft submittal dated April 3, 1992, GE revised the SSAR to indicate that the reserve auxiliary transformer will be separated from the unit auxiliary transformers by a minimum distance of 15 m (50 ft) and that each transformer will be provided with an oil collection pit and drain to a safe disposal area. In addition, GE indicated by letter dated April 6, 1992, that the reserve auxiliary transformer and its input feeders will be separated from the main power transformer and its input feeders and from the unit auxiliary transformers by a minimum of 15 m (50 ft). Section 9A.4.6 of SSAR Amendment 14 also indicated that the main, unit auxiliary, and reserve transformers will have oil collection pits and that each of these transformers will have automatic deluge water spray systems. This information was provided in response to DSER SECY-91-355 Open Item 23.

In order to assure that the COL applicant's design for offsite systems does not negate their proposed design for physical separation of transformers, GE indicated that the COL applicant's design will be required (i.e., by interface requirement) to be independent and compatible with the above described design commitments.

GDC 17 of 10 CFR Part 50, Appendix A, requires that provisions be included in the design of the offsite electric power system which will minimize to the extent practicable the likelihood of simultaneous failure of both normal and alternate offsite preferred power circuits under operating and postulated accident and environmental conditions. The staff concludes that a design which meets the above described commitments will provide reasonable assurance that the energy available from a credible failure of one of the offsite circuit transformers (such as, failure by fire or by explosion) will not propagate to the other offsite circuit and cause its failure. The oil collection pit will contain a transformer oil fire at the transformer. The automatic deluge water spray system will minimize the intensity of the heat that may be generated by a transformer fire. In addition, the 15 m (50 ft) of distance to other offsite circuit transformers will dissipate the energy from heat that may be generated by a transformer fire or by missiles that may be generated by transformer explosion.

Consequently, the design meets the above defined requirement of GDC 17 and is acceptable. Verification that GE incorporated the above commitments into a future Chapter 8.0 SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.2.2.1-1. GE has included these commitments in Sections 8.2.2.1(2) and 8.2.3(6) of SSAR Amendment 33, which is acceptable. Therefore, this item is resolved.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.2.2.1-1 related to the design description and the inspections, tests, and analyses, and acceptance criteria (ITAAC) for the physical separation of transformers. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

2. Physical Separation of Circuits

By its draft submittal dated April 3, 1992, in response to DSER Open Item 23, GE revised the SSAR to indicate that separation of the normal and alternate preferred power feeds within the turbine, control, and reactor buildings will be accomplished by floors and walls, except within the switchgear rooms where they must be routed to the same switchgear lineups. Based on discussions, GE further indicated that normal and alternate circuits in the switchgear rooms will be separated to the maximum extent feasible; that is, the circuits will be routed on opposite sides of the room and will be connected to the switchgear lineup on opposite ends. Also, based on discussions, GE indicated that the isolated phase bus duct and/or cables located outside the turbine, control, and reactor buildings that are affiliated with the normal preferred power circuits will be separated by a minimum of 15 m (50 ft) from the reserve auxiliary transformer. Likewise, the isolated phase bus duct and/or cables located outside the turbine, control, and reactor buildings that are affiliated with the alternate preferred offsite circuit will be separated by a minimum of 15 m (50 ft) from the unit auxiliary and main transformers.

In order to assure that the COL applicant's design for offsite system does not negate their proposed design for physical separation of circuits, GE indicated that the COL applicant's design will be required (i.e., by interface requirement) to be independent and compatible with the above described design commitments.

GDC 17 of 10 CFR Part 50, Appendix A, requires that provisions be included in the design of the offsite electric power system which will minimize to the extent practicable the likelihood of simultaneous failure of both normal and alternate offsite preferred power circuits under operating and postulated accident and environmental conditions.

The staff concludes that a design which meets the above described commitments will provide reasonable assurance that failure of one circuit will not propagate and cause failure of the other offsite circuit. Consequently, the design meets the above defined requirements of GDC 17 of 10 CFR Part 50, Appendix A, and is acceptable. Verification that GE has included appropriate changes in a future SSAR amendment to reflect the above commitments was DFSER (SECY-92-349) Confirmatory Item 8.2.2.1-2. GE has included these commitments in Section 8.2.1.3 and 8.2.3 of SSAR Amendment 33, which is acceptable. Therefore, this item is resolved.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.2.2.1-2 related to the design description and the ITAAC for the physical separation for offsite preferred circuit. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

8.2.2.2 Physical Separation of Instrumentation and Control (I&C) Cables for the Offsite Power System

Based on GE's draft submittal dated April 3, 1992, in response to DSER (SECY-91-355) Open Item 23 and subsequent discussions, GE indicated that instrumentation and control cables that are affiliated with the normal and alternate preferred offsite circuits will be separated as follows:

- The instrumentation and control cables that are affiliated with the normal preferred offsite circuit will be routed in raceways corresponding to the load group of their power source.
- The instrumentation and control cables that are affiliated with the alternate preferred offsite circuit will be routed in dedicated raceways. The alternate preferred offsite instrumentation and control circuit cables will not share raceways with any other cables.
- The separation between the normal and alternate preferred offsite instrumentation and control cables will be the same as the separation between the normal and alternate preferred offsite power circuits (that is: floors, walls, or 15 m (50 ft) of physical separation).

In order to assure that the COL applicant's design for offsite systems does not negate their proposed design for physical separation of I&C circuits, GE indicated that the COL applicant's design will be required (i.e., by interface requirement) to be independent and compatible with the above described design commitments.

GDC 17 of 10 CFR Part 50, Appendix A, requires that electric power from the transmission network to the onsite electric distribution system shall be supplied by two physically independent circuits designed and located so as to minimize to the extent practicable the likelihood of their simultaneous failure under operating and postulated accident and environmental conditions.

The staff concludes that a design which meets the above commitments will provide reasonable assurance that failure of one circuit will not cause the failure of the other circuit. The design, therefore, meets the above defined requirements of GDC 17 of 10 CFR Part 50, Appendix A, and is acceptable. Verification that GE has provided the above commitments in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.2.2.2-1.

GE included the above design commitments in Section 8.2.1.3 of an SSAR markup dated March 31, 1993, and Section 8.2.3 of SSAR Amendment 33 with the exception of I&C circuits located in the control room area and interlock circuitry required to prevent paralleling of the two offsite sources. GE indicated that these circuits are not separated by floors, walls, or 15 m (50 ft) but instead indicated that these circuits will be electrically isolated and will not be routed together in the same raceway. The staff concludes that a design which meets the above described commitments as revised will also provide reasonable assurance that failure of one circuit will not cause failure of the other circuits. Consequently, the revised offsite system design also meets the above defined requirements of GDC 17 of 10 CFR Part 50, Appendix A, and is acceptable. GE included the additional design provisions in Section 8.2.1.3 of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

During ITAAC development, GE recognized that the offsite system outside of their scope of design was unnecessarily restricted by design requirements stipulated within their scope of design. The requirements did not allow sharing the non-Class 1E dc systems within GE's scope of design between the independent offsite circuits. During ITAAC review, the staff agreed that an offsite system design which utilized the non-Class 1E systems within GE's scope of design and shared these systems between the independent offsite circuits could meet regulatory requirements and could thus be found acceptable. In order to remove this limitation, GE further revised Section 8.2.1.3 of its SSAR markup dated March 31, 1993, to indicate the following:

a. Instrumentation and control circuits at their dc power sources may be electrically isolated, and not routed together in the same raceway instead of being separated by floors, walls, or 15 m (50 ft).

b. Instrumentation and control circuits for the independent offsite circuits will not rely on a single common dc power source.

The staff agrees that these revisions will allow utilization of the non-Class 1E dc systems within GE's scope of design and sharing of these dc systems between independent offsite circuits within a COL applicant's offsite system design.

If non-Class 1E dc power systems within GE's scope of design are utilized as the dc source for instrumentation and control in a COL applicant's proposed offsite system design and these dc power sources are shared between the independent offsite circuits, the staff concludes above a design which meets the above described commitments as further revised will also provide reasonable assurance that failure of one circuit will not cause failure of the other circuit.

Consequently, the design meets the above defined requirements of GDC 17 of 10 CFR Part 50, Appendix A, and is acceptable.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.2.2.2-1 related to the design description and the ITAAC for the physical separation of power, instrumentation, and control cables for the offsite preferred power system. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

8.2.2.3 Electrical Independence

Based on GE's draft SSAR submittal dated April 3, 1992, in response to DSER (SECY-91-355) Open Item 23 and discussions, GE indicated that there will be no electrical interconnections between the normal and alternate preferred power, instrumentation, and control circuits except where the power circuits connect to common Class 1E and non-Class 1E switchgear lineups. At the common switchgear, one open and one closed circuit breaker will maintain the electrical independence. These circuit breakers will be interlocked so that the closed breaker must be opened before the open breaker can be closed. Transfer from normal to alternate (or alternate to normal) preferred power circuits will be manual. Instrumentation and control circuits (including their power supply) that are affiliated with the normal preferred offsite circuit will be electrically independent from (that is, they will be electrically isolated from or will have no electrical interconnection with), the instrumentation and control circuits (including their power supply) affiliated with the alternate preferred power supply.

In order to assure that the COL applicant's design for offsite systems does not negate their proposed design for electrical independence, GE indicated that the COL applicant's design will be required (i.e., by interface requirement) to be independent and compatible with the above described design commitments.

GDC 17 of 10 CFR Part 50, Appendix A, requires that electric power from the transmission network to the onsite electric distribution system shall be supplied by independent circuits designed and located so as to minimize to the extent practicable the likelihood of their simultaneous failure under operating and postulated accident and environmental conditions.

The staff concludes that a design which meets the above described design commitments will provide reasonable assurance that failure of one circuit will not cause failure of the other circuit. Consequently, the design meets the above defined requirements of GDC 17 of 10 CFR Part 50, Appendix A, and is acceptable.

Verification that GE provided the above design commitments in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.2.2.3-1. GE included these commitments in Sections 8.1.3.1.1.1, 8.2.3, and 8.2.1.3 of SSAR Amendment 33 which is acceptable. Therefore, this item is resolved.

As indicated above in Section 8.2.2.2 of this report, GE further revised Section 8.2.1.3 of SSAR markup dated March 31, 1993, to remove design restrictions within its scope of design which did not allow sharing of the non-Class 1E dc systems within GE's scope of design between the independent offsite circuits.

If non-Class 1E dc power systems within GE's scope of design are utilized as the dc power source for instrumentation and control in a COL applicant's proposed offsite system design and these dc power sources are shared between the independent offsite circuits, the staff concludes that a design which meets the above described commitments as revised will also provide reasonable assurance that failure of one circuit will not cause failure of the other circuit.

Consequently, the design meets the above defined requirements of GDC 17 of 10 CFR Part 50, Appendix A, and is acceptable.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.2.2.3-1 related to the design description and the ITAAC for the electrical independence. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

8.2.2.4 Testing for the Offsite Power System

Based on GE's draft submittal dated April 3, 1992, in response to DSER (SECY-91-355) Open Item 25 and discussions, GE indicated that all systems, equipment, and components that are affiliated with the normal and alternate offsite preferred power circuits within the GE scope of design — except generator breakers — will have the capability to be tested periodically during normal plant operation. Based on this commitment, GE indicated that the design will permit verification of the following offsite power system capabilities:

- The generator breaker can open on demand.
- The instrumentation, control, and protection systems, equipment, and components that are affiliated with the normal and alternate offsite preferred circuits are properly calibrated and perform their required functions.
- All required Class 1E and non-Class 1E loads can be powered from their designated preferred power supply within the capacity and capability margins specified in the SSAR for the offsite system circuits.
- The loss of the offsite preferred power supply can be detected.
- Transfer between preferred power supplies can be accomplished.
- The batteries and chargers that are affiliated with the preferred power system can meet the requirements of their design loads.

In addition, GE indicated that the design of high- and medium-voltage bus ducts and cables provide ready access for regularly inspecting, cleaning, and tightening terminals, and for inspecting and cleaning insulators. The bus duct design also includes provisions for excluding debris and fluids, and for draining condensate.

GDC 18 of 10 CFR Part 50, Appendix A, requires that the offsite electric power systems be designed (1) to permit appropriate periodic inspection and testing of important areas and features (such as wiring, insulation, connections, and switchboards) to assess the continuity of the systems and the condition of their components, (2) the capability to test periodically the operability and functional performance of the components of the systems, and (3) the capability to test periodically the operability of the systems as a whole and (under conditions as close to design as practical) the full operation sequence that brings the systems into operation.

The staff concludes that a design which meets these testability requirements will ensure that the offsite electric power systems, equipment, and components will be designed with the capability to be periodically tested. Consequently, the design meets the above defined requirements of GDC 18 of 10 CFR Part 50, Appendix A, and is acceptable.

Verification that GE provided the above testability requirements in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.2.2.4-1. GE included these testability requirements in Section 8.2.2.1(3) of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

With regard to periodic testing of the systems, equipment, and components, GE indicated the following:

- Periodic verification will ensure that the normal and alternate offsite power circuits are energized and connected to the appropriate Class 1E distribution system division at least once every 12 hours.

- Tests and inspections will be performed at appropriately scheduled intervals for each of the items highlighted above.

- The test and inspection intervals will be established and maintained according to industry recommended practice defined in Section 6.5, "Test Intervals," of IEEE 338-1977.

Verification that GE specified in a future SSAR amendment a statement that the COL applicant must include the above specified periodic tests and inspections in appropriate plant procedures was DFSER (SECY-92-349) COL Action Item 8.2.2.4-1. GE included this action item in Section 8.2.4.1 of SSAR Amendment 32, which is acceptable.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.2.2.4-1 was related to the design description and the ITAAC for testing for the offsite power system. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

8.2.2.5 Generator Breaker

The low-voltage generator breaker must open on a turbine trip to maintain the normal preferred power supply to the safety buses. The generator breaker cannot be tested during normal plant operation without tripping the main turbine generator.

Based on discussions, GE indicated that the opening of the generator breaker to establish the normal offsite preferred power circuit to safety buses will be verified each time the reactor is shutdown (at intervals in accordance with the plant's technical specifications (TS)). In addition, based on information presented in the draft SSAR submittal of April 3, 1992, GE indicated that there are industry published test results showing a reliability number of 0.9967 for the generator breaker.

The draft SSAR submittal of April 3, 1992, also indicated that during all modes of plant operation (including shutdown, refueling, startup, and run), the normal preferred power supply will be connected to two of the three safety buses, and the alternate preferred power supply will be connected to one of the three safety buses. If the normal preferred supply is lost because the generator breaker fails to open, offsite power will still be available immediately through the alternate preferred power supply to one of the three safety buses. It will also be available on a delayed basis (within minutes by manual action from the control room) to the two other safety buses through the alternate preferred power supply.

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. GDC 18 of 10 CFR Part 50, Appendix A, requires that the offsite electric power systems be designed (1) to permit appropriate periodic inspection and testing of important areas and features (such as wiring, insulation, connections, and switchboards) to assess the continuity of the systems and the condition of their components, (2) the capability to test periodically the operability and functional performance of the components of the systems, and (3) the capability to test periodically the operability of the systems as a whole and (under conditions as close to design as practical) the full operation sequence that brings the systems into operation.

The staff concludes that a design which meets the above described provisions, will minimize to the extent feasible the likelihood of simultaneous failure of both normal and alternate offsite preferred power circuits and will be testable. Consequently, the design meets the above defined requirements of GDC 17 and 18 of 10 CFR Part 50, Appendix A, and is acceptable.

Verification that GE specified in a future SSAR amendment a statement that the COL applicant must include surveillance requirements for generator breakers and the above design provisions in appropriate plant procedures was DFSER (SECY-92-349) COL Action Item 8.2.2.5-1. GE included this action item in Sections 8.2.4.1(7) and 8.3.4.9 of SSAR Amendment 32, which is acceptable.

8.2.2.6 Capacity and Capability of the Offsite Power System

Based on discussions with GE and information presented in the April 3, 1992, draft submittal of SSAR Chapter 8.0, in response to DSER (SECY-91-355) Open Item 26, GE indicated that the offsite power system will be designed to provide the following capacity and capabilities:

- Each circuit of the preferred power supply will be designed to provide sufficient capacity and capability to power equipment required to ensure that
  - fuel design limits and design conditions of the reactor coolant pressure boundary will not be

- exceeded as a result of anticipated operational occurrences

- in the event of plant design-basis accidents, the core will be cooled, and containment integrity and other vital functions will be maintained

- When used for normal operation, each preferred power supply will be sized to supply the maximum expected coincident Class 1E and non-Class 1E loads.

- The secondary winding of the reserve auxiliary transformer, which supplies the Class 1E load groups, will have an oil/air rating greater than or equal to the combined load of the three Class 1E load groups.

- The normal and alternate offsite preferred power circuits will be designed with sufficient capacity and capability to limit variations of the operating voltage of the onsite power distribution system to a range appropriate to ensure

  - normal and safe steady-state operation of all plant loads

  - starting and acceleration of the limiting drive system with the remainder of the loads in service

  - reliable operation of the control and protection systems under conditions of degraded voltage

  Specifically, when measured at the load terminals, the voltage variation at any voltage level will not exceed the following limits

  - plus or minus 10 percent of the load-rated voltage during all modes of steady-state operation

  - minus 20 percent of the motor-rated voltage during motor starting

- Voltage levels at the low-voltage terminals of the auxiliary and reserve transformers will be analyzed to determine the maximum and minimum load conditions that are expected throughout the anticipated range of voltage variations of the offsite transmission system and the main generator. Separate analyses will be performed for each possible circuit configuration of the offsite power supply system.

- During their operation, normal and alternate preferred power circuits are subject to environmental conditions (such as, wind, ice, snow, lightning, temperature variations, or flood). These circuits will be designed in accordance with industry-recommended practice in order to minimize the likelihood that they will fail while operating under the environmental conditions to which they are subject.

- During their operation, normal and alternate preferred power circuits can be subjected to the transmission system's steady-state and transient conditions (such as switching and lightning surges, maximum and minimum voltage ranges for heavy and light load conditions, frequency variation, or stability limits). The preferred power circuits will be designed such that these conditions will not subject the onsite Class 1E systems, equipment, and components to conditions that are beyond the limits for which they are designed and qualified. These design considerations apply to all onsite Class 1E loads and systems that use the services of the preferred power supply during startup, normal operation, safe shutdown, accident, and post-accident operation. The staff's evaluation of this item is addressed in Section 8.2.3.8 of this report.

- Performance and operating characteristics of the normal and alternate preferred power circuits will be required to meet operability and design-basis requirements. These requirements include but are not limited to; the ability to withstand short-circuits, equipment capacity, voltage and frequency transient response, voltage regulation limits, step load capability, coordination of protective relaying, and grounding.

- The generator circuit breaker will be designed to withstand the maximum root mean squared and crest momentary currents. Further, the breaker will be designed to interrupt the maximum asymmetrical and symmetrical currents determined to be produced by a three-phase fault at the location that results in the maximum fault currents.

- The main step-up transformers and the unit auxiliary and reserve transformers will be designed and constructed to withstand the mechanical and thermal stresses produced by external short circuits. In addition, these transformers will meet the corresponding requirements of the latest revisions of ANSI C57.12.00, "General Requirements for Liquid-Immersed Distribution, Power, and Regulating Transformers."

- Circuit breakers and disconnecting switches will be sized and designed according to the latest revision of ANSI C37.06, "Preferred Ratings and Related Capabilities for ac High-Voltage Circuit Breakers Rated on a Symmetrical Current Basis."

In order to assure that the COL applicant's design for offsite systems does not negate their proposed design for capacity and capability of the offsite system, GE indicated that the COL applicant's design will be required (by interface requirements) to meet the following:

- Voltage variations of the offsite transmission network during steady state operation will not cause voltage variations at the loads of more than plus or minus 10 percent of the loads nominal ratings.
- The normal steady state frequency of the offsite transmission network will be within plus or minus 2 hertz of 60 hertz during recoverable periods of system instability.
- The offsite transmission circuits from the transmission network through and including the main step-up power and reserve auxiliary transformers will be sized to supply their load requirements, during all design operating modes, of their respective Class 1E divisions and non-Class 1E load groups.
- The impedance of the main step-up power and reserve transformers will be compatible with the interrupting capability of the plant's circuit interrupting devices.
- Instrumentation and control system loads will be compatible with the capacity and capability design requirements of dc systems within the ABWR standard plant scope.

GDC 17 of 10 CFR Part 50, Appendix A, requires that the offsite electric power system have sufficient capacity and capability to permit safety systems to perform their required safety function.

The staff concludes that a design which meets the above described commitments and characteristics will have sufficient capacity and capability to ensure that:

- Specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary will not be exceeded as a result of anticipated operational occurrences.
- The core will be cooled, and containment integrity and other vital functions will be maintained in the event of postulated accidents.

The design, therefore, meets the above defined requirements of GDC 17 of 10 CFR Part 50, Appendix A, and is acceptable.

Verification that GE provided the above noted design commitments and characteristics in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.2.2.6-1. GE included this information in Sections 8.1.2.1, 8.2.1.2, 8.2.2.1(2), and 8.2.2.1(8) of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved. (Note: A portion of this confirmatory item is discussed in Section 8.2.3.8 of this report.)

In response to DFSER (SECY-92-349) COL Action Item 8.2.2.6-1, GE clarified operational restrictions as follows for the offsite system:

- Operational restrictions will require the reserve auxiliary transformer to supply one of three Class 1E load groups during normal plant operation.
- Operational restrictions will assure that the forced oil/air ratings of the reserve auxiliary transformer, or any unit auxiliary transformer, will not be exceeded under any operating mode.
- Continued plant operation will be appropriately limited when one of the three unit auxiliary transformers is inoperable (that is, when one of the three safety buses will not have access to both normal and preferred offsite circuits).
- Continued plant operation will be appropriately limited when the reserve auxiliary transformer is inoperable.

GE included the above listed aspects of this action item in Sections 8.2.4.2, 8.2.4.5, and 8.3.4.9 of SSAR Amendment 32, which is acceptable.

As indicated above in Section 8.2.2.2 of this report, GE further revised Section 8.2.1.3 of SSAR markup dated March 31, 1993, to remove design restrictions within its scope of design which did not allow sharing of the non-Class 1E dc systems within GE's scope of design between the independent offsite circuits.

If non-Class 1E dc power systems within GE's scope of design are utilized as the dc power source for instrumentation and control in a COL applicant's proposed offsite system design and if these dc power sources are (or are not) shared between the independent offsite circuits, the staff concludes that a design which meets the above design commitments and characteristics as revised will also have sufficient capacity and capability to ensure that:

- Specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary will not be exceeded as a result of anticipated operational occurrences.

- The core will be cooled, and containment integrity and other vital functions will be maintained in the event of postulated accidents.

The design, therefore, meets the above defined requirements of GDC 17 of 10 CFR Part 50, Appendix A, and is acceptable.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.2.2.6-1 was related to the design description and the ITAAC for the capacity and capability of the offsite power system. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved. (Note: This open item is also addressed in Section 8.2.3.8 of this report).

8.2.2.7 Grounding and Lightning Protection

This section was moved to section 8.2.3.6 of this report.

8.2.2.8 Compliance With IEEE 765

Compliance with the guidelines of IEEE 765-1983 is addressed in Section 8.3.1.1 of this report.

8.2.2.9 Alternate Source of Power for Non-Safety Loads (NRC Policy Issue SECY-91-078, Section II.B of Enclosure 1 to SECY-93-087)

In SECY-91-078, the staff recommended that the Commission approve its position that an evolutionary plant design should include an alternate power source to the non-safety loads unless it can be demonstrated that the design margins will result in transients for a loss of non-safety power event that are no more severe than those associated with the turbine-trip-only event in current existing plant designs. In its August 15, 1991, SRM, the Commission approved the staff's position. The staff in NUREG-1242, "NRC Review of Electric Power Research Institute's Advanced Light Water Reactor Utility Requirements Document," Volume 2 (August 1992), clarified the intent of this position. Further, the staff has evaluated the electrical design for the ABWR relative to the provision of an alternate source of power for non-safety loads against the following proposed applicable regulation, which is based on the position included in NUREG-1242: "The electric power system of the standard design must include an alternate power source that is provided to a sufficient string of non-safety loads so that forced circulation could be maintained, and the operator has available the complement of non-safety equipment that would most facilitate the ability to bring the plant to a stable shutdown condition, following a loss of the normal power supply and plant trip."

GE revised the SSAR to indicate that the ABWR reserve auxiliary transformer provides the required alternate power source. The staff concluded that the ABWR design meets the requirements of this applicable regulation, and is acceptable. GE included this information in Section 8.3.1.2(4)(b) of SSAR Amendment 32.

8.2.3 Independence Between Offsite and Onsite Power Systems, Equipment, and Components

The preferred power supply furnishes electric power from the offsite system's transmission network (a common source of electric power) to redundant, onsite Class 1E systems, equipment, components, and loads. This common source of electric power may be used during all modes of plant operation to supply power to redundant Class 1E load groups. Because redundant load groups are powered from a common power source, they can be subjected to conditions which may cause their common failure due to single events or failures of this single source of electric power. This section of the staff's evaluation addresses ABWR design provisions for minimizing the probability of: (1) common mode failure of redundant onsite Class 1E systems due to single events or failures of this common source of electric power; and, (2) failure of the onsite system from causing loss of the offsite system in accordance with the requirements of GDC 17 of 10 CFR Part 50, Appendix A. The staff's DSER (SECY-91-355) addressed this part of the design in Open Item 28.

The following sections discuss the areas addressed in the staff's evaluation of ABWR design provisions intended to ensure an adequate level of independence between offsite and onsite systems.

8.2.3.1 Independence Between Offsite Circuits and Onsite Class 1E dc Systems

DC control, protection, and instrumentation power for offsite circuits, (originally proposed for the ABWR design), were derived from the Class 1E dc system through dc to dc converters that GE considered isolation devices. By the draft SSAR revision of April 3, 1992, GE eliminated all electrical interconnections between the offsite control, protection, and instrumentation circuits and the onsite Class 1E dc systems, equipment, and components.

GE indicated that the offsite system circuits will derive their control, protection, and instrumentation power from a non-Class 1E dc system that is independent of the onsite Class 1E dc system. 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 power from the onsite electric power supplies. The staff concludes that a design which meets the above described design commitment (1) will provide reasonable assurance that failure in the offsite system will neither challenge nor possibly cause the loss of onsite Class 1E dc systems, (2) will provide reasonable assurance that any single failure of a Class 1E dc system or its component parts will not cause loss of offsite power, and (3) will provide reasonable assurance that common-cause failure will not occur between offsite and onsite power sources that are affiliated with a single load group. Therefore, the design meets the above defined requirements of GDC 17 of 10 CFR Part 50, Appendix A, and is acceptable.

Verification that GE provided the above design commitments in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.2.3.1-1. GE has included these commitments in Section 8.2.1.2 of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.2.3.1-1 was related to the design description and the ITAAC for the electrical independence between offsite circuits and onsite Class 1E DC system. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

8.2.3.2 Independence During Loss of or Degraded Offsite Voltage

The ABWR design incorporates two distinct levels of protection to ensure the independence of offsite and onsite systems during loss of offsite voltage or degraded offsite voltage conditions:

(1) Loss of Offsite Voltage (First Level of Protection)

In the draft SSAR revision of April 3, 1992, Section 8.3.1.1.7(1), GE indicated that the onsite Class 1E systems will be normally energized from the offsite normal and alternate preferred power system. Should the voltage on the Class 1E bus decay to less than 70 percent of its nominal rated value for a predetermined time, a bus transfer will be initiated. As a result of this transfer, the onsite Class 1E buses will be transferred so that they are powered by the onsite standby diesel generators (rather than the offsite normal and alternate preferred power systems). Upon initiation of a bus transfer, a signal will be generated to open the offsite supply breaker to the Class 1E bus.

In the draft SSAR revision of April 3, 1992, the first paragraph of Section 8.3.1.1.7 indicated that the time delay for bus transfer initiation will change (be reduced) from 3 to 0.4 seconds if a loss-of-coolant accident (LOCA) signal is present when a loss of preferred power (LOPP) occurs. (The 3 and 0.4 second time delay setpoints for initiation of bus transfer are considered to be approximations of the actual setpoint that will be in effect during plant operation. The actual setpoints will be established as part of an overall system voltage and load analysis.) Discussions with GE confirmed that the proposed ABWR design will include a time delay relay to establish the predetermined time before bus transfer initiation. When only a LOPP signal is present, the time delay relay will be set for 3 seconds. When both LOPP and LOCA signals are present, the time delay relay will be set for 0.4 seconds. The purpose of the time delay is to reduce unnecessary transfer from offsite to onsite sources during offsite power system transients. If voltage on the offsite system drops below 70 percent for less than 3 seconds (or 0.4 seconds when both LOCA and a LOPP signals are present), both the LOPP and time delay relay will reset after voltage recovery and transfer will not be initiated.

(2) Degraded Offsite Voltage (Second Level of Protection)

In draft SSAR revision of April 3, 1992, Section 8.3.1.1.7(8), GE indicated that when the bus voltage degrades to 90 percent or less of its rated value and remains degraded throughout a time delay, an undervoltage condition will be annunciated in the control room. Simultaneously, a 5-minute timer will be started, allowing the operator to take necessary corrective action. (The 5 minute time delay setpoints for initiation of bus transfer are considered to be approximations of the actual setpoint that will be in effect during plant operation. The actual setpoints will be established as part of an overall system voltage and load analysis.) After 5 minutes, the feeder breaker affected by the degraded voltage will be tripped. Should a LOCA occur during the 5 minute time delay, the affected feeder breaker will be tripped immediately.

In draft SSAR submittal of April 3, 1992, Section 8.3.1.2.1(3)(d), GE indicated that the design for degraded offsite voltage will meet the guidelines of Branch Technical Position, Power Systems Branch 1.

GE indicated that the ABWR electric system design will comply with the guidelines of IEEE 308-1980 and IEEE 603-1980, as specified by Section 8.3.1.2.1(2)(c) of the April 3, 1992, draft SSAR revision and Section 1.8.2 (Tables 1.8-20 and 1.8-21) of SSAR Amendment 17. Because both levels of protection are required to support safety-related systems, the staff concluded that GE's commitment to IEEE standards indicates that the electric system design will meet the requirements of Section 5.2 of IEEE 308-1980. Section 5.2 of IEEE 308-1980 requires that components, equipment, or systems required to provide some protective action, such as containment integrity protection, or utilized to provide isolation protection are covered by all safety system requirements which are defined in IEEE 603-1980. The systems described above protect safety-related electrical systems from degraded or loss of voltage conditions. Thus, they provide some protective action and are covered by all safety system requirements defined in IEEE 603-1980. The staff, therefore, concluded that GE's commitment to IEEE standards indicates that systems, equipment, and components included in the design for both the first and second levels of voltage protection will meet all requirements of IEEE 603-1980.

GE's expressed commitment to IEEE Standards also applies to safety-related equipment (including the reactor trip system, engineered safety features, auxiliary supporting features, and other auxiliary features equipment) that requires ac power from the offsite system to perform safety functions. Specifically, GE indicated that such safety-related equipment will be designed and qualified (by type test, previous operating experience, analysis, or any combination of these three methods) to be capable of performing their required safety functions before, during, and after the following design basis operating conditions:

- voltages at the load at either +10 percent or -10 percent of the nominal voltage rating
- for 5 minutes with voltages at the load at 70 percent of the nominal voltage rating
- for 3 seconds with voltages at the load below 70 percent (e.g., for 3 seconds at 35 percent) of the nominal voltage rating

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. The staff concludes that a design which meets the above commitments will provide reasonable assurance that the common failure of both offsite and onsite systems will not occur as a result of loss of or degraded voltage conditions. The design, therefore, meets the above defined requirements of GDC 17 of 10 CFR Part 50, Appendix A and is acceptable.

Verification that GE provided the above design commitments in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.2.3.2-1. GE included this information in Section 8.3.1.1.7 of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

GE also indicated that both first- and second-level voltage protection will be tested periodically. Verification that GE included a COL action item to include these periodic tests in appropriate plant procedures was DFSER (SECY-92-349) COL Action Item 8.2.3.2-1. GE included this action item in Section 8.3.4.20 of SSAR Amendment 32, which is acceptable.

During the process of verifying inclusion in the SSAR of the above design information and periodic tests, the staff noted and expressed the concern that the diesel generator may be overloaded if there is a LOCA with a delayed LOPP. Section 8.3.1.1.7(3) of SSAR Amendment 21, LOPP following LOCA, indicates that all loads that have been connected to the Class 1E bus and are operating from the offsite power supply in response to the LOCA will be connected in one block to the standby diesel generator (operating at no load due to the LOCA) if there is a LOPP. The diesel generator may not have sufficient capacity to supply, in one block, the loads which may be operating on the Class 1E bus at the time of the LOPP. Because this item involved the capacity of the diesel generator, it has been moved to Section 8.3.8.4 of this report which addresses the capacity of the diesel generator.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.2.3.2-1 related to the design description and the ITAAC for the electrical independence during loss of, or degraded, offsite voltage. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

8.2.3.3 Independence During Parallel Operation of the Offsite and Onsite Systems During Periodic Load Tests of the Diesel Generator

(1) LOCA During Parallel Operation

Section 8.3.1.1.7(5) of SSAR Amendment 4 states that if a LOCA occurs when the diesel generator is being operated in parallel with the preferred power source during testing, and the test is being conducted from the local control panel, control must be returned to the main control room or the test operator must trip the diesel generator breaker. GE subsequently revised Section 8.3.1.1.7(4) of SSAR Amendment 4 in Section 8.3.1.1.7(5) of the draft SSAR revision dated April 3, 1992 (see the response to Question 435.19). Through that revision, GE changed the design commitment to indicate that if a LOCA occurs when the diesel generator is being operated in parallel with the offsite system, the diesel generator will automatically be disconnected from the 6.9 kV emergency bus, regardless of whether the test is being conducted from the local control panel or the main control room.

In addition, Section 8.3.1.1.8.8 of the draft SSAR revision dated April 3, 1992, indicated that the ABWR design will include interlocks to the LOCA and LOPP sensing circuits, in order to terminate parallel operation and cause the diesel generator to automatically revert to its standby mode if either a LOCA or a LOPP signal appears during a test. GE further indicated that the interlock design will have the capability to be tested periodically.

The ABWR design will include provisions for automatic switchover from system test mode to operating mode in case of either an accident signal or a loss of preferred offsite power signal, regardless of whether the test is being conducted from the local control panel or the main control room. 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. The staff concludes that a design which meets the above described commitments will provide reasonable assurance that loss of electric power from the offsite and onsite systems will not occur due to loss of power from the nuclear power unit due to a LOCA. The design, therefore, meets the above defined requirements of GDC 17 of 10 CFR Part 50, Appendix A, and is acceptable.

Verification that GE provided the above design commitments in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.2.3.3-1. GE has included these commitments in Sections 8.3.1.1.7(5), 8.3.1.1.7(6), and 8.3.1.1.8.8 of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

GE also indicated that the interlock design (which terminates parallel operation and causes the diesel generator to automatically revert to its standby mode) will periodically be tested. Verification that GE would include a COL action item to include this periodic test in appropriate plant procedures was DFSER (SECY-92-349) COL Action Item 8.2.3.3-1. GE included this action item in Section 8.3.4.21 of SSAR Amendment 32, which is acceptable.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.2.3.3-1 was related to the design description and the ITAAC for LOCA during parallel operation. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

(2) LOPP During Parallel Operation

In Section 8.3.1.1.7(6) of the draft SSAR revision dated April 3, 1992, GE indicated that the diesel generator circuit breaker will automatically trip on overcurrent if the offsite power supply is lost during the diesel generator paralleling test. In addition, Section 8.3.1.1.8.8 of that revision indicated that interlocks to the LOPP sensing circuits will terminate a parallel operation test, causing the diesel generator to automatically revert to its standby mode if a LOPP signal appears during a test.

When a standby power supply is being operated in parallel with the preferred power supply, Section 5.1.4.3 of IEEE 741-1986 requires that protection be provided to separate the two supplies if either degrades to an unacceptable level. However, this protection shall neither lock out nor prevent the availability of the power supply that is not degraded. In addition, Section 6.2.4.6.3 of IEEE 308-1992 requires provisions to detect a LOPP during testing, when the standby generator is connected to the offsite power source.

The staff concluded that a design complying with these industry-recommended IEEE practices will minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power from the transmission network. This design meets the above defined requirements of GDC 17 of 10 CFR Part 50, Appendix A, and is acceptable.

Based on GE's design commitment that interlocks to the LOPP sensing circuits will be included in the ABWR design to terminate a parallel operation test and cause the diesel generator to automatically revert to its standby mode if LOPP signal appears during a test, the staff concluded that the ABWR design meets the above described industry-recommended interlock practice.

The staff concludes that a design that meets the above described design commitments will provide reasonable assurance that electric power from the onsite diesel generator supplies will not be lost due to loss of the offsite transmission network. The design, therefore, meets the above defined requirements of GDC 17 of 10 CFR Part 50, Appendix A, and is acceptable.

Verification that GE provided the above design commitments in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.2.3.3-2. GE has included these commitments in Section 8.3.1.1.7(6) of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.2.3.3-2 was related to the design description and the ITAAC for LOPP during parallel operation. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

(3) Duration of Parallel Operation

Table 1.8-21 of SSAR Amendment 17 and Section 1.8.2 of SSAR Amendment 12 indicated that the ABWR design complies with IEEE 308-1980. Based on this statement of compliance, GE indicated that the ABWR design will satisfy Section 6.1.3 of IEEE 308-1980, which requires that the design minimize the duration of the connection between the preferred and standby power supplies. In addition, Section 8.3.1.1.8.1 of the draft SSAR revision dated April 3, 1992, indicated that the ABWR design requires that each diesel generator set be operated independently of the other sets and be connected to the utility power system only by manual control during testing or for bus transfer.

The staff concludes that a design which meets these commitments will 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. Consequently, this design meets the above defined requirements of GDC 17 of 10 CFR Part 50, Appendix A, and is acceptable.

Verification that GE included in a future SSAR amendment an action item that the COL applicant include the above commitments in appropriate plant procedures was DFSER (SECY-92-349) COL Action Item 8.2.3.3-2. GE included this action item in Section 8.3.4.21 of SSAR Amendment 32, which is acceptable.

(4) Diesel Generator Protective Relaying with the Diesel Generator Operating in Parallel with the Offsite System

Section 8.3.1.1.6.4 of the draft SSAR revision dated April 3, 1992, indicated that protective relaying of the diesel generator (generator differential, engine overspeed, low jacket water pressure, loss of excitation, anti-motoring (reverse power) overcurrent voltage restraint, high jacket water temperature, and low lube oil pressure) will be used to protect the machine when it is operated in parallel with the normal power system during periodic tests. In addition, Section 8.3.1.1.6.2 of the draft SSAR revision dated April 3, 1992, indicated that each diesel generator will be high-resistance grounded to maximize availability.

Section 8.3.1.2.1(2)(c) of the draft SSAR revision dated April 3, 1992, and Section 1.8.2 (Tables 1.8-20 and 1.8-21) of SSAR Amendment 17 indicated that the ABWR electric system design will comply with the requirements of IEEE 308-1980 and IEEE 603-1980. Through this commitment to these IEEE standards, GE indicated that the electric system design will satisfy the requirements of Section 5.2 of IEEE 308-1980, and that systems, equipment, and components included in the design for protective relaying will satisfy all requirements of IEEE 603-1980. Because protective relaying is required to minimize the likelihood of simultaneous loss of both offsite and onsite sources during testing, these components are required to satisfy all requirements of IEEE 603-1980 during parallel operation of offsite and onsite power supplies.

The staff concludes that a design which meets the above described commitments will provide additional assurance that there will not be a common failure between onsite and offsite power supplies during testing. The design, therefore, meets the above defined requirements of GDC 17 of 10 CFR Part 50, Appendix A, and is acceptable.

Verification that GE provided the above design commitments in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.2.3.3-3. GE included this information in Sections 8.3.1.1.6.2 and

8.3.1.1.6.4 of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

GE also indicated that protective relaying of the diesel generator will be tested periodically. Verification that GE included in a future SSAR amendment a statement that the COL applicant will include these periodic tests in appropriate plant procedures was DFSER (SECY-92-349) COL Action Item 8.2.3.3-3. GE included this action item in Section 8.3.4.22 of SSAR Amendment 32, which is acceptable.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.2.3.3-3 was related to the design description and the ITAAC for the diesel generator protective relaying when the diesel generator is operating in parallel with the offsite system. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

(5) Synchronizing Interlocks

Based on discussions, GE indicated that the ABWR design will meet the guidelines of Section 5.1.4.2 of IEEE 741-1986 which requires synchronizing interlocks to prevent incorrect synchronization whenever a standby power source is required to operate in parallel with the preferred power supply. GE also indicated that the synchronizing interlocks will have the capability to be tested periodically.

The staff concludes that a design complying with the above described commitments will reduce the likelihood of simultaneous loss of both offsite and onsite power supplies as a result of synchronization. The design, therefore, meets the above defined requirements of GDC 17 of 10 CFR Part 50, Appendix A, and is acceptable.

Verification that GE provided the above design commitments in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.2.3.3-3. GE included this information in Section 8.3.1.1.6.4 of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

GE also indicated that synchronizing interlocks will be tested periodically. Verification that GE included in a future SSAR amendment a statement that the COL applicant will include these periodic tests in appropriate plant procedures was DFSER (SECY-92-349) COL Action Item 8.2.3.3-4. GE included this action item in Section 8.3.4.23 of SSAR Amendment 32, which is acceptable.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.2.3.3-4 was related to the design description and the ITAAC for the synchronizing interlocks. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

8.2.3.4 Independence of Safety Systems During Operation or Failure of Non-Class 1E Loads (NRC Policy Issue SECY-91-078, Section II.B of Enclosure 1 to SECY-93-087)

The ABWR design must minimize the effects that operation or failure of non-Class 1E loads may have on Class 1E systems. To achieve this design objective, the staff took the position that the Class 1E system should be connected directly to a winding of the offsite power system's transformers that is separate from the winding that feeds the non-Class 1E loads. In other words, Class 1E and non-Class 1E loads should not be powered from the same transformer winding.

In SECY-91-078, the staff recommended that the Commission approve this position for evolutionary plant designs. The evolutionary plant design should include at least one offsite circuit to each redundant safety division supplied directly from one of the offsite power sources with no intervening non-safety buses in such a manner that the offsite source can power the safety buses upon a failure of any non-safety bus. In its August 15, 1991, SRM, the Commission approved the staff's position.

Therefore, the staff's proposed applicable regulation for offsite power sources is as follows:

> The electric power system of standard plant design must include at least one offsite circuit to each redundant safety division supplied directly from one of the offsite power sources with no intervening non-safety buses in such a manner that the offsite source can power the safety buses upon a failure of any non-safety bus.

The initial design proposed by GE for the ABWR offsite preferred power system satisfied this staff position. However, by the draft SSAR revision dated April 3, 1992, GE changed the design to use a single transformer winding to supply power to both Class 1E and non-Class 1E loads. As a result, the staff became concerned that operation or failure of non-Class 1E loads could adversely affect operation and thus the independence of Class 1E systems.

Based on discussions, GE indicated that any single failure of a non-Class 1E load or load group will affect only one of the three Class 1E redundant load groups. The ABWR design will consist of three, non-Class 1E load groups, as well as three Class 1E divisional load groups and three transformers. Each of the three non-Class 1E load groups will be affiliated with only one of the three Class 1E divisional load groups by being powered from the same offsite power system transformer winding. Thus, failure of any one of three transformers due to failure of non-Class 1E loads or load groups can affect only one of the three Class 1E divisional load groups. In addition, the alternate offsite preferred circuit design (which does not include provision for powering of non-Class 1E loads from the same transformer winding as Class 1E loads) and operating procedures (which require that one of the three divisional buses be fed by the alternate offsite power source during normal operation) will also provide assurance that the three Class 1E divisional load groups will not be subjected to common abnormal conditions due to failure of non-Class 1E loads or load groups.

In addition, GE indicated during discussions with the staff that the ABWR design will include provisions to limit the harmonic effect on the Class 1E divisional load group power supply to less than 5 percent for operation or failure of reactor internal pumps (RIPs) or other non-Class 1E loads.

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. The staff concludes that a design which meets the above described commitments will minimize the effects of non-Class 1E system operation/failure causing loss of electric power to safety systems or failure of Class 1E safety systems and loads. The design, therefore, meets the above defined requirements of GDC 17 of 10 CFR Part 50, Appendix A, and the staff's proposed applicable regulation for offsite power sources is acceptable.

Verification that GE provided the above design commitments in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.2.3.4-1. GE included this information on Figure 8.3-1 and in Sections 8.3.1.0.1 and 8.3.4.9 of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

<u>Certified Design Material</u>

DFSER (SECY-92-349) Open Item 8.2.3.4-1 was related to the design description and the ITAAC for the independence of Class 1E systems from the influences of non-Class 1E loads. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

8.2.3.5 Physical Separation Between Offsite and Onsite Class 1E Circuits

Based on discussions with GE and information presented in Section 8.2.1.3 of the April 3, 1992, draft revision to the SSAR, the ABWR design is such that the offsite circuits will be physically separated from any Class 1E systems, equipment, components, cables, or loads by floors or walls up to the point where the offsite circuits enter the reactor building. From the point where the alternate preferred circuit enters the Division II side of the reactor building to the Class 1E switchgear rooms, and from the point where the normal preferred circuit enters the Divisions I and III side of the reactor building to the Class 1E switchgear rooms, GE has indicated (by their commitment to IEEE 384), that the offsite circuits would be physically separated from circuits of the Class 1E systems by a minimum physical separation distance of 0.9 m (3 ft) horizontal and by 1.5 m (5 ft) vertical. In addition, GE indicated that safety systems (for example, rotating equipment with potential for being a missile hazard) whose failure could potentially affect the operation of an offsite circuit will not be located in the same rooms with the normal or alternate offsite circuits, or barriers will be installed to preclude possible interaction between offsite and onsite 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. The staff concludes that a design which meet the above described commitments will provide reasonable assurance that failure of offsite circuits will not cause loss of onsite circuits and failure of onsite safety-related equipment will not cause loss of offsite circuits. The design, therefore, meets the above defined requirements of GDC 17 of 10 CFR Part 50, Appendix A, and the staff's proposed applicable regulation for offsite power sources, and is acceptable.

Verification that GE provided the above design commitments in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.2.3.4-2. GE has included these commitments in Sections 8.2.1.3 and 8.3.3.1 of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

In addition, Figure 8.3.1 of the draft SSAR revision dated April 3, 1992, indicated that the offsite power connection from the reserve auxiliary transformer is normally supplied through the Division II Class 1E equipment areas to the Division III load group. Similarly, GE indicated that the offsite connection from the unit auxiliary transformers is normally supplied through the Divisions I and III Class 1E equipment areas to the Divisions I and II load groups. To further minimize the likelihood of interaction between the offsite and onsite systems during operation, the staff contended that the normal configuration for the connection of the offsite circuit to the onsite Class 1E distribution system should be configured with the reserve auxiliary transformer normally connected to the Division II load group and the unit auxiliary transformers normally connected to the Divisions I and III load groups. This was DFSER (SECY-92-349) Open Item 8.2.3.4-2.

This is an operation rather than design issue. Therefore, in response to this item, GE specified that the COL applicant would establish the preferred configuration of offsite circuits for normal operation based on the reliability/stability of offsite circuits, the Class 1E bus loads, and the separation of the offsite feeds as they pass through the divisional areas. Based on this requirement, the staff concludes that the operational configuration of offsite circuits will 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. GE has included this COL action item in Section 8.3.4.9 of SSAR Amendment 32, which is acceptable. Therefore, DFSER Open Item 8.2.3.4-2 is resolved.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.2.3.5-2 was related to the design description and the ITAAC for the physical separation between offsite and onsite class 1E circuits. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

8.2.3.6 Grounding and Lightning Protection

This section addresses the staff's evaluation of GE's response to DFSER (SECY-92-349) Confirmatory Item 8.2.2.7-1 and Open Item 8.2.2.7-1.

The ACRS, in an April 13, 1992, letter to the staff discussed a concern that SSAR Chapter 8 did not discuss any requirements or design considerations for station grounding. In response GE has committed to meet the following Electric Power Research Institute (EPRI) plant grounding guidelines:

- A station grounding grid, consisting of bare copper cables, will be provided that will limit step-and-touch potentials to safe values under all fault conditions.
- Bare copper risers will be furnished for all underground electrical ducts and equipment, and for connections to the grounding systems within buildings.
- The design and analysis of the grounding system will follow the procedures and recommendations specified by the latest revision of IEEE 665, "Guide for Generation Station Grounding."
- Each building will be equipped with grounding systems connected to the station grounding grid. As a minimum, every other steel column of each building perimeter will connect directly to the grounding grid.
- The plant's main generator will be grounded with a neutral grounding device. The impedance of that device will limit the maximum phase current under short-circuit conditions to a value not greater than that for a three-phase fault at its terminals.
- Provisions will be included to ensure proper grounding of the isophase buses when the generator is disconnected.
- The onsite, medium-voltage ac-distribution system will be resistance grounded at the neutral point of the low-voltage windings of the unit auxiliary and reserve transformers.
- Grounding of the neutral point of the generator windings of the onsite standby power supply units (Class 1E diesel generators and non-Class 1E combustion turbine generator (CTG)) will be through distribution-type transformers and loading resistors, sized for continuous operation in the event of a ground fault.

- The neutral point of the low-voltage ac-distribution systems will be either solidly or impedance grounded, as necessary, to ensure proper coordination of ground fault protection.

- The dc systems will be left ungrounded.

- Each major piece of equipment, metal structure, or metallic tank will be equipped with two ground connections diagonally opposite each other.

- The ground bus of all switchgear assemblies, MCCs, and control cabinets will be connected to the station ground grid through at least two parallel paths.

- One bare copper cable will be installed with each underground electrical duct run, and all metallic hardware in each manhole will be connected to this cable.

- Plant instrumentation will be grounded through separate radial grounding systems consisting of isolated instrumentation ground buses and insulated cables. The instrumentation grounding systems will be connected to the station grounding grid at only one point and will be insulated from all other grounding circuits.

- Separate instrumentation grounding systems shall be provided for plant analog and digital instrumentation systems.

- A lightning protection system will be provided for each major plant structure, including the containment enclosure building. The design and installation of these systems will comply with the National Fire Protection Association's Lightning Protection Code, NFPA-78, and the Nuclear Energy Property Insurance Association's "Basic Fire Protection for Nuclear Power Plants" document.

- Lightning arresters will be provided in each phase of all tie lines connecting the plant electrical systems to the switching station(s) and offsite transmission system. These arresters will be connected to the high-voltage terminals of the main step-up and reserve transformers.

- Plant instrumentation and monitoring equipment located outdoors or connected to cabling that runs outdoors will be equipped with built-in surge suppression devices to protect the equipment from lightning-induced surges.

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. The staff concludes that a design for plant structures, systems, and equipment which meets the above described commitments will be appropriately grounded and protected from lightning. The design, therefore, meets the above defined requirements of GDC 17 and is acceptable.

Verification that GE included the above design commitments in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.2.2.7-1. GE has included these commitments in Section 8.3.1.1.6.2 and Appendix 8A.1 of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.2.3.6-2 related to the design description and the ITAAC for the grounding and lightning protection. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

8.2.3.7 Operating Restrictions on the Offsite Preferred Power Circuits

Section 8.3.1.1.1 of the draft SSAR revision dated April 3, 1992, in response to DSER (SECY-91-355) Open Item 28, indicated that during normal operation (which includes: shutdown, refueling, startup, and run modes of plant operation), the normal preferred power supply feeds two of the three Class 1E load groups. The remaining load group is fed from the alternate power source. SSAR Section 8.3.4.9 specifies that COL applicants that implement the ABWR design must include in their operating procedures that one of the three divisional buses must be fed by the alternate power source during normal operation. The intent of this provision is to arrange the offsite power supply circuits to the Class 1E buses so that all three Class 1E divisional buses are not simultaneously de-energized on the loss of only one of the offsite power supplies.

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. The staff concludes that a design which includes this operating procedure will minimize the probability of losing all Class 1E buses as a result of, or coincident with, the loss of power generated by the nuclear power unit or the loss of power from the transmission network. The design, therefore, meets the above defined requirements of GDC 17 of 10 CFR Part 50, Appendix A, and is acceptable.

Verification that GE included in a future SSAR amendment the statement that the COL applicant include the above operating procedure was DFSER (SECY-92-349) COL Action Item 8.2.4-1. GE included this action item in Sections 8.2.4.2 and 8.3.4.9 of SSAR Amendment 32, which is acceptable.

8.2.3.8 Protection of Onsite Systems

The normal and alternate preferred offsite power circuits can be subjected to transmission system's steady-state and transient conditions (such as switching and lightning surges, maximum and minimum voltage ranges for heavy and light load conditions, frequency variation, or stability limits). GE indicated that provisions will be included in the design of the offsite and onsite systems to minimize the probability of losing electric power from any of the remaining sources as a result of these conditions in accordance with the requirements of GDC 17 of 10 CFR Part 50, Appendix A. In response to DSER (SECY-91-355) Open Item 26, GE revised the SSAR to indicate that these provisions include:

- Switching and lightning surge protection is provided by the station grounding and surge protection systems described in Appendix 8A of the SSAR, by independent feeds (i.e., normal and alternate preferred power circuits described in SSAR Section 8.2.1.2), and by grounding and lighting protection specified in SSAR Section 8.2.3.

- Maximum and minimum voltage ranges are specified in SSAR Section 8.2.3 and transformers are designed per SSAR Sections 8.2.1.2 and 8.2.3. Allowable frequency variation or stability limitations are addressed in SSAR Section 8.2.3.

- Protection for degraded voltage conditions is discussed in SSAR Section 8.3.1.1.7(8).

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. The staff concludes that a design which meets these design provisions will minimize the probability of losing electric power from any of the remaining sources as a result of transmission system's steady-state and transient conditions. The design, therefore, meets the above defined requirements of GDC 17 of 10 CFR Part 50, Appendix A, and is acceptable.

Verification that these design provisions were included in a future SSAR Amendment was part of DFSER (SECY-92-349) Confirmatory Item 8.2.2.6-1. GE included these provisions in Section 8.2.2.1(2) of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

<u>Certified Design Material</u>

This part of DFSER (SECY-92-349) Open Item 8.2.2.6-1 related to the design description and the ITAAC for the protection of offsite and onsite circuits from transmission system's steady-state and transient conditions. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

8.2.4 Power Supply for the Reactor Internal Pumps

Section 15.3.1.1.1 of SSAR Amendment 10 stated that, since four buses are used to supply power to the ten reactor internal pumps (RIPs), the worst single failure can only cause three RIPs to trip. In addition, the response to Question 435.4 of SSAR Amendment 10 stated that the probability of any additional RIP trips is low (less than 10E-6 per year). Therefore, the simultaneous trip of more than three RIPs was classified by GE as a limiting fault. This classification was identified as DSER (SECY-91-355) Open Item 29.

The staff subsequently classified this postulated event in the special category of anticipated transients involving a common-mode software failure, and developed special acceptance criteria for the radiological dose calculation. The staff's evaluation of this postulated event in a special category is addressed in Section 15.2 of this report.

8.3 Onsite Class 1E Power System

GE provided in the SSAR markup dated March 31, 1993 as amended by SSAR markup dated May 11, 1993, a description of the ABWR design of the onsite Class 1E power system as being comprised of the following systems:

- <u>Class 1E Alternating Current (ac) Power System</u>

  The Class 1E ac power system will consist of three redundant Class 1E ac safety system divisions (Divisions I, II, and III). Each of these divisions will include a Class 1E diesel generator standby power supply, ac-distribution system, and load group. Each of the three Class 1E diesel generator standby power supplies will consist of all components from the stored energy (fuel) to the connection to the distribution system's supply circuit breaker. Each Class 1E ac-distribution system will consist of the following Class 1E ac-distribution systems:

- 6.9 kV medium voltage ac-distribution system

  Each of the three Class 1E 6.9 kV medium voltage ac-distribution systems (one per division) will consist of all equipment in the distribution circuit, from the power side of the offsite and onsite power supply breakers to and including the 6.9 kV Class 1E safety system loads. Equipment in each of the three Class 1E 6.9 kV ac-distribution circuits will include; one Class 1E circuit connection from the Class 1E diesel generator standby power supply, three circuit connections from non-Class 1E power supplies (the normal preferred offsite power circuit, the alternate preferred offsite power circuit, and the CTG), one 6.9 kV medium voltage Class 1E ac switchgear, the Class 1E circuit connections to and including one or more 6.9 kV medium voltage Class 1E safety system loads, the Class 1E circuit connections to two Class 1E 480-volt low voltage ac distribution systems, one Class 1E circuit connection to ground, and the Class 1E circuit connections through the Class 1E 480-volt low voltage ac-distribution systems to each of the following systems:

  - one Class 1E 120-volt ac I&C distribution system,
  - one or more Class 1E 120/240-volt ac-distribution systems,
  - one or two Class 1E dc power systems, and
  - one or two Class 1E vital 120-volt ac I&C power systems.

  Equipment in Division I of the Class 1E 6.9 kV ac-distribution circuits will also include three circuit connections to non-Class 1E fine motion control rod drive (FMCRD) motor loads (one circuit connection to each of the three FMCRD motor load groups). Each of the three circuit connections will consist of all equipment in the distribution circuit from the 6.9 kV medium voltage switchgear to and including the 480-volt FMCRD motor loads. Equipment contained in the circuit will include a Class 1E zone selective interlock circuit between the Class 1E medium voltage supply and load breakers, a pair of Class 1E 6.9 kV interlocked breakers, one non-Class 1E 6.9 kV/480-volt transformer, one non-Class 1E 480-volt MCC, and circuit connections from medium voltage switchgear through the interlocked breakers, 6.9 kV/480-volt transformer, and 480-volt MCC to FMCRD system motor loads. The circuit connection between the Class 1E medium voltage switchgear load breaker and the pair of interlocked breakers will be classified as associated. The circuit connections from interlocked breakers to the FMCRD loads and from the non-Class 1E medium voltage switchgear to the interlocked breakers will be classified as non-Class 1E. Control power for the interlocked breakers will be from the Division I Class 1E 125-volt dc system.

- 480-volt low-voltage ac-distribution system

  Each of the six Class 1E 480-volt low voltage ac-distribution systems (two per division) will consist of all equipment in the distribution circuit from the 6.9 kV medium voltage side of the 6.9 kV/480-volt transformer to and including the 480-volt Class 1E safety system loads. Equipment in each of the six Class 1E 480-volt ac-distribution circuits will include one Class 1E circuit connection from the Class 1E medium voltage distribution system, one Class 1E 6.9 kV/480-volt transformer, one Class 1E 480-volt switchgear, Class 1E circuit connection to and including one or more 480-volt Class 1E safety system loads, one Class 1E circuit connection to ground, and Class 1E circuit connection to and including one or more Class 1E 480-volt ac MCCs and their affiliated 480-volt Class 1E safety system loads.

- 120-volt I&C ac-distribution system

  Each of the three Class 1E 120-volt I&C ac-distribution systems (one per division) will consist of all equipment in the distribution circuit from the 480-volt side of the 480/120-volt transformer to and including the 120-volt Class 1E safety system I&C loads. Equipment in each of the three Class 1E I&C ac-distribution circuits will include one Class 1E 480/120-volt transformer, two Class 1E 120-volt ac-distribution panels, and Class 1E circuit connection to and including 120-volt Class 1E safety system I&C loads.

- 120/240-volt low-voltage ac-distribution system

  Each of the Class 1E 120/240-volt low voltage ac-distribution systems (one or more per division) will consist of all equipment in the distribution circuit from the 480-volt side of the Class 1E 480/120/240-volt transformer to and including Class 1E safety system loads. Equipment in each of the Class 1E 120/240-volt distribution circuits will include one Class 1E 480/120/240-volt transformer, one or more Class 1E 120/240-volt ac distribution panels, and Class 1E circuit connection to and including Class 1E 120- and 240-volt safety system loads.

- Class 1E Direct Current (dc) Power System

The Class 1E dc power system will consist of four redundant 125-volt Class 1E dc safety system divisions (Divisions I, II, III, and IV). Each of these divisions will include a Class 1E battery and battery charger power supply, 125-volt dc-distribution system, and load group. Each of the four Class 1E battery power supplies will consist of storage cells, connectors, and connections to the distribution system supply circuit interrupting device. Each of the four Class 1E battery charger power supplies will consist of all equipment from the connection to the 480-volt Class 1E ac-distribution system to its distribution system's supply breaker. Each of the four Class 1E 125-volt dc-distribution systems will consist of all equipment in the distribution circuit, from the power side of the battery interrupting device and the battery charger supply breaker to and including the 125-volt Class 1E dc safety system loads. Equipment in each of the four distribution circuits will include one or more Class 1E distribution panels and connections to and including 125-volt Class 1E dc safety system loads.

Divisions I and III of the 480-volt Class 1E ac-distribution system will feed the Divisions I and III battery charger power supplies, respectively. Division II of the 480-volt Class 1E ac-distribution system will feed the Divisions II and IV battery charger power supplies.

- Class 1E Vital ac I&C Power System

The Class 1E vital ac I&C power system will consist of four redundant Class 1E vital 120-volt ac I&C safety system divisions (Divisions I, II, III, and IV). Each of these divisions will include a Class 1E Constant Voltage Constant Frequency (CVCF) power supply, a 120-volt Class 1E ac-distribution system, and a load group. Each of the four CVCF power supplies will consist of the power source (the static inverter, ac and dc static transfer switches, and a regulating step down transformer as an alternate ac power supply) and its connection to the distribution supply circuit interrupting device. Each of the four Class 1E vital 120-volt ac I&C distribution systems will consist of all equipment in the distribution circuit from the power side of the constant CVCF power supply breaker to and including the Class 1E safety system I&C loads. Equipment in each of the four Class 1E vital 120-volt ac distribution circuits will include one or more 120-volt ac distribution panels and connections to and including vital 120-volt Class 1E safety system I&C loads.

Each divisional CVCF power supply will be supplied power from its affiliated divisional dc power system. (For example, the Division I CVCF power supply will be supplied from Division I 125-volt dc-distribution system.) In addition, Divisions I and III of the 480-volt ac-distribution system will supply power to the Divisions I and III CVCF power supplies, respectively. Similarly, Division II of the 480-volt ac-distribution system will supply power to the Divisions II and IV CVCF power supplies.

The staff concludes that the above design is consistent with the typical electrical system design defined by IEEE 308-1980. The staff's review of the ABWR electrical system design was based on this typical design. GE has included the above typical design in SSAR Amendment 32, which is acceptable.

In addition, GE indicated that operational restrictions would apply to the use of Class 1E receptacles which are powered from each of the Class 1E 120/240-volt distribution systems. Verification that these operational restrictions will be included in appropriate COL procedures to ensure compliance with the capacity, independence, and protection provisions required by GDC 2, 4, 17, and 18 of 10 CFR Part 50, Appendix A for Class 1E power systems was DFSER (SECY-92-349) COL Action Item 8.3-1. In response to this item, GE revised their electrical system design to eliminate Class 1E electrical receptacles. GE included their revised design in Section 8.3.1.1.3 of SSAR Amendment 32 negating the need for a specific COL action item which is acceptable.

GDC 17 of 10 CFR Part 50, Appendix A, requires that the Class 1E systems have sufficient (1) capacity and capability to permit safety systems to perform their required safety function and (2) sufficient independence and redundancy to perform their safety functions assuming a single failure. With the elimination of Class 1E receptacles, the potential for loss of independence between redundant Class 1E divisions and loss of sufficient capacity of the Class 1E power supplies due to incorrect use of the receptacles was eliminated from the design. Consequently, the design meets the above defined requirements of GDC 17 and is acceptable.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.3-1 related to the design description and the ITAAC for the onsite Class 1E power system design. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. On the basis of this evaluation, this item is resolved.

To ensure that the ABWR design incorporates sufficient capacity, capability, independence, redundancy, and testability of onsite Class 1E power systems, the staff's evaluation also addressed the areas discussed in the following sections.

8.3.1 Compliance with General Design Criteria

This section addresses the staff's evaluation of GE's response to DSER (SECY-91-355) Open Items 27 and 30.

Item (1)(b) of Section 8.3.1.2.1 of SSAR Amendment 10 indicated that the Class 1E ac power system complies with GDC 2, 4, 17, and 18 in part or as a whole, as applicable. GE's response to Question 435.26 (also of Amendment 10) provided clarification that there are no non-compliances, but also indicated that some portions of the GDC do not apply to the CVCF power supplies (for example, the statement in GDC 17 about two physically independent circuits from the transmission network). Based on the information presented, the staff could not ascertain which parts of the GDC GE considered not applicable to the CVCF power supplies.

In its draft submittal dated September 4, 1991, GE proposed modifying the response to Question 435.26 and Section 8.3.1.2.1 to indicate full compliance with the GDC. The proposed modification deleted (1) certain conflicting statements in the SSAR, (2) the example of non applicability to GDC 17, and (3) the phrase "the substance and intent of" from Section 8.3.1.4.2.1. In addition, GE agreed to revise Item 11 of Section 1.2.1.1.2 of SSAR Amendment 1 to clarify the systems or components to which IEEE 279, "Criteria for Protection Systems for Nuclear Power Generating Stations" (1971) applies and to correct inconsistencies concerning applicable SRP criteria within Table 8.1-1 and between Table 8.1-1 and Section 8.1.3.1.2.

The staff concludes that the ABWR electrical system design with the above mentioned changes will comply with GDC 17 requirements to GDC commitments and is acceptable. Verification that GE provided the above clarifications in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.3.1-1. GE included this information in Table 8.1-1 of SSAR Amendment 21 and in Sections 1.2.1.1.2(11), 8.3.1.2(1)(b), 8.3.3.6.2.1, and 8.1.3.1.2 of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

8.3.1.1 Compliance with IEEE Standards (NRC Policy Issue SECY-91-273, Section II.A of Enclosure 1 to SECY-93-087)

In the draft SSAR submittal dated September 4, 1991, GE indicated that there will be no limitation on the use of IEEE 384-1981 "IEEE Standard Criteria for Independence of Class 1E Equipment and Circuits," for separation in the ABWR design. The NRC staff has not formally reviewed and accepted the changes between the 1974 and 1981 versions of this standard. Also, IEEE 384 is not the only standard in this classification for which the NRC has not completed its formal evaluation and regulatory guide endorsement of the newer version of the standard. Thus, to allow the use of an updated IEEE standard which the NRC staff has not formally reviewed or endorsed, the staff felt that each difference between the old and new standard needed to be identified, justified, and approved for use. Such review is required in order to ensure that the design criteria of the new standards are equally conservative as those included in the standards currently approved by the staff. This was DSER (SECY-91-355) Open Item 41.

In order to identify, justify, and approve the differences between the old and new standards for use on the ABWR and also complete the evaluation of the electrical system design for the ABWR, the staff evaluated the guidelines in two of the newer standards with respect to the intent of criteria and guidelines contained in the SRP and existing regulatory guides. The staff believes that these two standards provide the majority of the basic criteria for the electrical power systems. The electrical design proposed for the ABWR was then evaluated using the newer standards.

The newer standards were revised to be consistent with IEEE 603-1980. The newer standards involved primarily clarification and amplification of guidelines contained in prior standards and were thus considered the more relevant base from which to evaluate the ABWR design. The newer standards included

- IEEE 308-1980, "IEEE Standard Criteria for Class 1E Power Systems for Nuclear Power Generating Stations" (This standard was specifically revised to be consistent with IEEE 603-1980, which was endorsed by the NRC staff in RG 1.153 in 1985.)

- IEEE-384-1981 "IEEE Standard Criteria for Independence of Class 1E Equipment Circuits"

In addition, IEEE has developed and issued other companion standards to provide additional guidance for certain areas. These standards include

- IEEE 741-1986, "Standard Criteria for the Protection of Class 1E Power Systems and Equipment in Nuclear Power Generating Stations"
- IEEE 765-1983, "Standard for Preferred Power Supply for Nuclear Power Generating Stations"
- IEEE 485-1983, "Recommended Practice for Sizing Large-Lead Storage Batteries for Generating Stations and Substations"
- IEEE 946-1985, "Recommended Practice for the Design of Safety-Related dc Auxiliary Power Systems for Nuclear Power Generating Stations"

Like the two standards cited above, these other companion standards have not been endorsed by an NRC regulatory guide. However, these standards were developed to be used with IEEE 308-1980 and they also clarify and amplify current SRP criteria and guidelines. The staff therefore considers these standards the more relevant base from which to evaluate the ABWR design.

In some cases, GE had not referenced these other companion standards in the ABWR SSAR. The staff proceeded with its review with the understanding that GE intended to use these other standards. Verification of this understanding in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.3.1.1-1. GE included a statement of conformance to these standards in Sections 8.3.1.2(5), 8.2.2.1(9), and 8.3.2.2.2(5) of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

8.3.1.2 Compliance with GDC 2 and 4

ABWR SSAR Chapter 8 was modified, in response to DSER (SECY-91-355) Open Item 31 and 63, by the draft revision dated April 3, 1992, as amended by an SSAR markup dated March 31, 1993, to include the following statements related to the compliance of the electrical system design to the requirements of GDC 2, "Design Bases for Protection Against Natural Phenomena," and GDC 4, "Environmental and Missiles Design Bases," of 10 CFR Part 50, Appendix A:

- "Electrical equipment and wiring for the Class 1E systems which are segregated into separate divisions are separated so that no design-basis event is capable of disabling more than one division of any ESF total function."
- "Redundant parts of the system are physically separated and electrically independent to the extent that in any design-basis event with any resulting loss of equipment, the plant can still be shut down with the remaining two divisions."
- "Class 1E electric equipment and wiring is segregated into separate divisions so that no single credible event is capable of disabling enough equipment to hinder reactor shut down and removal of decay heat by either of two unaffected divisional load groups or prevent isolation of the containment in the event of an accident."
- "Equipment arrangement and/or protective barriers are provided such that no locally generated force or missile can destroy any reactor protection system (RPS), nuclear steam supply system, (NSSS), emergency core cooling system (ECCS), or engineered safety feature (ESF) functions. In addition, arrangement and/or separation barriers are provided to ensure that such disturbances do not affect both high pressure core flooder (HPCF) and reactor core isolation cooling (RCIC) systems."
- "Containment penetrations are so arranged that no design-basis event can disable cabling in more than one division."
- "The protection system and ESF control, logic, and instrument panels/racks shall be located in a safety class structure in which there are no potential sources of missiles or pipe breaks that could jeopardize redundant cabinets and raceways."
- "The standby ac power system is capable of providing the required power to safely shut down the reactor after LOPP and/or LOCA and to maintain the safe-shutdown condition and operate the Class 1E auxiliaries necessary for plant safety after shutdown."

Based on the above stated capability to safely shut down, other SSAR commitments regarding physical protection of electrical divisions, and discussions, GE indicated that there will be a limited number of design-basis events for which Class 1E systems, equipment, and components will be protected by the capability to maintain safe plant shut down with any one of the three load groups.

GDC 2 and 4 of 10 CFR Part 50, Appendix A, requires that Class 1E systems, equipment, and components be designed (1) to withstand the effects of natural phenomena without loss of capability to perform their safety functions, (2) to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, and (3) be appropriately protected against dynamic effects. GDC 17 of 10 CFR Part 50, Appendix A, requires that the Class 1E systems, equipment, and components have sufficient independence to perform their safety functions assuming a single failure. The staff concludes that a design that meets the above described commitments will reasonably assure that no design basis event (that is, failure of any one safety-related system division or failure of non-Class 1E equipment) will cause failure of more than one safety-related system division. The design, therefore, meets the above defined requirements of GDC 2, 4, and 17 of 10 CFR Part 50, Appendix A, and is acceptable. Verification that GE has provided the above design commitments in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.3.1.2-1. GE has included information in Sections 8.3.3.1, 8.1.3.1.1.1, 8.3.3.6.1.1, 8.3.3.6.2.3.2, 8.3.3.6.2.2.3, and 8.1.2.2 of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.3.1.2-1 related to the design description and the ITAAC for the capability of performing a safe shutdown (performing the required minimum safety function). The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

8.3.2 Physical Independence

8.3.2.1 Conduits to Open Tray Separation (Scram Cables)

Section 8.3.3.6.2.3.1 of the draft SSAR revision dated April 3, 1992, documented the following design commitments in response to DSER (SECY-91-355) Open Item 32:

- The reactor protective system scram solenoid circuits, from the actuation devices to the solenoids of the scram pilot valves of the control rod drive hydraulic control units, will be run in grounded steel conduits, containing no other wiring.
- Separate grounded steel conduits will be provided for the scram solenoid wiring for each of four scram groups.
- Separate grounded steel conduits will also be provided for both the A and B solenoid wiring circuits of the same scram group.
- Scram group conduits will have unique identification and will be separately routed as Divisions II and III conduits for the A and B solenoids of the scram pilot valves, respectively. This corresponds to the divisional assignment of their power sources.
- Conduits containing the circuits for the A solenoids of the scram pilot valves (Division II) will be separated from their B solenoid counterpart (Division III) by a minimum separation distance of 2.54 cm (1 in.), in accordance with divisional separation requirements.
- The scram group conduits will not be routed within the confines of any other tray or raceway system.
- The conduits containing the scram solenoid group wiring of any one scram group will also be physically separated by a minimum distance of 2.54 cm (1 in.) from the conduit of any other scram group and from conduits or metal-enclosed raceways affiliated with any of the four Class 1E divisions or any non-Class 1E (non-divisional) circuits.
- The conduits containing the scram solenoid group wiring of any one scram group will also be physically separated from non-enclosed raceways associated with any of the four safety-related electrical divisions or any non-safety-related (non-divisional) circuits in accordance with IEEE 384 and RG 1.75, Revision 2.
- Separation "in accordance with IEEE 384" means that conduits containing scram solenoid group circuit wiring will be separated from any non-enclosed raceway containing either safety or non-safety-related circuits. Specifically, the vertical separation distance will be 1.5 or more m (5 or more ft) and the horizontal separation distance will be .9 or more m (3 or more ft).

GDC 4 of 10 CFR Part 50, Appendix A, requires that Class 1E systems, equipment, and components be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents. GDC 17 of 10 CFR Part 50, Appendix A, requires that the onsite electric distribution system have sufficient independence to perform their safety function assuming a single failure. The staff concludes that a design that meets the above described commitments will reasonably preclude the common failure of reactor protection system scram solenoid circuits and other Class 1E or non-Class 1E circuits. The design, therefore, meets the above defined requirements of GDC 4 and 17 of 10 CFR Part 50, Appendix A, and is acceptable.

Verification that GE provided the above design commitments in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.3.2.1-1. In response to this item, GE further clarified their design commitment for separation in accordance with IEEE 384. Specifically GE indicated that the RPS conduits containing the scram group wiring for the A and B solenoids of the scram pilot valves (associated with Divisions II and III, respectively), will be separated from non-enclosed raceways associated with any of the four electrical divisions or non-divisional cables by 0.9 m (3 ft) horizontally, or 1.5 m (5 ft) vertically, or with an additional barrier that is separated from any raceway by 2.5 cm (1 in.). The staff concludes that a design that meets the above described commitments as revised will also reasonably preclude the common failure of reactor protection system scram solenoid circuits and other Class 1E or non-Class 1E circuits. The design, therefore, meets the above defined requirements of GDC 4 and 17 of 10 CFR Part 50, Appendix A, and is acceptable. GE has included the revised design commitments in Section 8.3.3.6.2.3.1 of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.3.2.1-1 related to the design description and the ITAAC for separation between scram cables and between scram and other cables. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

8.3.2.2 Dedicated Neutron Monitoring Raceways

Section 8.3.1.3.1.3 of the draft SSAR revision dated April 3, 1992, (in response to DSER (SECY-91-355) Open Item 32) indicated that neutron monitoring cables will be routed in their own divisional conduits and cable trays, separate from all other power, instrumentation, and control cables.

GE also committed that neutron monitoring cables will be routed in their own dedicated raceways from termination to termination. These dedicated raceways will be separated from raceways containing all other Class 1E or non Class 1E power, instrumentation, and control cables by the same separation provided between scram and other cables described in Section 8.3.2.1 of this report.

GDC 4 of 10 CFR Part 50, Appendix A, requires that Class 1E systems, equipment, and components be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents. GDC 17 of 10 CFR Part 50, Appendix A, requires that the onsite electric distribution system have sufficient independence to perform their safety function assuming a single failure. The staff concludes that a design that meets the above described commitments will reasonably preclude the common failure of neutron monitoring circuits and other Class 1E or non-Class 1E circuits. The design, therefore, meets the above defined requirements of GDC 4 and 17 of 10 CFR Part 50, Appendix A, and is acceptable.

Verification that GE provided the above design commitments in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.3.2.2-1. In response to this item, GE further clarified in its design that neutron monitoring cables will be routed in their own dedicated raceways for the purpose of assuring their protection from the effects of electromagnetic interference (EMI). GE indicated that additional physical separation requirements are not necessary to assure protection from the effects of EMI.

The staff concludes that a design that meets the above described commitments as modified will also reasonably preclude the common failure of neutron monitoring circuits and other Class 1E or non-Class 1E circuits. The design, therefore, meets the above defined requirements of GDC 17 of 10 CFR Part 50, Appendix A, and is acceptable. GE has included this commitment in Section 8.3.3.5.1.3 of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

The issue related to protection of circuits from EMI is resolved in Chapter 7.0 of this report.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.3.2.2-1 related to the design description and the ITAAC for routing of I&C neutron monitoring circuits in dedicated raceways. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

8.3.2.3 Separation of dc Emergency Lighting Raceways

In Section 9.5.3 of the draft SSAR revision dated April 3, 1992, in response to DSER (SECY-91-355) Open Item 32, GE indicated that dc emergency lighting cables will be routed in their own divisional conduits and cable trays, separate from all other power, instrumentation, and control cables.

GE indicated that the dc emergency lighting cables will be routed in their own dedicated raceways from termination to termination. These dedicated raceways will be separated from raceways containing all other Class 1E or non-Class 1E power and I&C cables by the same separation provided between scram and other cables, as described in Section 8.3.2.1 of this report.

GDC 4 of 10 CFR Part 50, Appendix A, requires that Class 1E systems, equipment, and components be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents. GDC 17 of 10 CFR Part 50, Appendix A, requires that the onsite electric distribution system have sufficient independence to perform their safety function assuming a single failure. The staff concludes that a design that meets the above described commitments will minimize to the extent practicable the common failure of the standby and dc emergency lighting circuits. The design, therefore, meets the above defined requirements of GDC 4 and 17 of 10 CFR Part 50, Appendix A, and is acceptable.

Verification that GE provided the above design commitments in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.3.2.3-1. In response to this item, GE clarified their design to indicate that emergency dc lighting circuits will not share raceways with any other circuits in order to enhance lighting reliability. GE indicated that physical separation of these raceways within the same division is not required.

The staff concludes that a design that meets the above described commitment as modified will also minimize to the extent practicable the common failure of the standby and dc emergency lighting circuits. The design, therefore, meets the above defined requirements of GDC 4 and 17 of 10 CFR Part 50, Appendix A, and is acceptable. GE has included this design commitment in Sections 9.5.3 and 9.5.3.1.1(7)(d) of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.3.2.3-1 related to the design description and the ITAAC for routing of emergency dc lighting in dedicated raceways. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

8.3.2.4 Separation of Containment Electrical Penetrations

Separation Between Class 1E Penetrations of Redundant Divisions

Item (7) of Section 8.3.1.4.1.2 of SSAR Amendment 10 indicated that electric penetration assemblies of different Class 1E divisions will be separated by distance, barriers, and/or location in separate rooms or on separate floors. The use of barriers and/or location in separate rooms or on separate floors exceeds separation guidelines for penetrations and are acceptable approaches. Separation by distance may also meet separation guidelines; however, SSAR Amendment 10 did not clearly define what constitutes the minimum allowable distance between penetrations. This was DSER (SECY-91-355) Open Item 33.

In the draft SSAR revision dated April 3, 1992, item (7) of Section 8.3.1.4.1.2 similarly indicated that electrical penetration assemblies of different Class 1E divisions will be separated by 3-hour fire-rated-barriers (that is locations in separate rooms or on separate floors). Separation by distance (without barriers) will be allowed only within the inerted containment. Section 8.3.1.1.5.1 of the April draft revision further indicated that penetration assemblies will be located around the periphery of the containment and at different elevations to facilitate reasonably direct routing of cables to and from the equipment.

GDC 4 of 10 CFR Part 50, Appendix A, requires that Class 1E systems, equipment, and components be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents. GDC 17 of 10 CFR Part 50, Appendix A, requires that the onsite electric distribution system have sufficient independence to perform their safety function assuming a single failure. The staff concludes that a design that meets the above described commitments will reasonably ensure that failure of Class 1E (or associated) penetration circuits in any one division will not cause failure of Class 1E (or associated) penetration circuits in a different Class 1E division. The design, therefore, meets the above defined requirement of GDC 17 of 10 CFR Part 50, Appendix A, and is acceptable.

Verification that GE provided the above design commitments in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.3.2.4-1. GE has included these commitments in Sections 8.3.3.1 and 8.3.3.6.1.2 of SSAR Amendment 32, which is acceptable.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.3.2.4-1 related to the design description and the ITAAC for the separation between Class 1E penetrations of redundant divisions. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

Separation of Class 1E Penetrations From Non-Class 1E Penetrations

Section 8.3.1.4.1.2 of the draft SSAR revision dated April 3, 1992, in response to DSER (SECY-91-355) Open Item 34 indicated that separation between penetrations containing non-Class 1E circuits and those containing Class 1E or associated circuits will be in accordance with IEEE 384. GE indicated that "separation in accordance with IEEE 384" means a vertical separation of 1.5 or more m (5 or more ft) and a horizontal separation distance of 0.9 or more m (3 or more ft).

The staff concludes that a design which meets the above described commitments will reasonably ensure that failure of non-Class 1E system penetration circuits will not cause failure of Class 1E (or associated) penetration circuits. The design, therefore, meets the above defined requirements of GDC 4 and 17 of 10 CFR Part 50, Appendix A, and is acceptable.

Verification that GE provided the above design commitments in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.3.2.4-2. In response to this item, GE further clarified their design to indicate that the separation between Electrical penetration assemblies containing non-Class 1E circuits and penetration assemblies containing Class 1E or associated Class 1E circuits is by walls, barriers, or floors that have a three-hour fire-rating. The staff concludes that a design which meets the above described commitments will also reasonably ensure that failure of non-Class 1E penetration circuits will not cause failure of Class 1E (or associated) penetration circuits. The design, therefore, meets the above defined requirements of GDC 4 and 17 of 10 CFR Part 50, Appendix A, and is acceptable. GE has included this design commitment in Section 8.3.3.6.1.2(7) of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.3.2.4-2 related to the design description and the ITAAC for the separation between Class 1E and non-Class 1E penetrations. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

Separation of Class 1E Penetrations from Non-Class 1E Cables or Other Divisional Cables

This section addresses the staff's evaluation of GE's response to DSER (SECY-91-355) Open Item 35. GE indicated that penetrations containing Class 1E circuits will be separated from other divisional cables by routing through separate rooms and/or different floors outside containment and by maintaining a minimum separation of 0.9 m (3 ft) horizontal and 1.5 m (5 ft) vertical distance inside the inerted containment. In addition, separation between penetrations containing Class 1E circuits and non-divisional cables will be maintained at a minimum horizontal distance of 0.9 m (3 ft) and a vertical distance of 1.5 m (5 ft) both inside and outside of containment.

The staff concluded that a design which meets the above described commitments will also reasonably ensure (1) that failure of Class 1E (and associated) circuits of one division will not cause failure of Class 1E (or associated) penetration circuits in a different division and (2) that failure of non-Class 1E circuits will not cause failure of Class 1E (or associated) penetration circuits. The design, therefore, meets the above requirements of GDC 4 and 17 of 10 CFR Part 50, Appendix A, and is acceptable.

Verification that GE provided the above design commitments in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.3.2.4-3. In response to this item, GE further clarified their design to indicate that the separation between Electrical penetration assemblies containing Class 1E or associated circuits from other divisional or non-divisional cables is by walls, barriers, or floors that have a three-hour fire-rating. The staff concluded that this separation exceeds the guidelines of IEEE 384, meets the protection and independence requirements of GDC 4 and 17 of 10 CFR Part 50, Appendix A, and is acceptable. GE included this design information in Section 8.3.3.6.1.2(7) of SSAR Amendment 32, which is acceptable. Therefore, DFSER (SECY-92-349) Confirmatory Item 8.3.2.4-3 is resolved.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.3.2.4-3 related to the design description and the ITAAC for the separation between Class 1E penetrations to non-Class 1E cables or to other divisional cables. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

8.3.2.5 Separation and Protection of Cables Located Outside Cabinets and Panels

Section 8.3.1.1.5.1 of the draft SSAR revision dated April 3, 1992, documents the following design commitments related to separation of Class 1E cables in response to DSER (SECY-91-355) Open Item 36:

(1) Enclosed solid metal raceways are required for separation between Class 1E or associated cables of different safety divisions and between Class 1E or associated cables and non-Class 1E cables, if: the vertical separation is less than 1.5 m (5 ft), the horizontal separation distance is less than .9 m (3 ft), and the cables are located in the same fire area.

(2) Both groupings of cables requiring separation (as specified in item 1) must be enclosed in solid metal raceways.

GE indicated that all power, control, and instrumentation cables (including fiber optic cables) located outside cabinets and panels throughout the plant will be supported in raceways in accordance with IEEE-recommended practice for support of cable systems. When Class 1E (or associated) cables of different Class 1E divisions are separated from each other or from non-Class 1E cables by less than 1.5 m (5 ft) vertically or .9 m (3 ft) horizontally, the cables will be supported in enclosed solid metal raceways (such as rigid or flexible metal conduits or totally enclosed cable trays).

In addition, Section 8.3.1.2.1(2)(f) of the draft SSAR revision dated April 3, 1992, and Section 1.8.2 (Table 1.8-21) of SSAR Amendment 17 indicated that the ABWR electric system design will comply with the requirements of IEEE 384-1981. Based on this commitment, GE indicated that the separation distance will be at least 2.54 cm (1 in.) between solid metal raceways containing Class 1E (or associated) cables of different Class 1E divisions or between solid metal raceways containing Class 1E (or associated) cables and non-Class 1E cables.

GDC 4 of 10 CFR Part 50, Appendix A, requires that Class 1E systems, equipment, and components be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents. GDC 17 of 10 CFR Part 50, Appendix A, requires that the Class 1E systems, equipment, and components have sufficient independence to perform their safety functions assuming a single failure. The staff concludes that a design which meets the above described commitments will reasonably ensure (1) that failure of Class 1E (or associated) cables in any one division (located outside of cabinets and panels and in any single raceway) will not cause failure of Class 1E (or associated) cables in a different Class 1E division and (2) that failure of non-Class 1E cables (located outside of cabinets and panels and in any single raceway) will not adversely affect Class 1E (or associated) cables. Consequently, the staff concludes that the design meets the above defined requirements of GDC 4 and 17 of 10 CFR Part 50, Appendix A, and is acceptable.

Verification that GE provided the above design commitments in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.3.2.5-1. GE has included these commitments in Section 8.3.3.1 of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.3.2.5-1 related to the design description and the ITAAC for the separation and protection of cables located outside cabinets and panels. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

8.3.2.6 Separation of Cables Inside Cabinets and Panels

In response to DSER (SECY-91-355) Open Item 37, Sections 8.3.1.4.1, 8.3.1.4.1.2, 8.3.1.4.2, 8.3.1.3.1.3, and 8.3.1.4.2.2.3 of Amendment 10 to the SSAR and draft SSAR revision dated April 3, 1992, document the following design commitments related to separation of power, control, and instrumentation cables inside panels, racks, cabinets, and other enclosures located in the main control room and other areas of the plant.

- Single panels or instrument racks will not contain circuits or devices of different Class 1E safety system divisions, except under the following conditions:
  - Certain operator interface control panels may have operational considerations which dictate that Class 1E safety system circuits or devices of different divisions must be located in a single panel. These circuits and devices will be separated horizontally and vertically by a minimum distance of 15.2 cm (6 in.) or by steel barriers or enclosures.
  - The input and output circuits of isolation devices will be separated horizontally and vertically by a minimum distance of 15.2 cm (6 in.) or by steel barriers or enclosures.
- Class 1E circuits and devices will also be separated from the non-Class 1E circuits and devices which are present inside a panel. These circuits and devices will be separated from each other horizontally and vertically

by a minimum distance of 15.2 cm (6 in.) or by steel barriers or enclosures.

- If two panels containing circuits of different divisions are less than .9 m (3 ft) apart, there will be a steel barrier between the two panels. Panel ends closed by steel end plates will be considered to be acceptable barriers provided that terminal boards and wireways are spaced a minimum of 2.5 cm (1 in.) from the end plate.

- Penetration of separation barriers within a subdivided panel will be permitted, provided that such penetrations are sealed or otherwise treated so that fire generated by an electrical fault could not reasonably propagate from one section to the other and disable a protective function.

Based on the commitment to meet the guidelines of IEEE 384-1981 and RG 1.75, Revision 2, GE indicated that Class 1E or non-Class 1E power circuits located inside panels and cabinets will be limited to those required to operate systems, equipment, or components located inside the panels and cabinets. Power cables will not be permitted to traverse from one side of a panel or cabinet to the other without being terminated inside the panel. In addition, these circuits will be routed inside rigid or flexible conduits that will be physically separated from instrumentation and control cables by minimum horizontal and vertical distances of 15.2 cm (6 in.) or by steel barriers or additional enclosures.

GDC 4 of 10 CFR Part 50, Appendix A, requires that Class 1E systems, equipment, and components be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents. GDC 17 of 10 CFR Part 50, Appendix A, requires that the Class 1E systems, equipment, and components have sufficient independence to perform their safety functions assuming a single failure. The staff concludes that a design which meets the above described commitments will reasonably ensure (1) that failure of Class 1E (or associated) cables in any one division (located inside of cabinets or panels) will not cause failure of Class 1E (or associated) cables in a different safety division, (2) failure of non-Class 1E cables (located inside cabinets or panels) will not adversely affect Class 1E (or associated) cables, and (3) normal operation and/or failure of power circuits (Class 1E, associated Class 1E, or non-Class 1E) will not adversely affect I&C circuits. The design, therefore, meets the above defined requirements of GDC 4 and 17 of 10 CFR Part 50, Appendix A, and is acceptable.

Verification that GE provided the above design commitments in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.3.2.6-1. In response to this item, GE indicated that it was not their intent to commit to the routing of power circuits inside rigid or flexible conduits that will be physically separated from instrumentation and control cables of the same division by minimum horizontal and vertical distances of 15.2 cm. (6 in.) or by steel barriers or additional enclosures. GE further indicated that the IEEE guideline "installed in enclosed raceways that qualify as barriers" has been interpreted to mean that for EMI considerations, power cables will be routed in metallic conduit wherever they come in close proximity with low level (V1) cables. For independence between power and I&C cables within the same division, the staff agreed with GE's interpretation. The staff concludes that a design that meets the above described commitments as clarified will reasonably ensure that normal operation and/or failure of power circuits (Class 1E, associated Class 1E, or non-Class 1E) will not adversely affect I&C circuits. The design, therefore, meets the above defined requirements of GDC 4 and 17 of 10 CFR Part 50, Appendix A, and is acceptable. The staff's evaluation for the protection of circuits from EMI is addressed in Chapter 7.0 of this report. GE has included the above design commitments in Sections 8.3.3.6.1.1, 8.3.3.6.2.2.3, and 8.3.3.6.2.2.4 of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.3.2.6-1 related to the design description and the ITAAC for the separation of cables inside cabinets/panels. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

8.3.2.7 Separation of Cables Approaching and/or Exiting Cabinets and Panels

The response to Question 435.30 in SSAR Amendment 10, DSER (SECY-91-355) Open Item 40, stated that cable spreading areas do not apply to the ABWR and are not included in the plant layout because the majority of the I&C signals will be multiplexed to the control room. This response implied that the 0.3 m (1 ft) by 0.9 m (3 ft) separation guidelines allowed by Section 5.1.3 of IEEE 384-1974 (Section 6.1.3 of IEEE 384-1981) will not apply to the ABWR. The guidelines of Position C12 of RG 1.75 Revision 2, also will be irrelevant. The ABWR SSAR did not clearly address the criteria for the separation and protection of cables approaching or exiting cabinets and panels.

In discussions with the staff and in Section 8.3.1.4 of the draft SSAR revision dated April 3, 1992, GE clarified its design commitments as follows:

- I&C and optical cables (including metallic and fiber-optic cables) will be treated the same with respect to separation and protection throughout the plant.

- Each division of Class 1E power, instrumentation, and control cables will be routed to the control room complex through a cable chase or other means, so that different divisional areas will be separated by a 3-hour fire-rated barrier.

- Each cable chase will be ventilated.

- Separation between Class 1E and non-Class 1E cables within the cable chase will be the same as separation of cables located outside cabinets and panels as described in Section 8.3.2.5 of this report.

- Class 1E, associated, or non-Class 1E power circuits routed in a cable chase serving the control room or in the control room area will be limited to those required to operate systems, equipment, or components located in the control room area (power cables will be not be permitted to traverse through from one side of the control room area to the other without being terminated in the control room area).

- Class 1E, associated, or non-Class 1E power circuits routed in a cable chase or the control room area will be routed inside rigid or flexible conduits that will be physically separated horizontally and vertically from any I&C cables by a minimum distance of 15.2 cm (6 in.) or by steel barriers or additional enclosures.

- Power cables may be routed in flexible metallic conduit under the raised floor of the control room.

- Separation between divisional and between divisional and non-divisional power, instrumentation, and control cables within the control room area will be separated in the same way as cables located outside cabinets and panels described in Section 8.3.2.5 of this report.

- Power, instrumentation, and control cables of different Class 1E divisions will enter cabinets and panels through separate apertures. Similarly, Class 1E and non-Class 1E power, instrumentation, and control cables will enter cabinets or panels through separate apertures.

- Cable chases and the control room area will be non-hazard areas (Section 6.1.3 of IEEE 384-1981 defines non-hazard areas).

- Cable chases and the control room area will not contain potential hazards such as high energy switchgear, power distribution panels, transformers or rotating equipment, potential sources of missiles, pipe failure hazards, or fire hazards.

GDC 4 of 10 CFR Part 50, Appendix A, requires that Class 1E systems, equipment, and components be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents. GDC 17 of 10 CFR Part 50, Appendix A, requires that the Class 1E systems, equipment and components have sufficient independence to perform their safety functions assuming a single failure. The staff concludes that a design which meets the above described commitments will reasonably ensure (1) that failure of Class 1E (or associated) cables in any one division (located in a cable chase or the control room area) will not cause failure of Class 1E (or associated) cables in a different safety division, (2) that failure of non-Class 1E cables (located in a cable chase or the control room area) will not adversely affect Class 1E (or associated) cables, and (3) normal operation and/or failure of power circuits (Class 1E, associated, or non-Class 1E) will not adversely affect I&C circuits. The design therefore meets the above defined requirements of GDC 4 and 17 of 10 CFR Part 50, Appendix A, and is acceptable.

Verification that GE provided the above design commitments in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.3.2.7-1. In response to this item, GE indicated that it was not their intent to commit to the routing of power circuits inside rigid or flexible conduits that will be physically separated from instrumentation and control cables of the same division by minimum horizontal and vertical distances of 15.2 cm (6 in.) or by steel barriers or additional enclosures. GE further indicated that the IEEE guideline "installed in enclosed raceways that qualify as barriers" has been interpreted to mean that for EMI considerations, power cables will be routed in metallic conduit wherever they come in close proximity with low level (V1) cables. For independence between power and I&C cables within the same division, the staff agreed with GE's interpretation. The staff concludes that a design which meets the above described commitments as clarified will reasonably ensure that normal operation and/or failure of power circuits (Class 1E, associated, or non-Class 1E) will not adversely affect I&C circuits. The design, therefore, meets the above defined requirements of GDC 4 and 17 of 10 CFR

Part 50, Appendix A, and is acceptable. The staff's evaluation for the protection of circuits from EMI is addressed in Chapter 7.0 of this report. GE has included the above design commitments in Sections 8.3.3.1, 8.3.3.6.1.1, 8.3.3.6.1.2, 8.3.3.6.2.2.3, and 8.3.3.8.2 of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

In addition, based on the design commitment to meet the guidelines of IEEE 384-1981, GE indicated that administrative control of operations and maintenance activities will be used to control and limit introduction of potential hazards into cable chases and the control room area. Verification that GE included in a future SSAR amendment the statement that the COL applicant include these administrative controls in appropriate plant procedures was DFSER (SECY-92-349) COL Action Item 8.3.2.7-1. GE included this action item in Section 8.3.4.26 of SSAR Amendment 32, which is acceptable.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.3.2.7-1 related to the design description and the ITAAC for the separation of cables approaching and/or exiting cabinets/panels. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

8.3.2.8 Independence and Physical Separation of Equipment

This discussion addresses DSER (SECY-91-355) Open Item 36.

GE indicated in the draft SSAR revision dated April 3, 1992, and in discussions with the staff that Class 1E power supply and distribution systems, equipment, and components from the power supply through the power distribution panels of different Class 1E divisions will be separated by a 3-hour rated fire barrier and a missile barrier when the potential for missiles exist.

GDC 17 of 10 CFR Part 50, Appendix A, requires that Class 1E systems, equipment, and components have sufficient independence to perform their safety functions assuming a single failure. The staff concludes that a design which meets the above described commitment will ensure that any failure of or within one division of the Class 1E power system or its load group will not cause a loss of function in another division of the Class 1E power system. The design, therefore, meets the above defined requirements of GDC 17 of 10 CFR Part 50, Appendix A, and is acceptable.

Verification that GE provided the above design commitments in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.3.2.8-1. (Section 8.3.3.5 of this report addresses acceptable redundant circuits independence and protection from distribution system power panels to connected equipment loads which are not separated by fire and/or missile barriers.) GE included this commitment in Section 8.3.3.6.2.2.2 of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.3.2.8-1 related to the design description and the ITAAC for the independence/physical separation of equipment. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

8.3.2.9 Equipment, Cable, and Raceway Identification

This section addresses the staff's evaluation of GE's response to DSER (SECY-91-355) Open Item 39.

Identification of Power, Instrumentation, and Control Equipment, Cables, and Raceways

GE indicated in Section 8.3.1.3 of the draft SSAR revision dated April 3, 1992, that the ABWR electrical system design related to identification of power, control, and instrumentation systems, equipment, and components will meet the following commitments:

- The background of the nameplate for a division's equipment will be the same color as the electrical cable jacket markers and the cable raceway markers affiliated with that division.
- All exposed Class 1E and associated circuit raceways will be marked with the division color at 4.5 m (15 ft) intervals on; straight sections, at turning points, at points of entry to and exit from rooms and enclosed areas, at discontinuities, at pull boxes, and at origins and destinations of equipment.
- Class 1E and associated circuit raceways will be marked before their cables are installed.
- Before or during installation of all cables, for Class 1E and associated circuits, will be marked with the division color at intervals of approximately 1.5 m (5 ft).

- During installation cables for Class 1E and associated circuits that are routed in conduits will be marked with the division color at; points of entrance to and exit from conduits, at pull boxes, equipment, or enclosures where cables will or can be exposed. Such cables may not be marked at 1.5 m (5 ft) intervals inside conduits.

- All equipment, cables, and raceways will be marked in a manner of sufficient durability to be legible throughout the life of the plant and to facilitate initial verification that the installation conforms with the design separation criteria.

- All cables will be tagged with a permanent marker at each end with a unique identifying number (cable number) in accordance with the design drawings or cable schedule.

- The method used for identification will readily distinguish between different divisions of Class 1E systems, between Class 1E and non-Class 1E systems, and between associated cables of different divisions.

- Associated cables will be uniquely identified as such by a longitudinal stripe or other color-coded method.

- The color of the cable marker for associated cables will be the same color as the related Class 1E cable.

- Individual conductors (located inside panels or cabinets) exposed by stripping the jacket will be color coded or color-tagged at intervals not to exceed 0.3 m (1 ft) such that their division will still be discernable. Exceptions are permitted for individual conductors within cabinets or panels where all wiring is unique to a single division. Any non-divisional cable within such cabinets will be marked appropriately to distinguish it from the divisional cables.

- Class 1E wire bundles or cables (located inside panels or cabinets) will be identified in a distinct permanent manner at a sufficient number of points to readily distinguish between Class 1E wiring of different divisions and between Class 1E and non-Class 1E wiring.

- For a cabinet or compartment containing only Class 1E wiring of a single division, no distinctive identification will be required.

GDC 17 of 10 CFR Part 50, Appendix A, requires that Class 1E systems, equipment, and components have sufficient independence to perform their safety functions assuming a single failure. The staff concludes that a design which meets the above described commitments will provide reasonable assurance that cables will be installed in their affiliated divisional raceways and that Class 1E systems, equipment, and components will be installed in accordance with design-basis protection and independence requirements. Consequently, the design meets the above defined requirements of GDC 17 of 10 CFR Part 50, Appendix A, and is acceptable.

Verification that GE provided the above design commitments in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.3.2.9-1. GE has included these commitments in Section 8.3.3.5 of SSAR Amendment 32, which is acceptable. Therefore, DFSER Confirmatory Item 8.3.2.9-1 is resolved.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.3.2.9-1 related to the design description and the ITAAC for the identification of power, instrumentation, and control equipment, cables, and raceways. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

Identification of Neutron-Monitoring, Scram Solenoid, and dc Emergency Lighting Cables/Raceways

Section 8.3.1.3 of SSAR Amendment 10 indicated that cables of the neutron-monitoring system will be run in their own divisional conduits and cable trays separate from all other power, instrumentation, and control cables. Scram solenoid and dc emergency lighting cables will be similarly routed in their own conduits or cable trays separate from other cables. In addition, scram solenoid cables will be run in separate conduits for each rod scram group.

The following unique voltage class designations and markings will be used to help distinguish the neutron-monitoring and scram solenoid cables from other cable types:

- Neutron monitoring cables will be marked with a "VN" designation.
- Scram solenoid cables will be marked with a "VS" designation.

The staff concluded earlier in the review that the proposed identification of raceways and cables defined above does not meet the guidelines for the identification of raceways and cables defined by Section 6.1.2 of IEEE 384-1981 and position C10 of RG 1.75 (Rev. 2). The design did not include permanent, color raceway and cable markings to ensure that neutron monitoring, scram solenoid, and dc lighting cables will be installed in their associated raceways in accordance with design basis protection and independence requirements. This was DFSER (SECY-92-349) Open Item 8.3.2.9-2.

In response to this item, GE indicated that the "VN" or "VS" designation would be superimposed on the divisional color markings on the cable and raceway. Similarly, GE indicated a "DCL" designation would be superimposed on the color mark for dc emergency lighting cables and their dedicated raceways. The staff concludes that these designations and color coding meet the guidelines of RG 1.75 (Rev. 2). This aspect of the design therefore ensures that cables will be installed in their designated raceways in accordance with design basis requirements and is acceptable. GE included this design information in Sections 8.3.3.5.1.3 and 9.5.3 of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.3.2.9-3 related to the design description and the ITAAC for the identification of neutron-monitoring, scram solenoid, and dc emergency lighting cables/raceways. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

8.3.3 Protection of Electrical Systems, Equipment, and Components

Protection of Class 1E cable systems from non-Class 1E cable systems by spatial separation or barrier is addressed in Section 8.3.2. Protection of Class 1E cable systems by isolation devices is addressed in Section 8.3.4.

8.3.3.1 Protection of Electric Penetrations

This section addresses DSER (SECY-91-355) Open Item 43.

Item 7 of Section 8.3.1.4.1.2 of SSAR Amendment 10 indicated that power circuits passing through electric penetration assemblies are protected against overcurrent by redundant interrupting devices. In addition, GE's response to Question 435.31(b) of SSAR Amendment 10 indicated that the ABWR design requires that redundant interrupting devices be provided for electrical circuits passing through containment penetrations, if the maximum available fault current (including failure of upstream devices) is greater than the continuous current rating of the penetration.

Based on these design requirements, GE indicated that the proposed design will include redundant interrupting devices on all I&C circuits as well as power circuits that pass through containment when these circuits can produce sufficient energy (maximum available fault current) to exceed the current carrying capability of containment penetrations. When calculating maximum available fault current at the penetration, GE further indicated that current limiting devices will not be used in the calculation. For example, the worst case failure or shorting of the upstream or current limiting devices will be assumed as a given in the calculation. The staff concluded that the proposed design will include redundant protective devices (that is, current limiting and/or current interrupting) on all containment penetration circuits in accordance with guidelines in RG 1.63 (Rev. 3) as discussed below.

Based on the above, discussions with GE, and information presented in Section 8.3.4.4 of the draft SSAR revision dated April 3,. 1992, GE indicated that protection of electrical penetrations will meet the following commitments:

- The thermal capability of all electrical conductors within containment penetrations will be preserved and protected by two independent devices which meet requirements of IEEE 603-1980.

- The two independent devices will be located in separate panels or will be separated by barriers.

- The two independent devices will be independent such that failure of one will not adversely affect the other.

- The two independent devices will not depend on the same power supply to accomplish their safety-related function of protecting the containment penetration.

- Analysis will demonstrate that the maximum available fault current in the event of failure of either of two devices (that is short or open between input and output of a current limiting device or protective device fails open or closed) will be less than the maximum continuous current capacity of the conductor within the penetration and the maximum continuous current capacity rating of the penetration.

- Fault current clearing-time curves of the electrical penetrations' primary and secondary current interrupting devices plotted against the thermal capability (I^2t) curve of the penetration will show proper coordination.

- A simplified, one-line diagram will show the location of the protective or current limiting devices in the penetration circuit, the maximum available fault current of the circuit, and specific identification and location of power supplies used to provide external control power for tripping primary and backup electrical penetration breakers (if utilized).

* The devices will be capable of being functionally tested and calibrated.

In addition, Section 5.2 of IEEE 308-1980 indicates that components, equipment, or systems required to provide some protective action, such as containment integrity protection, are covered by safety system requirements defined in IEEE 603-1980. Based on GE's commitment to meet the guidelines of IEEE 308-1980, GE indicated that the devices used to protect containment integrity will be covered by all safety system requirements defined in IEEE 603-1980.

GDC 50 of 10 CFR Part 50, Appendix A, requires that the reactor containment structure, including penetrations, be designed so that the containment structure can, without exceeding the design leakage rate, accommodate the calculated pressure, temperature, and other environmental conditions resulting from any loss-of-coolant accident. The staff concludes that a design that meets above described commitments will provide protection of containment electrical penetration such that a failure of a circuit (i.e., the single failure during a design basis event) and single failure of a device providing protection to containment penetrations will not cause loss of containment integrity. The design, therefore, meets the above defined requirements of GDC 50 of 10 CFR Part 50, Appendix A, and is acceptable.

Verification that GE provided the above design commitments in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.3.3.1-1. GE has included these commitments in Sections 8.3.1.2(2)(c), 8.3.3.6.1.2, and 8.3.3.7 of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

In addition, GE indicated that the protective devices will periodically be tested to demonstrate their functional capability to perform their required safety functions. Verification that GE included in a future SSAR amendment a statement that the COL applicant will include periodic test and calibration of protective devices in appropriate procedures was DFSER (SECY-92-349) COL Action Item 8.3.3.1-1. GE included this action item in Section 8.3.4.4 of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

<u>Certified Design Material</u>

DFSER (SECY-92-349) Open Item 8.3.3.1-1 related to the design description and the ITAAC for the protection of electric penetrations. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

8.3.3.2 Design/Qualification of Electrical Equipment

Section 8.3.1.2.4 of the draft SSAR revision dated April 3, 1992, in response to DSER (SECY-91-355) Open Item 45, indicated that all Class 1E equipment is designed to operate in its normal service environment as well as in the environment expected in the area in which it is located during and after any design-basis event. In addition, by committing to meet the guidelines of IEEE 308-1980 (Table 1.8-21 of SSAR Amendment 17), GE indicated compliance with the requirements of Section 5.9, Equipment Qualifications of IEEE 308-1980. Section 5.9 of IEEE 308-1980 requires that all Class 1E power system equipment shall be qualified in accordance with IEEE 323-1974 to substantiate that it will be capable of meeting the performance requirements specified in the design basis.

Based on this commitment, information presented in the draft SSAR revision dated April 3, 1992, and discussions with the staff, GE indicated that each type of Class 1E equipment will be

* qualified by; analysis, successful use under similar conditions, or by actual test to demonstrate its ability to perform its function under normal and design-basis event environmental and operational conditions

* designed and qualified to survive the combined effects of temperature, humidity, radiation, and other conditions associated with a LOCA or other design-basis event environments at the end of their qualified and/or design life

* qualified to IEEE 344-1987, "Recommended Practices for Seismic Qualifications of Class 1E Equipment for Nuclear Power Generating Stations"

* qualified by test and/or analyzed to demonstrate its ability to meet its performance requirements during and following the design-basis seismic event

* located in seismic Category I structures

* seismically supported

* designed and qualified to operate within allowable design basis limits; (for example, able to operate for predetermined time when subject to voltage below 90 percent, to operate for a predetermined time when

voltage is below 70 percent, or to operate continuously when subjected to voltage variations of $\pm$ 10 percent of nominal).

All structures, systems, equipment, components, pipes and loads that are not Class 1E and whose failure could possibly prevent Class 1E systems, equipment, components, and circuits including connected loads from performing their required safety function will be appropriately designed and qualified to not fail in the normal and design-basis event environment for which the structures, systems, equipment, components, pipes and loads will be expected to function. In addition, variations of voltage, frequency, and waveform in the Class 1E power systems, during any mode of plant operation, will not degrade the performance of any safety-related system load below an acceptable level. The dc system equipment and loads will be designed and qualified to perform their required safety-related function while operating with voltages between 100 to 140 volts at the dc system's 125-volt distribution panels.

GDC 2 and 4 of 10 CFR Part 50, Appendix A, requires that Class 1E systems, equipment, and components be designed (1) to withstand the effects of natural phenomena without loss of capability to perform their safety functions, (2) to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, and (3) be appropriately protected against dynamic effects. GDC 17 of 10 CFR Part 50, Appendix A, requires that the Class 1E systems, equipment, and components have sufficient independence to perform their safety functions assuming a single failure. The staff concludes that a design that meets the above commitments will provide protection to Class 1E systems, equipment, and components during design basis events such that there will be reasonable assurance that Class 1E systems will be capable of performing their required function. The electrical system design, therefore, meets the above defined requirements of GDC 2, 4, and 17 of 10 CFR Part 50, Appendix A, and is acceptable.

Verification that GE provided the above design commitments in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.3.3.2-1. In response to this item, GE further modified their design to indicate that the Class 1E systems, equipment, and components conform to seismic Category I requirements and are housed in seismic Category I structures in accordance with the above design information except for some control sensors associated with the reactor protection system and the leak detection system which are housed in the turbine building which is not a seismic Category I structure. For these exceptions, GE has indicated that the feeders between Class 1E systems located in seismic Category I structures (safety class structures) and systems that are not located in seismic Category I structures will be provided with Class 1E protective devices (such as coordinated circuit breakers) located in a seismic Category I structure. By committing to meet the guidelines of IEEE 308-1980 (Table 1.8-21) of SSAR Amendment 17), GE indicated compliance with the requirements of Section 5.2 of IEEE 308-1980. Section 5.2 of IEEE 308-1980 requires that components, equipment, or systems utilized to provide isolation protection are covered by safety system design requirements defined in IEEE 603-1980. The staff concluded that the Class 1E classification for the protective devices utilized to provide an isolation protection function in the proposed modified design meets the guidelines of Section 5.2 of IEEE 308-1980. The staff concludes that a design that meets the above described commitments will also provide protection to Class 1E systems such that there will be reasonable assurance that Class 1E systems will be capable of performing their required function. The electrical system design, therefore, meets the above defined requirements of GDC 2, 4, and 17 of 10 CFR Part 50, Appendix A, and is acceptable. GE has included the above design commitments in Sections 3.11.2, 8.1.3.1.1.1, 8.3.1.1.5, 8.3.1.1.7, 8.3.2.1.3.1, and 8.3.3.4 of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.3.3.2-1 related to the design description and the ITAAC for the design/qualification of Class 1E electrical equipment. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

8.3.3.3 Seismic Qualification of Light Bulbs

In response to DSER (SECY-91-355) Open Item 45, GE provided draft SSAR revision dated April 3, 1992, Section 8.3.2.2.2, identifying an exception to the requirement that all Class 1E equipment is seismically qualified. GE indicated that the safety-related dc standby lighting system is powered from the Class 1E dc system and that the lighting system circuits from the Class 1E dc system power source to the lighting fixtures will be treated as Class 1E circuits (that is, these circuits will be classified as associated) and will be routed in seismic Category I raceways. The lighting fixtures themselves will not be seismically qualified, but will be seismically supported. The bulbs cannot be seismically qualified.

To justify this exception, GE indicated that bulbs can only fail open, and, therefore, do not represent a hazard to the Class 1E power source.

Based on subsequent discussions with GE, the staff determined that lighting fixtures will be seismically qualified. Light bulbs may fail during and/or following a seismic event thereby extinguishing the light; however, the light bulbs will be replaceable. (The staff's evaluation of lighting requirements is in Section 8.3.5 of this report.) In addition, bulbs will not fail in a manner that could cause failure of other safety-related systems and will not become a hazard to personnel or safety-related equipment during or following a seismic event.

GDC 4 of 10 CFR Part 50, Appendix A, requires that Class 1E systems, equipment, and components be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents. GDC 17 of 10 CFR Part 50, Appendix A, requires that the Class 1E systems, equipment, and components have sufficient independence to perform their safety functions assuming a single failure. The staff concludes that lighting circuits that are treated as Class 1E except for the seismic qualification of light bulbs and which include seismically qualified light bulb fixtures will provide protection to Class 1E systems such that there will be reasonable assurance that the lighting systems will not prevent the Class 1E systems from performing their required function. Consequently, the design meets the above defined requirements of GDC 4 and 17 of 10 CFR Part 50, Appendix A, and is acceptable.

Verification that GE provided the above design commitments in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.3.3.3-1. In response to this item, GE indicated that lighting fixtures are seismically supported but not seismically qualified as indicated above. GE further indicated that overcurrent protective devices and their coordination would provide protection and isolation of the Class 1E power supply from possible failure of non-seismically qualified fixtures during a seismic event. The staff concluded that Class 1E systems will be adequately protected from failure of the non-seismically qualified fixtures as well as the non-seismically qualified light bulbs during a seismic event. By committing to meet the guidelines of IEEE 308-1980 (Table 1.8-21 of SSAR Amendment 17), GE indicated compliance with the requirements of Section 5.2 of IEEE 308-1980. Section 5.2 of IEEE 308-1980 requires that components, equipment, or systems utilized to provide isolation protection are covered by safety system design requirements defined in IEEE 603-1980. The staff concluded that the overcurrent protective devices and their coordination which are utilized to provide an isolation protection function to the Class 1E system from failure of the non-seismically qualified fixtures will meet the guidelines of Section 5.2 of IEEE 308-1980 and thus will meet safety system design requirements defined in IEEE 603-1980.

The staff concludes that lighting circuits that are treated as Class 1E except for seismic qualification of light bulbs and fixtures and include protective devices which meet safety system requirements will provide protection to Class 1E systems such that there will be reasonable assurance that the lighting systems will not prevent the Class 1E systems from performing their required function. Consequently, the design meets the above defined requirements of GDC 4 of 10 CFR Part 50, Appendix A, and is acceptable. GE has included the above design commitments in Sections 8.3.2.2.2, 9.5.3.1.1, 9.5.3.2.2.1, and 9.5.3.2.3.1 of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.3.3.3-1 related to the design description and the ITAAC for the protection of Class 1E systems from the non-seismically qualified light fixtures and bulbs. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

8.3.3.4 Submergence

Item (6) of Section 8.3.1.4.2.3.2 of SSAR Amendment 10 stated that any RPS or ESF electrical equipment and/or raceway located in the suppression pool level swell zone will be designed to satisfactorily complete its function before being rendered inoperable due to exposure to the environment created by the level phenomena. In response to staff Question 435.36 of SSAR Amendment 10, GE identified electrical equipment that may be submerged as a result of suppression pool level swell phenomena or as a result of a LOCA. GE further indicated that the design specifications associated with this electric equipment would require that electrical terminations be sealed such that equipment operation would not be impaired by submersion. However, GE did not specifically address the qualification of this equipment in accordance with the guidelines of Section 4.7 of IEEE 308-1974.

Based on information presented, it appeared that electrical equipment subject to submergence was not qualified and only partially designed for submergence. This conclusion contradicted Section 8.3.1.2.1 of Amendment 10 to the SSAR which stated that all Class 1E equipment is qualified.

The staff was concerned that equipment failure due to submergence could adversely affect the safe operation of the plant and could adversely affect Class 1E power sources serving this equipment. This was DSER (SECY-91-355) Open Item 46.

GE indicated in Section 8.3.1.4.2.3.2 of the draft SSAR revision dated April 3, 1992, that the only Class 1E equipment located in the suppression pool level swell zone will be suppression pool temperature monitors, which have their terminations sealed such that their operation will not be impaired by submersion due to pool swell or a LOCA. Consistent with their Class 1E status, these devices will also be qualified to the requirements of IEEE 323 for the environment in which they are located.

GE also indicated that all Class 1E equipment (including affiliated systems and component parts) subject to submergence in the suppression pool level swell zone will either be designed and qualified to perform its required safety function without failing while submerged. If this is not possible, all Class 1E equipment will be appropriately protected from submergence and will be appropriately designed and qualified to perform its required safety function without failing in the normal and design-basis event environment for which the equipment is expected to operate.

GDC 4 of 10 CFR Part 50, Appendix A, requires that Class 1E systems, equipment, and components be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents. GDC 17 of 10 CFR Part 50, Appendix A, requires that the Class 1E systems, equipment, and components have sufficient independence to perform their safety function assuming a single failure. The staff concludes that a design which meets the above described commitments will provide reasonable assurance that the components subject to submergence will be capable of performing their required function. The design, therefore, meets the above defined requirements of GDC 4 and 17 of 10 CFR Part 50, Appendix A, and is acceptable.

Verification that GE provided the above design commitments in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.3.3.4-1. GE has included these commitments in Section 8.3.3.6.2.3.2(6) of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.3.3.4-1 related to the design description and the ITAAC for the submergence of Class 1E electrical equipment in the suppression pool swell zone. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

8.3.3.5 Redundant Class 1E Systems Subject to Common Design Basis Environments

Section 8.3.3.1 of SSAR Amendment 10 stated that the electrical cable installation is such that direct impingement of fire suppressant will not prevent safe reactor shutdown. It was not clear whether impingement of fire suppressant would or would not cause failure of cable systems. The staff was concerned that cables and other electric equipment might not be designed and qualified to perform their safety function while being subjected to the direct impingement of fire suppressant.

The draft information provided by GE on September 4, 1991, indicated that cables and other electric equipment will not be designed and qualified to perform their safety function while being subjected to the direct impingement of fire suppressant. In justifying this lack of design and qualification, GE indicated that redundant divisions are provided. In the event that the cable system or equipment in one division fails due to fire suppressant impingement, and single failure occurs in a second division, safe shutdown of the plant (the required minimum safety function) can be achieved by the third division. This was DSER (SECY-91-355) Open Item 47.

In the draft SSAR revision dated April 3, 1992, GE indicated that where fire suppressant impinged on cables of more than one division, each case had been analyzed and found to be acceptable for the worst case failure mode.

After reviewing information presented in Chapter 8 of the draft SSAR revision dated April 3, 1992, and Section 9A.5 through Amendment 20 to the SSAR, the staff was unable to reach conclusions as to the acceptability of the level of protection to be afforded Class 1E power systems due to failure of redundant Class 1E components that may be subjected to environments of the same design-basis event (including fire, fire suppressant, and non-seismic structures) for which they may not be designed or qualified. Information and design commitments presented in Chapter 8 and Section 9A.5 were found to be inconsistent. This was DFSER (SECY-92-349) Open Item 8.3.3.5-1.

In response to this item, GE provided the results of an analysis in SSAR Section 9A.5. These results assured that when common mode failure of redundant safety systems and their supporting Class 1E systems occurs because of (fire, fire suppressant impingement, or seismic event), sufficient remaining safety systems not affected by these same events are available to accomplish the required safety function. The results of this analysis, indicated that Class 1E power systems will be protected by circuit protective devices and their coordination or current limiting devices, in order to assure the continued operation of Class 1E systems in accordance with their required safety function. By committing to meet the guidelines of IEEE 308-1980 (Table 1.8-21 of SSAR Amendment 17), GE indicated compliance with the requirements of Section 5.2 of IEEE 308-1980. Section 5.2 of IEEE 308-1980 requires that components, equipment, or systems utilized to provide isolation protection are covered by safety system design requirements defined in IEEE 603-1980. The staff concluded that the overcurrent protective devices and their coordination or current limiting devices which are utilized to provide an isolation protection function to the Class 1E systems will meet the guidelines of Section 5.2 of IEEE 308-1980 and thus will meet safety system design requirements defined in IEEE 603-1980.

GDC 4 of 10 CFR Part 50, Appendix A, requires that Class 1E systems, equipment, and components be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents. GDC 17 of 10 CFR Part 50, Appendix A, requires that the Class 1E systems, equipment and components have sufficient independence to perform their safety function assuming a single failure. The staff concludes that a design that meets the above commitments will adequately protect Class 1E systems such that there will be reasonable assurance that redundant system will not fail. Consequently, the design meets the above defined requirements of GDC 4 and 17 of 10 CFR Part 50, Appendix A, and is acceptable. GE has included the results of this analysis and the above commitments in Sections 8.3.1.2(2)(c) and 9A.5 of SSAR Amendment 32, which is acceptable.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.3.3.5-2 related to the design description and the ITAAC for the protection of redundant Class 1E systems subject to common design basis environments. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

8.3.3.6 Associated Circuits

Based on discussions with GE and information presented in Section 8.3.1.1.5.1 of the draft SSAR revision dated April 3, 1992, in response to DSER (SECY-91-355) Open Item 38, GE indicated that the ABWR electrical system design related to associated circuits will meet the following commitments:

- Associated circuits will remain with or be physically separated in the same manner as those Class 1E circuits with which they are associated;

  or

  Associated circuits, will remain with or be physically separated in the same manner as those Class 1E circuits with which they are associated, from the Class 1E equipment to and including an isolation device.

- Associated circuits (including their isolation devices or their connected loads without isolation devices) will be subject to all requirements placed on Class 1E circuits.

- Non-Class 1E circuits powered from a Class 1E power supply will be limited to power circuits related to the FMCRDs and lighting systems.

GDC 4 of 10 CFR Part 50, Appendix A, requires that Class 1E systems, equipment, and components be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents. GDC 17 of 10 CFR Part 50, Appendix A, requires that the Class 1E systems, equipment, and components have sufficient independence to perform their safety function assuming a single failure. The staff concludes that a design which meets the above described commitments will provide adequate protection and independence for Class 1E systems. The design, therefore, meets the above defined requirements of GDC 4 and 17 of 10 CFR Part 50, Appendix A, and is acceptable.

Verification that GE provided the above design commitments in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.3.3.6-1. GE has included the above design commitments in Sections 8.3.3.1 and 8.3.1.1.1 of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.3.3.6-1 related to the design description and the ITAAC for associated circuits. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

The staff determined that a commitment was required in the design description of electrical systems that states that non-Class 1E circuits connected to the Class 1E system shall be limited to circuits in the FMCRD and lighting subsystems. This was Open Item F8.3.3.6-1 identified in the advance version of the SER. GE has included this commitment in the design description of Section 2.12.1 and 2.12.12 of the certified design material. The staff finds this acceptable. This resolved Open Item F8.3.3.6-1.

8.3.3.7 Diesel Generator Protective Relaying Bypass

Section 8.3.1.1.6.4 of SSAR Amendment 10 indicated that the following identified protective relaying will trip the diesel generator and will be retained under LOCA conditions. This relaying included the generator differential, bus differential, engine over speed, low diesel cooling water pressure (two out of two sensors), and low differential pressure of secondary cooling water (two out of two sensors). Other diesel generator protective trips will be bypassed during LOCA conditions.

GE responded to DSER (SECY-91-355) Open Item 49 in Section 8.3.1.1.6.4 of the draft SSAR revision dated April 3, 1992, where they indicated that only the generator differential relays and engine overspeed trip would be retained under accident conditions. Other protective relays, such as loss of excitation, antimotoring (reverse power) overcurrent voltage restraint, low jacket water pressure, high jacket water temperature, and low lube oil pressure are automatically removed from the tripping circuits during LOCA conditions.

By committing to meet the guidelines of IEEE 308-1980 (Table 1.8-21 of SSAR Amendment 17), GE indicated compliance with the requirements of Section 5.2 of IEEE 308-1980. Section 5.2 of IEEE 308-1980 requires that components, equipment, or systems that provide some protection to Class 1E systems are covered by safety system design requirements defined in IEEE 603-1980. The staff concluded that the diesel generator protective relying bypass circuitry which are utilized to provided protection to the Class 1E systems will meet the guidelines of Section 5.2 of IEEE 308-1980 and thus will meet safety system design requirements defined in IEEE 603-1980. By committing to meet the guidelines of RG 1.9 (Rev. 3) (Table 1.8-20 of SSAR Amendment 33), GE indicated compliance with the requirements of Position C-1.8 of the RG. Position C-1.8 requires that the diesel generator protective relay bypass system design include the capability for testing the status and operability of the bypass circuitry, for alarming in the control room for abnormal values of all bypass parameters, and for resetting the trip bypass function manually (automate reset is not acceptable). Based on the above commitment and the commitment to meet the guidelines of Position C-1.8 of RG 1.9 (Rev. 3) and Section 5.2 of IEEE 308-1980 contained in Section 8.3.1.2.1(2)(b) of the draft SSAR revision dated April 3, 1992, and Table 1.8-21 of SSAR Amendment 17, information presented in the draft SSAR revision dated April 3, 1992, and discussions, GE's indicated that:

- the design of the bypass circuitry will meet all the requirements of IEEE 603-1980
- abnormal values of all bypassed parameters will be alarmed in the control room so that the control room operator can react appropriately to the abnormal condition on the diesel generator unit
- the trip bypass function will be capable of being reset manually (capability for automatic reset is not acceptable)
- the protective relaying and its bypass circuitry will have the capability to be tested periodically

GDC 4 of 10 CFR Part 50, Appendix A, requires that Class 1E systems, equipment, and components be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents. GDC 17 of 10 CFR Part 50, Appendix A, requires (1) that offsite and onsite power systems have sufficient capacity and capability to permit safety systems to perform their required safety function and (2) that the Class 1E systems, equipment and components have sufficient independence and testability to perform their safety functions assuming a single failure. GDC 18 of 10 CFR Part 50, Appendix A, requires that electric power systems important to safety be designed to permit appropriate periodic inspection and testing of important areas and features. The staff concludes that a design which meets the above described commitments will provide reasonable assurance that the protective relaying (to be installed on Class 1E diesel generators to protect the diesel generator from failure) will be bypassed during accident conditions so that the diesel generator will not be prevented from performing its required safety function under accident conditions due to operation or failure of the protective scheme. Consequently, the design meets the above defined requirements of GDC 4, 17, and 18 and is acceptable.

Verification that GE provided the above design commitments in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.3.3.7-1. GE included this information in Section 8.3.1.1.6.4 of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

In addition, GE indicated that the protective relays and their bypass circuitry will be periodically tested.

Verification that GE specified in a future SSAR amendment that the COL applicant will include periodic testing of protective relays and their bypass circuitry in appropriate procedures was DFSER (SECY-92-349) COL Action Item 8.3.3.7-1. GE included this action item in Section 8.3.4.22 of SSAR Amendment 32, which is acceptable.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.3.3.7-1 related to the design description and the ITAAC for the diesel generator protective relaying and their bypass. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, the above open item is resolved.

8.3.3.8 Thermal Overloads

GE's response to Question 435.60 in SSAR Amendment 10 indicated that thermal overload protection for Class 1E motor operated valves (MOVs) is in effect only when the MOVs are in test mode. The thermal overload protection is bypassed at all other times by means of closed contacts in parallel with the thermal overload contacts.

GE responded to DSER (SECY-91-355) Open Item 50, in Section 8.3.1.2.1(2)(g) and Section 8.3.2.2.2(2)(f) of the draft SSAR revision dated April 3, 1992, where it was indicated that the thermal overload protection for Class 1E MOVs will be in effect during normal plant operation but the overloads will be bypassed under accident conditions as specified by Position 1.(b) of RG 1.106 (Rev. 1).

By committing to meet the guidelines of IEEE 308-1980 (Table 1.8-21 of SSAR Amendment 17), GE indicated compliance with the requirements of Section 5.2 of IEEE 308-1980. Section 5.2 of IEEE 308-1980 requires that components, equipment, or systems that provide some protection to Class 1E systems are covered by safety system design requirements defined in IEEE 603-1980. Based on this commitment, the staff concluded that the thermal overload protection bypass circuitry for Class 1E MOVs (which is utilized to provide protection to motors of MOVs in the Class 1E system) will meet the guidelines of Section 5.2 of IEEE 308-1980 and thus the safety system design requirements defined in IEEE 603-1980. In addition, GE indicated that the thermal overload and its bypass circuitry will have the capability to be tested periodically.

GDC 4 of 10 CFR Part 50, Appendix A, requires that Class 1E systems, equipment, and components be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents. GDC 17 of 10 CFR Part 50, Appendix A, requires (1) that offsite and onsite power systems have sufficient capacity and capability to permit safety systems to perform their required safety function and (2) that the Class 1E systems, equipment, and components have sufficient independence to perform their safety function assuming a single failure. GDC 18 of 10 CFR Part 50, Appendix A, requires that Class 1E power systems be designed to permit appropriate periodic inspection and testing of important areas and features. The staff concludes that a design which meets the above described commitments will provide reasonable assurance that the thermal overload protection to be installed on Class 1E motor operated valves will be bypassed during accident conditions so that the Class 1E valve motor will not be prevented from performing its required safety function under accident conditions due to operation or failure of the thermal overload devices. Consequently, the design meets the above defined requirements of GDC 4, 17, and 18 and is acceptable.

Verification that GE provided the above design commitments in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.3.3.8-1. GE has included these commitments in Sections 8.3.1.2(2)(g) and 8.3.2.2.2(2)(f) of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

In addition, GE indicated that the thermal overloads and their bypass circuitry will periodically be tested. Verification that GE specified in a future SSAR amendment that the COL applicant will include periodic testing of thermal overloads and their bypass circuitry in appropriate procedures was DFSER (SECY-92-349) COL Action Item 8.3.3.8-1. GE included this action item in Section 8.3.4.24 of SSAR Amendment 32, which is acceptable.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.3.3.8-1 related to the design description and the ITAAC for the thermal overloads. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

8.3.3.9 Breaker Coordination

Section 8.3.1.1.2.1 of SSAR Amendment 10 stated that tripping of the Class 1E bus feeder breaker is normal for faults that occur on its Class 1E loads. The staff disagreed with this statement. Class 1E load breakers should be coordinated with the Class 1E bus feeder breaker so that faults which occur on its Class 1E loads will, to the extent possible, not cause the bus feeder breaker to trip. A design which utilizes coordinated breakers minimizes the potential for loss of safety-related systems. This was DSER (SECY-91-355) Open Item 51.

The draft information provided by GE on September 4, 1991, revised the SSAR to delete the statement that tripping of the bus supply breaker is normal for faults that occur on its Class 1E loads. GE further indicated that the Class 1E load and bus supply breakers will be coordinated.

GDC 4 of 10 CFR Part 50, Appendix A, requires that Class 1E systems, equipment, and components be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents. GDC 17 of 10 CFR Part 50, Appendix A, requires (1) that offsite and onsite power systems have sufficient capacity and capability to permit safety systems to perform their required safety function and (2) that the Class 1E systems, equipment, and components have sufficient independence to perform their safety function assuming a single failure. The staff concludes that a design which meets the above commitment will minimize to the extent practicable the effect of single Class 1E components or equipment failure. The design, therefore, meets the above defined requirements of GDC 4 and 17 of 10 CFR Part 50, Appendix A, and is acceptable.

Verification that GE provided the above design commitment in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.3.3.9-1. GE has included these commitments in Section 8.3.1.1.1 of SSAR Amendment 32, which is acceptable.

8.3.3.10 Protective Relaying

This section addresses DSER (SECY-91-355) Open Item 52.

Experience with protective relay applications has established that relay trip setpoints will drift with conventional relays. Setpoint drift at nuclear power plants has resulted in premature tripping of redundant, safety-related pump motors when they were required to be operative. While the staff recognizes the basic need for proper fault protection for feeders and equipment (and while such protection may be required for some design-basis events such as fire), the total non-availability of redundant safety systems due to spurious trips of protective relays is not acceptable. GE responded to this concern (Question 435.58 of SSAR Amendment 10) and indicated that loads, such as motors, will be designed with sufficient current carrying capability or overload margins so that setpoints of protective devices can be set sufficiently above the operating current point of loads to allow for setpoint drift. The use of loads, such as motors, with sufficient overload margins resolves the staff's concern if one assumes the following:

- Specific design parameters clearly define the overload margin requirements with respect to; protective device trip setpoints, the margin between the trip setpoint and operating current point of loads, setpoint drift, and the margin between the trip setpoint and overload rating of loads.
- The protective device trip setpoint is periodically verified and calibrated.
- The protective device is periodically subjected to a functional test to demonstrate that it does not trip at its design rating (the normal operating current of load plus margin) and that it does trip when subjected to a fault current.

The staff was concerned earlier in the review that the ABWR design may not satisfy the above assumptions.

By committing to meet the guidelines of IEEE 308-1980 (Table 1.8-21 of SSAR Amendment 17), GE indicated compliance with the requirements of Section 5.2 of IEEE 308-1980. Section 5.2 of IEEE 308-1980 requires that components, equipment, or systems that have no direct safety function and are only provided to increase the availability or reliability of the Class 1E power systems shall meet those safety system design requirements defined in IEEE 603-1980 to assure that those components, equipment, and systems do not degrade the Class 1E power system below an acceptable level. Based on the design commitment to meet the guidelines of Section 5.2 of IEEE 308-1980, information presented in the draft SSAR revision dated April 3, 1992, and discussions, GE indicated that:

- protective relaying design will meet the above defined assumptions (that is, there will be protective device trip setpoint margin and capability to functionally test and calibrate the protective relaying)
- protective relaying—as well as all other components, equipment, and systems within the Class 1E power system (that have no direct safety function and are only provided to increase the availability or reliability of the Class 1E power systems) including the diesel generator protective relaying and thermal overload protective devices which are bypassed during accident conditions—will meet those requirements of IEEE 603-1980 that assure that the consequences of

any operation or failure is acceptable to the Class 1E power system

GDC 4 of 10 CFR Part 50, Appendix A, requires that Class 1E systems, equipment, and components be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents. GDC 17 of 10 CFR Part 50, Appendix A, requires (1) that offsite and onsite power systems have sufficient capacity and capability to permit safety systems to perform their required safety function and (2) that the Class 1E systems, equipment, and components have sufficient independence and testability to perform their safety functions assuming a single failure. GDC 18 of 10 CFR Part 50, Appendix A, requires that Class 1E power systems be designed to permit appropriate periodic inspection and testing of important areas and features. The staff concludes that a design that meets the above commitments will assure that when these components, equipment, or systems are used that have no direct safety function their operation or failure will not significantly reduce the capability of the Class 1E power system from performing its safety function when required. The design, therefore, meets the above defined requirements of GDC 4, 17, and 18 of 10 CFR Part 50, Appendix A, and is acceptable.

Verification that GE provided the above design commitments in a future SSAR amendment was Confirmatory Item 8.3.3.10-1. GE has included these commitments in Sections 8.3.1.2(2)(c) and 8.3.2.2.2(2)(b) of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

In addition, GE indicated that the protective relaying will periodically be tested. Verification that GE specified in a future SSAR amendment that the COL applicant will include these periodic tests in appropriate plant procedures was DFSER (SECY-92-349) COL Action Item 8.3.3.10-1. GE included this action item in Section 8.3.4.27 of SSAR Amendment 32, which is acceptable.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.3.3.10-1 related to the design description and the ITAAC for protective relaying. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

8.3.3.11 Fault Interrupting Capacity

Section 8.3.1.1.5.2(4) of SSAR Amendment 10 stated that the interrupting capacity of switchgear, load centers, MCCs, and distribution panels is compatible with the short-circuit current available at the Class 1E buses. It was not clear whether the interrupting capacity of this equipment would be equal to or greater than the maximum available fault current to which it would be exposed for all modes of operation (for example, with the diesel generator operating in parallel with the grid).

Section 8.3.1.1.5.2(4) of the draft SSAR revision dated April 3, 1992, in response to DSER (SECY-91-355) Open Item 53, indicated that the interrupting capacity of switchgear, load centers, MCCs, and distribution panels will be equal to or greater than the maximum available fault current to which the equipment is exposed under all modes of operation.

GDC 4 of 10 CFR Part 50, Appendix A, requires that Class 1E systems, equipment, and components be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents. GDC 17 of 10 CFR Part 50, Appendix A, requires (1) that offsite and onsite power systems have sufficient capacity and capability to permit safety systems to perform their required safety function and (2) that the Class 1E systems, equipment, and components have sufficient independence and testability to perform their safety functions assuming a single failure. The staff concludes that a design which meets the above described design commitments will have sufficient capacity and capability to interrupt the worst case fault. The design, therefore, meets the above defined requirements of GDC 4 and 17 and is acceptable.

Verification that GE provided the above design commitments in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.3.3.11-1. GE has included these commitments in Section 8.3.1.1.5(4) of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.3.3.11-1 related to the design description and the ITAAC for the fault interrupting capacity of Class 1E protective equipment. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

8.3.3.12 Control of Design Parameters for Motor Operated Valves

Valve problems such as excess friction which may be caused by excessively tight packing can result in an operational condition where the current drawn will exceed the design rating or capability of the insulation system used in the valve motor winding. Operating experience has shown that excessive current, if undetected during operation, can cause premature or unexpected failure when the valve is next operated. The ABWR SSAR did not present methods, design provisions, alarms, or procedures to ensure that the valve motor will not be operated with excessive currents without operator knowledge (or will always be operated within their design limits).

The draft information provided by GE on September 4, 1991, indicated that thermal overloads will provide protection at all times for non-Class 1E MOVs and will provide protection during testing or maintenance for Class 1E MOVs. At all other times, the Class 1E MOVs will not be protected. The staff was concerned by this lack of protection for Class 1E MOVs. This was DSER (SECY-91-355) Open Item 54.

Section 8.3.1.2.1(2)(g) of the draft SSAR revision dated April 3, 1992, indicated that Class 1E MOVs which are required to open and/or close to satisfy their safety function, will have the thermal overload protective device on the valves' motor in force during normal plant operation. The thermal overload protective device for these valves will be bypassed under accident conditions provided that safety function completion is not jeopardized or that other safety systems are not degraded as per Regulatory Position 1.(b) of Revision 1 of RG 1.106.

GDC 4 of 10 CFR Part 50, Appendix A, requires that Class 1E systems, equipment, and components be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents. GDC 17 of 10 CFR Part 50, Appendix A, requires (1) that offsite and onsite power systems have sufficient capacity and capability to permit safety systems to perform their required safety function and (2) that the Class 1E systems, equipment, and components have sufficient independence and testability to perform their safety functions assuming a single failure. The staff concludes that a design, which keeps the thermal overload in force during normal plant operation as well as during test and maintenance in accordance with the above described commitments, will provide reasonable assurance that the MOV will not be operated with excessive currents without operator knowledge (or will be operated within their design limits). The design, therefore, meets the above defined requirements of GDC 4 and 17 of 10 CFR Part 50, Appendix A, and is acceptable.

Verification that GE provided the above design commitments in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.3.3.12-1. GE included these commitments in Sections 8.3.1.2(2)(g) and 8.3.2.2.2(2)(f) of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.3.3.12-1 related to the design description and the ITAAC for the control of design parameters for MOVs. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

8.3.3.13 Protection of Cable Systems from Internally Generated Fires

Section 8.3.3.2 of SSAR Amendment 10 indicated that spatial separation is used as a method of preventing the spread of fire between adjacent cable trays of different divisions (for example, inside primary containment). The design objective should be to separate cable trays of different divisions with structural fire barriers such as floors, ceilings, and walls. Where such barriers are not possible, divisional trays should be separated spatially by .9 m (3 ft) horizontally and 1.5 m (5 ft) vertically. Where this .9 by 1.5 m (3 by 5 ft) spatial separation is not possible, fire-rated barriers are used to separate divisional cable trays.

GDC 17 of 10 CFR Part 50, Appendix A, requires that the onsite electric distribution systems have sufficient independence to perform their safety function assuming a single failure.

For a fire initiated by a cable fault within one division, the staff concludes that a design which meets the above described commitments will provide reasonable assurance that a fire in one division will not propagate to a redundant division. The design, therefore, meets the above defined requirements of GDC 17 of 10 CFR Part 50, Appendix A, and is acceptable. This was DSER (SECY-91-355) Open Item 24.

In the draft SSAR revision dated April 3, 1992, GE revised Section 8.3.3.2 of SSAR Amendment 10 to indicate that separation will be achieved by using totally enclosed raceways separated by a least 2.54 cm (1 in.) when spacial separation is less than .9 by 1.5 m (3 by 5 ft). The staff concludes that a design which meets this commitment will provide reasonable assurance that a fire initiated in one division will not propagate to a redundant division. Consequently, the design meets the above defined requirements of GDC 17 of 10 CFR Part 50, Appendix A, and is acceptable.

Verification that GE has provided the above design commitment in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.3.3.13-1. GE has included these commitments in Section 8.3.3.8.2 of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.3.2.5-1 related to the design description and the ITAAC for separation of raceways. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

8.3.3.14 Electrical Protection for Scram and MSIV solenoids Electrical Protection Assemblies (EPAs) (The need for EPAs was established as part of the staff's evaluation of I&C Systems. I&C Systems are addressed in Chapter 7 of this SER.)

A generic letter issued to all operating BWRs on September 24, 1980, requires two independent EPAs on the output of RPS power supplies. Two EPAs are required to satisfy the single-failure criterion for non-fail-safe type failures, which may be caused by under-voltage, over- voltage, and under-frequency conditions.

GE's response to Question 435.7 included in SSAR Amendment 10 indicated that EPAs will not be used in the ABWR design because of special design features. These special features included voltage and frequency monitoring, automatic transfer of power supply input sources when the voltage or frequency exceeds pre-established limits, control room alarm for abnormal conditions, operator action in response to alarm of abnormality, and design and qualification of equipment to preclude failure after operation for a period of time under the allowable abnormality of voltage and frequency.

The staff determined that these special features should provide reasonable assurance that any abnormality in voltage and frequency (which can cause failure of fail-safe-type equipment) will be promptly disconnected by alarms and operator action. The special features, however, do not meet the single failure criterion. Failure of the special features to alarm or failure of the operator to take prompt appropriate action are single failures which may cause a non-fail-safe type failure. The capability to scram the reactor could thus be compromised. This was DSER (SECY-91-355) Open Item 55.

Based on discussions, GE indicated that one EPA will be installed in each of the distribution circuits between the CVCF power supply and the RPS scram and main steam isolation valve (MSIV) solenoid valves (the fail-safe-type equipment). The CVCF abnormality in voltage or frequency alarm will be a Class 1E circuit and the CVCF alarm system and EPAs will be designed with the capability of being tested periodically.

The staff concludes that single failure of the EPA or the Class 1E CVCF power supply will not cause a non-fail-safe type failure of RPS scram or MSIV solenoid valves and is acceptable.

Verification that GE provided the above design commitments in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.3.3.14-1. GE included this information in Sections 8.3.1.1.4.2.1 and 8.3.1.1.4.2.2 of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

In addition, GE indicated that the CVCF alarm system and the EPA will be tested periodically. Verification that GE specified in a future SSAR amendment that the COL applicant will include these periodic tests in appropriate plant procedures was DFSER (SECY-92-349) COL Action Item 8.3.3.14-1. GE included this action item in Section 8.3.4.28 of SSAR Amendment 32, which is acceptable.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.3.3.14-1 was related to the design description and the ITAAC for the electrical protection for scram and MSIV solenoids. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

8.3.3.15 Safety Bus Grounding

On every bus shown in Figure 8.3-1 of the draft SSAR revision dated April 3, 1992, there is shown one circuit connected to ground through a circuit breaker. The circuit breaker or bus grounding device provides a safety ground on buses during maintenance operations. The bus grounding device includes the following interlocks

- under-voltage relays must be actuated
- related breakers must be in the disconnect position
- voltage for bus instrumentation must be available

The staff agreed that the proposed grounding device should be included in the design because it may be an important protection enhancement for personnel performing maintenance on Class 1E buses. The staff was concerned, however, that the proposed interlocks may not be sufficient to prevent inadvertent closing of the device during non-maintenance operation. This was DSER (SECY-91-355) Open Item 44.

GE indicated that annunciation will be provided in the design to alarm in the control room whenever the breakers are racked in for service. The staff concludes that a design which meets the above described commitments, together with administrative control and annunciation of the bus grounding system, will be sufficient to prevent inadvertent actuation of the grounding system. Consequently, this design is acceptable.

Verification that GE provided the above design commitments in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.3.3.15-1. GE included this information in Section 8.3.1.1.1 of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

In addition, GE indicated that there will be administrative controls to keep these circuit breakers racked out (that is in the disconnect position) whenever corresponding buses are energized. Verification that GE specified in a future SSAR amendment that the COL applicant will include these administrative controls in appropriate plant procedures was DFSER (SECY-92-349) COL Action Item 8.3.3.15-1. GE included this action item in Section 8.3.4.14 of SSAR Amendment 32, which is acceptable.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.3.3.15-1 related to the design description and the ITAAC for the safety bus grounding systems. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

8.3.3.16 Control of Access to Class 1E Power Equipment

By committing to meet the guidelines of IEEE 308-1980 (Table 1.8-21 of SSAR Amendment 17), GE indicated compliance with the requirements of Section 5.12 of IEEE 308-1980. Section 5.12 of IEEE 308-1980 requires that the plant design permit the administrative control of access to Class 1E power equipment. Based on this design commitment, information included in the draft SSAR revision dated April 3, 1992, and discussions, GE indicated that Class 1E power supplies and distribution equipment (including diesel generators, batteries, battery chargers, CVCF power supplies, 6.9 kV switchgear, 480-volt load centers, and 480-volt MCCs) will be located in areas with access doors that can be administratively controlled. In addition, ac and dc distribution panels will be located in the same or similar areas as Class 1E power supplies and distribution equipment; otherwise, the distribution panels will be designed to be locked so that access to circuit breakers located inside the panel can be administratively controlled. The plant physical design of the ABWR will permit the administrative control of access to Class 1E power equipment areas.

GDC 4 of 10 CFR Part 50, Appendix A, requires that Class 1E systems, equipment, and components be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents. The staff concludes that a design which meets the above described design commitments will permit control of the conditions that equipment may be subjected to during normal operation, maintenance, testing, and postulated accidents. The design, therefore, meets the above defined requirements of GDC 4 and is acceptable.

Verification that GE provided the above design commitments in a future SSAR amendment was DFSER (SECY-92-349) 8.3.3.16-1. GE has included these commitments in Section 8.3.3.6.1.1(5) of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

In addition, GE indicated that there will be administrative control of access to Class 1E power equipment areas and/or distribution panels. Verification that GE specified in a future SSAR amendment that the COL applicant will include these administrative controls in appropriate plant procedures was DFSER (SECY-92-349) COL Action Item 8.3.3.16-1. GE included this action item in Section 8.3.4.19 of SSAR Amendment 32, which is acceptable.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.3.3.16-1 related to the design description and the ITAAC for control of access for Class 1E power equipment. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

8.3.4 Electrical Independence

Based on discussions with GE and information included in the draft SSAR revision dated April 3, 1992, GE indicated that electrical independence is addressed in the design as follows:

- The protective actions (that is, the initiation of a signal with the sense and command features, or the operation of equipment within the execute features, for the purpose of accomplishing a safety function) of each load group will be independent of the protective actions provided by redundant load groups.

- Each onsite Class 1E power supply (for example, the diesel generator) will have provisions for automatic connection to one Class 1E load group, but will have no automatic connection to any other redundant Class 1E or non-Class 1E load group. If nonautomatic (manual) interconnecting means are furnished, provisions that prevent paralleling of the redundant onsite Class 1E power supplies will be included.

  - The ABWR electrical system design will not include provisions for the manual connection of the onsite Class 1E power supply of one Class 1E divisional load group to any other redundant Class 1E divisional or non-Class 1E non-divisional load group (except for the spare battery chargers)

    + The ABWR design will include provisions to allow one spare battery charger to be connected to either of two divisions and another spare battery charger to be connected to either of two other divisions.

    + The spare chargers for the dc power supply may be manually connected to either of two designated divisions, but only when their loads are switched to the same division. Key interlocks will mechanically ensure that these standby chargers can only be used in one division at a time.

  - The ABWR electrical system design will not have interconnections between redundant Class 1E divisions except as noted in Sections 8.2.2.3 and 8.3.4.1 of this report.

  - The divisional battery charger will normally be fed from its assigned Class 1E divisional 480-volt MCC bus.

- Each standby power system division includes the diesel generator, its auxiliary systems, and the distribution of power to various Class 1E loads through the 6.9 kV and 480-volt systems. Each of these divisions will be segregated and separated from the other divisions. No automatic interconnection will be provided between the Class 1E divisions. Each diesel generator set will operate independently of the other sets.

- Control power (for the Class 1E 480-volt auxiliaries) will be from the Class 1E 125-volt dc power system of the same division.

- Each Class 1E dc system load group will have its own battery charger with no provision for automatic interconnection with other redundant Class 1E load groups.

- There will be no provision for automatically interconnecting redundant dc system load groups.

- No provision will be made for automatically transferring loads between Class 1E dc power sources.

- The ABWR design will not have manual interconnections between redundant Class 1E divisions of the dc system except those that involve the battery chargers.

- Each Class 1E battery will be independent of other redundant battery supplies.

- Each Class 1E battery charger will be independent of other redundant battery chargers.

- The ac and dc switchgear power circuit breakers in each division will receive control power from their respective load groups to provide the following assurances

  - loss of one Class 1E 125-volt dc system division will not jeopardize the Class 1E power supply to the Class 1E buses of the other load groups

  - the differential relays in one division and all the interlocks affiliated with these relays will be from one 125-volt Class 1E dc system division. There will be no cross connections between the redundant dc system divisions through protective relaying

GDC 17 of 10 CFR Part 50, Appendix A, requires that the Class 1E systems, equipment, and components have sufficient independence and redundancy to perform their safety functions assuming a single failure. The staff concludes that a design which meets the above described commitments will include sufficient independent and redundant systems such that their will be reasonable assurance that a single failure of one set of systems (that is, one Class 1E division) will not prevent the remaining sets of interconnected system components, modules, and equipment from accomplishing the minimum required safety function. The design, therefore, meets the above defined requirements of GDC 17 of 10 CFR Part 50, Appendix A and is acceptable.

Verification that GE provided the above design commitments in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.3.4-1. In response to this item, GE further clarified their design to indicate that, (in addition to the manual transfer capability of the battery chargers discussed above) the ABWR electrical design will permit the manual transfer capability of the FMCRDs motors from a non-divisional to a divisional supply as well as the manual transfer capability of any one of the Class 1E diesel generators to any one of the Class 1E divisional or non-divisional load groups by backfeeding power from the diesel generator through the combustion turbine bus. For these design clarifications, the staff also concludes that a design which meets the above described commitments as clarified will include sufficient independent and redundant systems such that there will be reasonable assurance that a single failure of one set of systems (that is, one Class 1E division) will not prevent the remaining sets of interconnected system components, modules, and equipment from accomplishing the minimum required safety function. The design, therefore, meets the above defined requirements of GDC 17 of 10 CFR Part 50, Appendix A and is acceptable.

GE included the above design commitments in Sections 8.3.3.1, 8.3.1.1.8.1, 8.3.1.1.2.1, 8.3.1.2(4)(b), 8.3.2.1.3, 8.3.2.1.3.1, and 8.3.2.2.1 of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

In addition, GE indicated that the keys for the above described interlocks, manual interconnections, and manual transfer of power supplies will be administratively controlled by the COL applicant. Verification that GE specified in a future SSAR amendment a statement that the COL applicant will include these administrative controls in appropriate plant procedures was DFSER (SECY-92-349) COL Action Item 8.3.4-1. GE included this action item in Sections 8.3.4.15 and 8.3.4.18 of SSAR Amendment 32, which is acceptable.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.3.4-1 related to the design description and the ITAAC for electrical independence. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

8.3.4.1 Interconnections

Chapter 7 of this SER addresses design requirements for I&C system isolation devices. These devices are to be used to maintain independence between Class 1E and non-Class 1E circuits (such annunciators or data loggers and computer circuits) and between redundant Class 1E trip channels.)

Figure 8.3-8 of SSAR Amendment 10 showed two interconnections between redundant safety divisions:

(1) The Division III 480-volt bus is connected to the Division I 480-volt bus through circuit breakers and a mechanical interlock. Section 8.3.2.1 of SSAR Amendment 10 indicates that this interconnection is used to transfer the 250-volt dc normal battery charger between Division I and III load centers.

(2) The Division III 480-volt MCC is connected to Division I 480-volt MCC through battery chargers, breakers, and key interlocked breakers. Section 8.3.2.1 of SSAR Amendment 10 indicates that this interconnection is used for selection of the normal or standby battery chargers.

A staff concern over this issue was discussed in DSER (SECY-91-355) as Open Item 56.

In Section 8.3.2.1.4 and Figure 8.3-7 of the draft SSAR revision dated April 3, 1992, GE eliminated the interconnection between Divisions III and I (item 1 above), which was to be used to transfer the 250-volt dc battery charger between Class 1E divisions. In the new proposed design, power for the non-safety-related 250-volt dc battery charger is supplied from either the non-safety-related load group A or C turbine building load centers. With the elimination of the interconnection, this item was considered resolved.

Verification that GE provided the above design information in an SSAR amendment was one aspect of DFSER (SECY-92-349) Confirmatory Item 8.3.4.1-1. GE included this information in Section 8.3.2.1.3.3 of SSAR Amendment 32, which is acceptable.

In regard to the interconnection described in item 2 above, Section 8.3.2.1.2 and Figure 8.3-7 of the draft SSAR revision dated April 3, 1992, indicates that electrical interconnections will continue to exist between Divisions I and II and between Divisions I and III, so that two redundant divisions can share one standby charger. Similarly, Division I, and Division II, Division III, and Division IV, and Division I and Division III can be interconnected through the standby charger.

GDC 17 of 10 CFR Part 50, Appendix A requires that the Class 1E systems, equipment, and components have sufficient independence to perform their safety functions assuming a single failure. To meet this GDC 17 requirement with respect to electrical interconnection between redundant Class 1E divisions, it is the staff position that two independent open disconnect links, locked open breakers, or other equivalent open devices shall be maintained between the redundant Class 1E divisions. To meet this staff position, GE indicated that key interlocks will be installed as part of their electrical system design which will mechanically ensure that two open devices are always maintained between redundant divisions in accordance with the above staff position. (Also see Section 8.3.4 of this report.)

The staff concludes that the proposed key interlock design which meets the above described design commitments will maintain independence between them by using two open devices. Failure of one device will not challenge or cause failure of the remaining redundant divisions. Therefore, this design meets the above defined requirement of GDC 17 of 10 CFR Part 50, Appendix A, and is acceptable.

Verification that GE provided the above design commitments in a future SSAR amendment was the second aspect of DFSER (SECY-92-349) Confirmatory Item 8.3.4.1-1. GE has included these commitments in Section 8.3.2.1.3.1 of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

In addition, GE indicated that the keys for the interlock described above will be administratively controlled. Verification that GE specified in a future SSAR amendment that the COL applicant will include these administrative controls in appropriate plant procedures was DFSER (SECY-92-349) COL Action Item 8.3.4.1-1. GE included this action item in Section 8.3.4.18 of an SSAR markup dated March 31, 1993, which is acceptable.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.3.4.1-1 related to the design description and the ITAAC for interconnections between redundant divisions. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

8.3.4.2 Constant Voltage Constant Frequency Power Supplies

Section 8.3.1.1.4.2 of SSAR Amendment 10 indicated that each of the four independent trip systems of the reactor protection logic and control system is powered by four CVCF control power buses (one each for Divisions I, II, III, and IV). This section stated that each of these buses is independently supplied from an inverter which, in turn is supplied from one of four independent and redundant ac and dc power supplies. Subsequent sections and Figure 8.3-6 of SSAR Amendment 10, however, indicate that the ac supply for Divisions I and IV originates from a single 480-volt MCC (C14). A single 480-volt MCC is not "independent and redundant" as stated in Section 8.3.1.1.4.2. This was DSER (SECY-91-355) Open Item 57.

Based on discussions with and draft information provided by GE on September 4, 1991, November 26, 1991, and April 3, 1992, GE indicated that ac power to Divisions I and IV is supplied from the 6.9 kV Division I bus through a single 6.9 kV to 480-volt ac transformer and MCC to the vital ac system's CVCF power supplies and the dc system's battery charger power supplies for Divisions I and IV. GE indicated that Divisions I and IV ac and dc systems may be subject to a single common failure of the 6.9 kV to 480-volt transformer. In addition, GE indicated that they would revise the SSAR to indicate that there are four independent and redundant dc systems and three (versus four) independent and redundant ac systems.

Verification that GE provided the above design information in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.3.4.2-1. GE included this information in Section 8.3.1.1.4.2.1 of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

Subsequently, GE revised their design so that the ac supply for Division IV originates from a Division II (versus Division I) 480-volt MCC.

Because ac power to Divisions II and IV is supplied from a single 6.9 kV Division II bus through a single 6.9 to 480-volt ac transformer and MCC, Divisions II and IV ac and dc systems may be subject to common failure due to the single failure of the 6.9 kV to 480-volt transformer. The issue related to the lack of independence between Divisions II and IV of I&C equipment is resolved in Chapter 7 of this report.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.3.4.2-1 related to the design description and the ITAAC for constant voltage, constant frequency power supplies. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

8.3.4.3 Power Supply Circuits for Safety/Relief Valves

Section 19E.2.1.2.2.2 of SSAR Amendment 10 indicated that portions of each safety relief valve (SRV) control circuit use non-safety grade power. This power is taken from the Class 1E dc system through dc to dc converters or isolation devices connected to each of the four redundant and independent Class 1E dc system buses. Section 19E.2.1.2.2.2 implied that control power for each SRV comes from a minimum of two different Class 1E power source divisions. One source directly from the Class 1E dc bus, and the other from a different Class 1E dc bus through the dc-to-dc converter. The staff was concerned that the proposed design for powering the SRV may not provide sufficient independence between the redundant dc power sources, as required by GDC 17. This was DSER (SECY-91-355) Open Item 58.

Draft information provided by GE on September 4, 1991, modified SSAR Section 19E.2.1.2.2.2 to delete a reference to the use of non-safety grade power taken from safety grade batteries for a portion of each SRV control circuit. In addition, the information indicated that non-divisional power is not utilized in either the SRV or the automatic depressurization system (ADS) functions.

GE indicated that SRVs will be powered only from Class 1E sources and that there will be no electrical interconnection between power supplies. This design will ensure electrical independence between the redundant power supplies, and is acceptable.

Verification that GE provided the above design commitments in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.3.4.3-1. GE included this information in Section 19E.2.1.2.2.2 of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

8.3.4.4 Isolation Between Class 1E Buses and Loads Designated as non-Class 1E

Section 8.3.1.1.2.1 of SSAR Amendment 10 indicated that isolation breakers will be provided between the Class 1E and non-Class 1E buses. In addition zone-selective interlocking was provided between each isolation breaker and its upstream Class 1E bus feeder breaker. Section 8.3.1.2.1 of SSAR Amendment 10 indicated that even though the isolation breaker is only fault current actuated and does not meet the guidelines of Position 1 of RG 1.75 (Rev. 2), the zone selective interlocking technique met the intent of this RG. GE therefore concluded that this design met the recommendations of RG 1.75 (Rev. 2).

With respect to protecting Class 1E systems from failure of non-Class 1E systems and components, the staff agreed with GE that coordinated breakers with zone selective interlocking met the intent of Position 1 of RG 1.75 (Rev. 2), as well as the protection requirements of GDC 2 and 4. However, with respect to the independence requirement of GDC 17, the staff disagreed with GE's assessment (DSER (SECY-91-355) Open Item 48). Figure 8.3-5 of SSAR Amendment 10 showed non-safety-related computers and transient recorder loads with provisions included in their power supply design for automatically transferring these loads from Class 1E Divisions I to Division III and from Class 1E Divisions II to Division III. In addition, it appeared that the power supply design may have included provision for automatic transfer of loads between Divisions I and II. This design did not meet the guidelines of RG 1.6 or the intent of Position 1 of RG 1.75 (Rev. 2).

Subsequently, in Section 8.3.1.1.1 of the draft SSAR revision dated April 3, 1992, GE eliminated automatic transfer of loads between redundant divisions by indicating that only Class 1E Division I will have a non-safety-related load and this is acceptable. Verification that GE provided the above design information in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.3.4.4-1. GE included this information in Section 8.3.1.1.1 of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

By SSAR markup dated March 31, 1993, GE revised their design described in Section 8.3.1.1.1 of the draft SSAR revision, dated April 3, 1992, for supplying power to the Non-Class 1E FMCRD motors from the Class 1E system. The following evaluation addresses GE response to DFSER (SECY-92-349) Confirmatory Item 8.3.4.4-2.

GE indicated that the non-Class 1E loads, to be connected to the Class 1E power system through isolation devices, will consist of three separate groups of non-Class 1E FMCRD motors. The rod drive motors are considered non-Class 1E but important to safety because of their backup scram function. In addition, GE indicated that these non-Class 1E loads will be restricted to Division I and will be isolated from Class 1E systems by a Class 1E fault-actuated breaker, a zone-selective interlock, and a design restriction that the circuits on the load side of the transfer switch be classified non-Class 1E (so that these circuits will not be routed as associated circuits with cables of any Class 1E division). Upon loss of power from the Class 1E source, the non-Class 1E FMCRD loads will automatically be transferred between the Class 1E Division I power supply and a non-Class 1E power supply. The design will not allow automatic transfer of this non-Class 1E load back from the non-Class 1E to the Class 1E power supply. This non-Class 1E load can only be manually transferred from the non-Class 1E to the Class 1E power supply.

GE indicated in Section 8.3.1.1.1 of the SSAR markup dated March 31, 1993, the following

- the fault interrupt capability of breakers supplying Class 1E loads, including the Class 1E breakers supplying the Division I non-Class 1E FMCRD loads, will be coordinated with the fault interrupting capability of each load's upstream supply breaker so that failure of a greater part of the Class 1E division due to the single failure of a load will be minimized to the extent feasible

- the Class 1E load breakers for the non-Class 1E FMCRD loads will have zone selective interlocks with the Class 1E supply breaker to provide additional assurance that failure of a portion of a Class 1E division due to the failure of the non-Class 1E load will be minimized to the extent feasible

- the FMCRD motor circuits from the output of the 6.9 kV switchgear through the transfer switch will be classified as associated

- the FMCRD motor circuits from the associated transfer switch through the load and the feeder circuits from the non-Class 1E bus to the transfer switch will be classified non-Class 1E

- the fault interrupt capability of all Class 1E breakers, fault interrupt coordination between the supply and load breakers for each Class 1E load and each Division I non-Class 1E load, the zone selective interlock feature of the breaker for the non Class 1E loads, and the transfer of power from Class 1E to non-Class 1E sources will have the capability of being tested

- the Division I Class 1E onsite power supplies, the non-Class 1E offsite power supplies, and the Division I distribution system will have sufficient capacity and capability with margin to supply all Class 1E loads and the additional non-safety loads during all modes of plant operation

  - each FMCRD power train has current limiting features that are Class 1E to limit the FMCRD motor fault current

  - continuous operation of the FMCRD motors at the limiting fault current will not degrade operation of any Class 1E loads

  - the Division I diesel generator has sufficient capacity margin to supply overload currents up to the trip setpoint of the Class 1E feeder breaker to FMCRDs

In addition, by committing to meet the guidelines of IEEE 308-1980 (Table 1.8-21 of SSAR Amendment 17), GE indicated compliance with the requirements of Section 5.2 of IEEE 308-1980. Section 5.2 of IEEE 308-1980 requires that components, equipment, or systems utilized to provide isolation protection are covered by all the safety system design requirements defined in IEEE 603-1980. The staff concluded that the overcurrent protective devices and their coordination together with the zone select interlocks which are utilized to provide an isolation protection function to the Class 1E systems will meet the guidelines of Section 5.2 of IEEE 308-1980 and thus will meet safety system design requirements defined in IEEE 603-1980. By committing to meet the guidelines of IEEE 384-1981 (Table 1.8-21 of SSAR Amendment 33) and RG 1.75 (Rev. 2) (Table 1.8-20 of SSAR Amendment 33), GE indicated compliance with the guidelines of Sections 5.5.2(1) and 5.5.2(2) and Position 4 of RG 1.75, Revision 2. These guidelines require that associate circuits should be subject to all requirements placed on Class 1E circuits. The staff concluded that the non-Class 1E circuit that have been classified as associated will meet safety system design requirements defined in IEEE 603-1980. Also, by committing to the guidelines of IEEE 384-1981, GE indicated compliance with Section 7.1.2.1 of IEEE 384-1981. Section 7.1.2.1 requires that the circuit breaker being used for isolation shall be coordinated with upstream breakers, requires that the isolation breakers and their coordination be testable and be periodically tested, and requires that the Class 1E power supply to the loads being isolated have sufficient capacity and capability to supply fault current. The staff concluded that GE's proposed design for isolating non-Class 1E circuits includes design commitments which are in accordance with these guidelines of IEEE 384.

GDC 4 of 10 CFR Part 50, Appendix A, requires that Class 1E systems, equipment, and components be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents. GDC 17 of 10 CFR Part 50, Appendix A, requires (1) that offsite and onsite power systems have sufficient capacity and capability to permit safety systems to perform their required safety function and (2) that the Class 1E systems, equipment and components have sufficient independence to perform their safety functions assuming a single failure.

The staff concludes that an isolation system design which meets the above described commitments will adequately protect the Class 1E power system from the failure of the non-Class 1E rod control loads, provides sufficient capability and capacity from either the non-Class 1E offsite or the Class 1E onsite power systems to supply Class 1E loads in addition to the non-Class 1E rod control loads, and provides sufficient independence between redundant Class 1E divisions. The proposed ABWR design, for isolating the non-Class 1E FMCRD motor loads from the Class 1E power system, therefore, meets the above defined requirements of GDC 4 and 17 of 10 CFR Part 50, Appendix A and is acceptable. GE included the above design commitments in Section 8.3.1.1.1 of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

In addition, GE indicated that the design permits periodic calibration and testing of the fault interrupt capability of all Class 1E breakers, fault interrupt coordination between the supply and load breakers for each Class 1E load and the Division I non-Class 1E load, and the zone selective interlock feature of the breaker for the non-Class 1E load. Verification that GE specified in a future SSAR amendment that the COL applicant must include these periodic calibrations and functional tests in appropriate plant procedures was DFSER (SECY-92-349) COL Action Item 8.3.4.4-1. GE included this action item in Section 8.3.4.29 of SSAR Amendment 32, which is acceptable.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.3.4.4-1 related to the design description and the ITAAC for the isolation between Class 1E buses and non-Class 1E loads. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

8.3.5 Lighting Systems

In its DSER (SECY-91-355), the staff identified 17 concerns in Open Item 59. GE provided a response in the draft SSAR revision dated April 3, 1992. In that document GE indicated that there will be four lighting systems: the normal ac lighting system, the standby ac lighting system, the emergency dc lighting system, and the guide lamp lighting system.

The Normal Lighting System

The normal ac lighting system will be used to provide up to 50 percent of the lighting needed for operation, inspection, and repairs during normal plant operation. Normal lighting will be installed throughout the plant in areas containing non-safety-related equipment. Normal lighting will not be installed in passageways and stairwells. The normal lighting system will be part of the plant's non-safety-related system and as such will be supplied by the non-safety-related power system buses and will be energized as long as power from an offsite power source is available. Normal lighting will not be available following a LOPP event.

The ac Standby Lighting System

The ac standby lighting system will be comprised of two parts. The non-Class 1E ac standby system and the associated ac standby system. The Class 1E associated ac standby system will serve the safety-related equipment areas and their passageways. The non-Class 1E ac standby system will serve both safety and non-safety-related equipment areas and their passageways.

The associated ac standby system will be comprised of three subsystems. Each subsystem will be supplied from a different Class 1E standby divisional power supply (diesel generator). Each subsystem will supply a minimum of 50 percent of the lighting needs for the areas containing safety-related equipment in its respective division. Each subsystem will also supply 50 percent of the lighting in passageways and stairwells leading to its respective equipment areas. In addition, the subsystem associated with Division II will supply a minimum of 50 percent of the lighting needs for areas containing Division IV safety-related equipment (the Division IV battery room and other Division IV I&C areas). The subsystems associated with Divisions II and III will each supply a minimum of 50 percent of the lighting needs of the main control room. Each of the subsystem circuits will be treated as Class 1E (that is, they will be classified as associated) and as such will meet all requirements of a Class 1E circuit. In regard to the lighting system's fixtures and bulbs, GE has taken exception to the Class 1E requirement for seismic qualification. The staff's evaluation of this exception is addressed in Section 8.3.3.3 of this report.

The non-Class 1E ac standby system will be comprised of three non-Class 1E load groups. Each load group will be supplied from a different plant investment protection bus, which can be connected to the non-Class 1E standby power supply CTG. The non-Class 1E ac standby lighting system will supply a minimum of 50 percent of the lighting needed for its affiliated equipment areas and will supply 100 percent (50 percent from each of two different plant investment protection buses) of the lighting needs in passageways and stairwells leading to equipment areas containing non-safety-related equipment. In addition, the non-Class 1E ac standby lighting system will supply up to 50 percent of the lighting needs in areas containing safety-related equipment and in passageways and stairwells leading to them. The non-Class 1E lighting in the areas containing safety-related equipment and the passageways and stairwells leading to them will be supplied from the same unit auxiliary transformer power source as the safety-related equipment in the area. (Each unit auxiliary transformer can supply a Class 1E and non-Class 1E distribution system and load group. For example, the unit auxiliary transformer that supplies power to Division I of the Class 1E distribution system and load group will also supply a non-Class 1E distribution system and load group.) The non-Class 1E lighting system's fixtures in areas containing safety-related equipment and in the passageways and stairwells leading to them will be seismically supported and will be designed with appropriate grids or diffusers such that broken material from either the fixture or bulb will be contained and will not become a hazard to personnel or safety equipment during or following a seismic event.

The dc Emergency Lighting System

The dc emergency lighting system will provide dc powered backup lighting to prevent total blackout in areas which are or may be occupied during periods when ac lighting is lost until the normal or standby lighting systems are energized. The dc emergency lighting system will be comprised of two parts. The non-Class 1E dc emergency lighting system and the associated dc emergency lighting system. The non-Class 1E dc emergency lighting system supplies the lighting needed in plant areas containing non-safety-related equipment. The associated dc emergency lighting system supplies the lighting needed in plant areas containing safety-related equipment.

Plant areas containing safety-related equipment and associated dc emergency lighting include the main control room, the remote shutdown panel rooms, the diesel generator areas and associated control rooms, the safety-related electrical equipment rooms, and dc electric equipment rooms (including battery rooms). Electrical power for the associated dc emergency lighting system in these safety-related equipment areas (except the main control room) is supplied from the Class 1E 125-volt dc system in the same division as the equipment in the area (for example, Class 1E dc Division I will supply power to the associated dc emergency lighting system in those rooms containing Division I safety-related equipment). Electrical power for the associated dc emergency lighting system in the main control room will be supplied from the Divisions II and III Class 1E 125-volt dc systems. Each of the dc emergency lighting circuits will be treated as Class 1E (that is, they will be classified as associated) and as such will meet all requirements of a Class 1E circuit. In regard to the lighting system's fixtures and bulbs, GE has taken exception to the Class 1E requirement for the seismic qualification. The staff's evaluation of this exception is addressed in Section 8.3.3.3 of this report.

Plant areas containing non-safety-related equipment and non-Class 1E dc emergency lighting include the radwaste building control room, the CTG area and control room, and the non-safety-related electrical equipment areas (both ac and dc). Electrical power for the non-Class 1E dc emergency lighting system in the radwaste building control room will be supplied from the non-Class 1E 250-volt dc system. Electrical power for the non-Class 1E dc emergency lighting system in electrical equipment rooms containing non-safety-related equipment will be supplied from the same non-Class 1E 125-volt dc system that supplies power to equipment in the room. Electrical power for the non-Class 1E dc emergency lighting in the non-Class 1E CTG area and control room will be supplied from one of the three non-Class 1E 125-volt dc systems. The non Class 1E dc emergency lighting system circuits providing the needed lighting in non-safety-related equipment areas will be classified as non-Class 1E. (By letter dated March 31, 1993, GE modified their design to remove security lighting from their scope of design.)

The Guide Lamp Light System

The guide lamp light system will illuminate stairways, exit routes, and major control areas such as the main control room and remote shutdown panel areas. Each guide lamp unit will have a lighting fixture with two incandescent sealed-beam lamps and a self-contained battery pack unit containing a rechargeable battery with an 8-hour capacity. Each guide lamp unit will also contain a charger and an initiating switch, which energizes the fixture from the battery in the event of loss of the ac power supply and de-energizes the fixture upon return of ac power to the standby light following a time delay of 15 minutes. The guide lamp units will be supplied ac power from the same power source that supplies the associated standby lighting system in the area in which they are located. The guide lamp light system will be seismically qualified and will meet Class 1E requirements in plant areas containing Class 1E equipment.

GE provided information pertaining to the design of the normal, standby, emergency, and guide lamp lighting systems to demonstrate that the lighting is adequate in plant areas containing safety-related equipment as well as passageways to and from these areas. GE indicated that the ABWR design for each of these lighting systems will meet or exceed the lighting level requirements of the Illuminating Engineering Society Lighting Handbook and will have the capability of being functionally tested on a periodic basis.

The staff concluded that a lighting system that meets the above described commitments will provide adequate levels of light to permit the required operation and maintenance of equipment in safety-related equipment areas, and passageways to and from these areas, under normal operating conditions. This design is, therefore, acceptable. Verification that GE provided the above design commitments in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.3.5-1. GE included this information in Section 9.5.3 of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

In addition, GE indicated that the ac standby and dc emergency lighting systems which are normally energized will be periodically inspected and bulb replacement will be performed. In addition, the guide lamp system will be inspected and tested periodically to ensure operability of lights and switching circuits. Verification that GE has specified in a future SSAR amendment the requirement that the COL applicant must include these periodic tests, inspections, and bulb replacement in appropriate plant procedures was DFSER (SECY-92-349) COL Action Item 8.3.4.4-1. GE included this requirement in Section 8.3.4.25 of SSAR Amendment 32, which is acceptable.

For other off-normal conditions, it was not clear that the design adequately considers lighting needs for areas containing safety-related equipment and for passageways to and from these areas where plant operations are or may be required by emergency procedures. For example, under certain failures, it appears that the main control room may have only a portion (50 percent) of its lighting. Therefore, GE needed to further address the adequacy of the lighting in areas containing safety-related equipment under off-normal conditions. This was DFSER (SECY-92-349) Open Item 8.3.5-1.

GE indicated that because of the redundancy provided by the four lighting systems described above, the complete loss of lighting in any of the critical areas is not credible. The standby lighting and emergency lighting systems provide totally independent low level illumination in areas vital to safe shutdown of the reactor. In addition, the safety-related control systems will automatically bring the plant to safe-shutdown conditions. The control systems do not require the lighting system to perform their safety function of bringing the plant to a safe-shutdown condition.

The staff concluded that a lighting system design which meets the above described commitments will provide reasonable assurance that sufficient illumination will be available to perform emergency procedures as may be required during off normal conditions. The lighting system design is, therefore, acceptable. GE included the above commitments in Section 9.5.3.2.5 of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.3.5-2 related to the design description and the ITAAC for the lighting system. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

8.3.6 Design Control

8.3.6.1 Control of the Electrical Design Process

Recently, a number of problems have been identified with the electrical system design at nuclear power plants. Although the majority of these problems arose as a result of modifications performed after plant licensing, some were (and all could have been) the result of poor original design. GL 88-15 dated September 12, 1988, addressed a number of these problems that have occurred primarily as a result of inadequate control over the design process. These problems have occurred in areas of electrical system design which have historically well-established, comprehensive design criteria and guidelines available for the design engineer such as circuit breaker coordination and fault current interruption capability. The adequacy of the design is a function of the designers proper exercise of the well-established design criteria and guidelines.

In Sections 8.3.4.17 and 8.3.5 of draft SSAR revision dated April 3, 1992, GE indicated that purchase specification for both Class 1E and non-Class 1E equipment will contain a list of appropriate common industrial standards to ensure quality manufacturing. Based on this commitment, the staff concluded that this concern (DSER SECY-91-355 Open Item 60) is resolved. Verification that GE provided the above design information in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.3.6.1-1. GE included this information in Sections 8.3.4.17 and 8.3.5 of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.3.6.1-1 related to the design description and the ITAAC for the control of the electrical design process. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

8.3.6.2 Control of the Electrical Design Bases

The design bases described and presented in the ABWR SSAR should, for the most part, be useable as the bases by which the NRC issues a plant combined operating license. However, a review of the bases presented in Chapter 8 and other related chapters, revealed numerous inconsistencies as noted in DSER (SECY-91-355) Open Item 61. Consequently, it appears that GE used a deficient process to control the design bases presented in the ABWR SSAR.

At the time the DFSER was issued, GE had indicated that a formal engineering review and update of the SSAR was in progress to identify and correct inconsistencies within and between sections of the SSAR. Verification that GE revised the design bases in a future SSAR amendment to eliminate conflicting information was DFSER (SECY-92-349) Confirmatory Item 8.3.6.2-1. Staff review of SSAR Amendment 32 did not show these that inconsistencies had been resolved. Subsequently, the staff has reviewed the SSAR through Amendment 34 and found no significant inconsistencies which affect safety findings in this chapter. Therefore, Confirmatory Item 8.3.6.2-1 is resolved.

8.3.7 Testing and Surveillance

This section addresses DSER (SECY-91-355) Open Item 62. By committing to meet the guidelines of IEEE 308-1980 (Table 1.8-21 of SSAR Amendment 17), GE indicated compliance with the requirements of Section 7 of IEEE 308-1980. Section 7 of IEEE 308-1980 defines the guidelines for surveillance and preoperational and periodic equipment and system tests for electrical systems. Based on this commitment to Section 7 of IEEE 308-1980, information presented in the draft SSAR dated April 3, 1992, and discussions, GE indicated the following:

- The ABWR electrical system design will provide controls and indicators in the main control room.
- The design will include provisions for control and indication outside the main control room for
  - circuit breakers that switch Class 1E buses between the preferred and the standby power supply
  - the standby power supply
  - circuit breakers and other equipment as required for safety systems that must function to bring the plant to a safe-shutdown condition.
- Operational status information will be provided for Class 1E power systems.
- Class 1E power systems required to be controlled from outside the main control room will also have operational status information provided outside the central control room at the equipment itself, at its power supply, or at an alternate central location.
- The operator will be provided with accurate, complete, and timely information pertinent to the status of the execute features in the control room.
- Indication of protective actions and execute features unavailability, will be provided in the control room.
- Electric power systems and equipment will have the capability of being periodically tested.
- Testability of electrical systems and equipment will not be so burdensome operationally that required testing at intervals of 1, 2, or 3 months cannot be included in the TS if deemed necessary.

GDC 17 of 10 CFR Part 50, Appendix A, requires that Class 1E systems, equipment, and components have sufficient testability to perform their safety functions assuming a single failure. GDC 18 of 10 CFR Part 50, Appendix A requires that Class 1E power systems be designed (1) to permit appropriate periodic inspection and testing of important areas and features (such as wiring, insulation, connections, and switchboards) to assess the continuity of the systems and the condition of their components, (2) the capability to test periodically the operability and functional performance of the components of the systems, and (3) the capability to test periodically the operability of the systems as a whole and (under conditions as close to design as practical) the full operation sequence that brings the systems into operation.

Except as noted below, the staff concludes that an electrical system design which meets the above commitments will be testable. The design, therefore, meets the above defined requirements of GDC 17 and 18 of 10 CFR Part 50, Appendix A and is acceptable.

Verification that GE provided the above design commitments in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.3.7-1. GE has included these commitments in Section 8.1.3.1.1.3 of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

GE also indicated that the electrical systems and equipment will be periodically tested. Verification that GE specified in a future SSAR amendment that the COL applicant will include this periodic testing of electrical systems and equipment in appropriate plant procedures was DFSER (SECY-92-349) COL Action Item 8.3.7-1. GE included this action item in Section 8.3.4.30 of SSAR Amendment 32, which is acceptable.

Section 8.3.1.1.5.3, of SSAR Amendment 10 indicated that the ABWR design of Class 1E equipment permits periodic testing of the chain of system elements from sensing devices through driven equipment. This testing ensures that Class 1E equipment is functioning according to design requirements. This section also implies that the requirements of the single-failure criterion described in IEEE 379-1977, "Standard Application of the Single Failure Criterion to Nuclear Power Generating Station Safety Systems," will be met with respect to testing of Class 1E equipment.

The staff interpreted these statements to mean that one complete electrical system division may be taken out of service for maintenance, testing, and/or repair during any mode of plant operation and still leave the remaining electrical systems in compliance with the single-failure criterion. The staff, therefore, concluded that the design provision for testability of electrical systems (as interpreted) meets the sufficient testability requirement of GDC 17 and is acceptable.

To confirm this interpretation, the staff further evaluated the capability of the electric power system to be tested during normal plant operation while meeting single failure requirements with remaining systems for any design-basis event.

GE's design commitment that the ABWR design meets Section 5 of IEEE 338-1977, which is endorsed by RG 1.118 (Rev. 2), reflected the following:

- On-line testing will be greatly enhanced by the design which utilizes three independent divisions, any one of which can safely shutdown the plant. However, the design will not meet the single-failure criterion with respect to all required safety-related systems for all design-basis events with one of the three electrical system divisions out of service.

- An acceptable level of reliability for the remaining operable safety systems will exist when one train is taken out of service for a specified period of time for preplanned or unplanned maintenance while the single-feature criterion is not met.

- When one Class 1E division is out of service, an acceptable level of reliability will be established by GE for the remaining operable safety systems by a PRA.

Based on the information presented in the ABWR SSAR, the staff was unable to determine what constitutes an acceptable level of reliability. This was DFSER (SECY-92-349) Open Item 8.3.7-1.

GE revised Sections 8.2.2.1(3) and 8.3.1.1.5.3 of SSAR Amendment 21 to indicate that all equipment can be tested, as necessary, to assure continued and safe operation of the plant. For equipment which will not be tested during operation, GE indicated that the equipment's reliability will be such that testing can be performed during plant shutdown. Based on the above commitment as revised, the staff concludes that all systems and component parts of an electrical system design which met the above commitments as revised will have the capability of being tested. In addition, the reliability assurance program as discussed in Chapter 17.3 of this report will ensure that important reliability assumptions of the probabilistic risk assessment will be considered throughout the plant life. The design therefore meets the above defined requirements of GDC 17 and 18 of 10 CFR Part 50, Appendix A, and is acceptable. GE included the above design commitments in Sections 8.2.2.1(3) and 8.3.3.2 of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

<u>Certified Design Material</u>

DFSER (SECY-92-349) Open Item 8.3.7-2 related to the design description and the ITAAC for the testing and surveillance of electrical equipment. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

8.3.8 Capacity and Capability

8.3.8.1 Non-Class 1E dc Power Systems

In response to DSER (SECY-91-355) Open Item 64, regarding interaction between safety and non-safety-related systems, GE provided in Sections 8.3.2.1.3 and 8.3.2.1.4 of the draft SSAR revision dated April 3, 1992, a description of the non-Class 1E 125-and 250-volt dc systems. The 125-volt dc non-Class 1E system will provide power to non-Class 1E switchgear, valves, converters, transducers, controls, and instrumentation. The non-Class 1E 125-volt dc system will have three load groups with one battery, charger, and bus per load group. The 250-volt dc non-Class 1E system will provide power for non-safety-related computers and the turbine turning gear motor. The 125- and 250-volt dc systems will provide power only to non-safety-related loads and will be physically and electrically independent of the Class 1E ac and dc systems.

Based on the design commitments included in the draft SSAR revision, staff concerns relating to interactions between safety and non-safety-related systems associated with the previously proposed common dc power supply have been resolved.

Verification that GE provided the above design commitments in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.3.8.1-1. GE included this information in Sections 8.3.2.1.3.2 and 8.3.2.1.3.3 of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.3.8.1-1 related to the design description and the ITAAC for the independence between Class 1E and non-Class 1E dc power systems. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

8.3.8.2 Capacity of the Class 1E 125-volt dc Battery Supply

This section addresses the staff's evaluation of GE's response to DSER (SECY-91-355) Open Item 65.

Based on information presented in Section 8.3.2.1.1.1 of the draft SSAR revision dated April 3, 1992, GE indicated that each of the four Class 1E 125-volt batteries will

- be capable of starting and operating its required steady state and transient loads
- be immediately available during both normal operations and following the loss of power from the alternating current system
- have sufficient stored energy to provide an adequate source of power for starting and operating all required LOCA and/or LOPP loads and circuit breakers for two hours with no ac power
- have sufficient stored energy to provide power in excess of the capacity of the battery charger when needed to supply transient loads
- be sized in accordance with industry recommended practice defined in IEEE 485-1983
- have a capacity design margin of 5 to 15 percent to allow for less than optimum operating conditions
- have a 25-percent capacity design margin to compensate for battery aging
- have a 4-percent capacity design margin to allow for the lowest expected electrolyte temperature of 21 °C (70 °F)
- have a number of battery cells that matches the battery-to-system voltage limitations
- base the first minute of the batteries' duty cycle on the sum of all momentary, continuous, and noncontinuous loads that can be expected to operate during the one minute following a LOCA and/or LOPP
- be installed in accordance with IEEE 484-1987
- meet the recommendations of Section 5 of IEEE 946-1985
- be designed so that each battery's capacity can periodically be verified

In addition to having sufficient stored energy to operate all required LOCA and/or LOPP loads and circuit breakers for 2 hours, GE indicated that the Division I battery will have sufficient stored energy to provide an adequate source of power to start and operate all required loads and circuit breakers for approximately 8 hours with no ac power. Further, the heating/ventilation system will maintain battery electrolyte temperature above 21 °C (70 °F).

GDC 17 of 10 CFR Part 50, Appendix A, requires that the Class 1E 125-volt dc battery supply have sufficient (1) capacity and capability to permit safety systems to perform their required safety function and (2) testability to perform their safety function assuming a single failure. GDC 18 of 10 CFR Part 50, Appendix A requires that the Class 1E 125-volt dc battery supply systems be designed (1) to permit appropriate periodic inspection and testing of important areas and features (such as wiring, insulation, connections, and switchboards) to assess the continuity of the systems and the condition of their components, (2) the capability to test periodically the operability and functional performance of the components of the systems, and (3) the capability to test periodically the operability of the systems as a whole and (under conditions as close to design as practical) the full operation sequence that brings the systems into operation. The staff concludes that Class 1E

125-volt battery supply design that meets the above commitments will have sufficient capacity and capability to supply required loads following a LOCA and/or LOPP and a station blackout event and will be testable. The design, therefore, meets the above defined requirements of GDC 17 and 18 of 10 CFR Part 50, Appendix A, and is acceptable.

Verification that GE provided the above design commitments in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.3.8.2-1. In response to this item, GE modified the above commitments to indicate that the batteries' capacity design margin to allow for the lowest expected electrolyte temperature was changed from 21 °C (70 °F) to 10 °C (50 °F). This change was based on the capability of the heating and ventilation system to maintain room temperature above 10 °C (50 °F). The capability of the heating and ventilation system to maintain a temperature range of 10 to 40 °C (50 °F to 104 °F) is addressed in Chapter 9.0 of this SER.

The staff concludes that a Class 1E 125-volt battery supply design that meets the above commitments as modified, will also have sufficient capacity and capability to supply required loads following a LOCA and/or LOPP and a station blackout event and will be testable. The design, therefore, meets the above defined requirements of GDC 17 and 18 of 10 CFR Part 50, Appendix A, and is acceptable. GE has included the above design commitments in Sections 8.3.2.1.1.1, 8.3.2.1.3.1, and 8.3.2.2.2(5)(c) of SSAR Amendment 32, which is acceptable.

In addition, GE indicated that the capacity and capability of the dc system batteries and the capability of the batteries to supply power to their connected loads will periodically be tested in accordance with the recommendation of IEEE 450-1985. Verification that GE specified in a future SSAR amendment that the COL applicant will include these periodic tests in appropriate plant procedures was DFSER (SECY-92-349) COL Action Item 8.3.8.2-1. GE included this action item in Sections 8.3.4.32 and 8.3.4.33 of SSAR Amendment 32, which is acceptable.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.3.8.2-1 related to the design description and the ITAAC for the Class 1E 125-volt dc battery supply. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

8.3.8.3 Use of Silicon Diode in the dc System

Figure 8.3-7 and GE's response to Question 435.51 in SSAR Amendment 10 indicated that a silicon diode would be installed in series with the output of the battery and battery chargers to create a voltage drop of 10 volts. During normal operation (that is, when battery charger output voltage is set at 140 volts to equalize the charge of battery cells) the switch in parallel with the silicon diode will be open so that the voltage from the battery charger to the dc bus will remain at 130 volts (140 volts minus the 10-volt drop across the silicon diode), while 140 volts is supplied to the battery to equalize the charge of battery cells.

In response to DSER (SECY-91-355) Open Item 66, GE provided the draft SSAR revision dated April 3, 1992, which removed the use of the silicon diode from the ABWR design and restated the commitment to design the dc system distribution equipment, component, and loads to function at 140 volts during equalization charge.

This item is, therefore, considered resolved. Verification that GE revised the SSAR in a future amendment, indicating the removal of the silicon diode from the ABWR design was DFSER (SECY-92-349) Confirmatory Item 8.3.8.3-1. GE included this revision in SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

8.3.8.4 Class 1E ac Standby Power System (Diesel Generator)

This section addresses the staff's evaluation of GE's response to DSER (SECY-91-355) Open Item 67.

Based on discussions with GE and information presented in the draft SSAR revision dated April 3, 1992. GE indicated that each standby (diesel generator) power source will

- be capable of starting, accelerating to rated speed, and supplying, in the required sequence, all the required safety system loads
- be capable of attaining rated frequency (i.e., full speed) and voltage within 20 seconds after receipt of a start signal
- have a continuous load rating of 6.25 megavolt amps (MVA) @ 0.8 power factor

- have a short time rating (operation at this rating does not limit the use of the diesel generator unit at its continuous rating) of 110 percent of the continuous load rating for a 2-hour period out of any 24-hour period, without exceeding the manufacturer's design limits and without reducing the maintenance interval established for the continuous rating
- be available following the loss of the preferred power supply within a time consistent with the requirements of the safety function under normal and accident conditions
- have stored energy (fuel) at the site in its own storage tank with the capacity to operate the standby diesel generator power supply, while supplying post-accident power requirements to a unit for seven days
- have stored energy (fuel) at the site in its own day tank with the capacity to operate the standby diesel generator power supply while supplying post-accident power requirements for 8 hours
- have a fuel transfer system with the capability of automatically replenishing the day tank from the storage tank such that the 8-hour fuel capacity of the day tank is maintained
- be capable of operating in its service environment during and after any design-basis event, without support from the preferred power supply
- be capable of starting, accelerating, and being loaded with the design load, within an acceptable time
  - from the diesel engine's normal standby condition
  - with no cooling available, for a time equivalent to that required to bring the cooling equipment into service with energy from the diesel generator unit
  - on a restart with an initial engine temperature equal to the continuous rating full load engine temperature
- be capable of accepting design load following operation at light or no load for a period of 4 hours
- be capable of maintaining voltage and frequency at the generator terminals within limits that will not degrade the performance of any of the loads comprising the design load below their minimum requirements, including the duration of transients caused by load application or load removal
- be capable of carrying its continuous load rating for 22 hours following 2 hours of operation at its short time rating
- start from each automatic and remote manual signal and then accelerate to rated voltage and frequency, and then properly sequence its loads if there is no offsite power available or operate at no load if offsite power is available
- start but not sequence its loads by a local manual start signal
- be capable of being manually started without ac external electric power
- be capable of automatic acceleration to rated voltage and frequency without ac external electric power
- be capable of allowing the bus to be manually energized without ac external electric power

GE also indicated that

- the maximum loads expected to occur for each division (according to nameplate ratings) will not exceed 90 percent of the continuous power output rating of the diesel generator
- each diesel generator's air receiver tanks will have sufficient capacity for five starts without recharging
- following one unsuccessful automatic start of the diesel generator with and without ac external power, each diesel generator's air receiver tanks will have sufficient air remaining for three more successful starts without recharging
- automatic load sequence will begin at $\leq$ 20 seconds and will end at $\leq$ 65 seconds
- following application of each load during load sequencing, voltage will not drop more than 25 percent from nominal voltage measured at the bus
- following application of each load during load sequencing frequency will not drop more than 5 percent from nominal frequency measured at the bus
- frequency will be restored to within 2 percent of nominal, and voltage will be restored to within 10 percent of nominal within 60 percent of each load sequence time interval

- during recovery from transients caused by step load increases or resulting from the disconnection of the largest single load, the speed of the diesel generator unit will not exceed the nominal speed plus 75 percent of the difference between nominal speed and the overspeed trip setpoint or 115 percent of nominal, whichever is lower

- the transient following the complete loss of load will not cause the speed of the unit to attain the overspeed trip setpoint.

- bus voltage and frequency will recover to 6.9 kV$\pm$10-percent at 60$\pm$2-percent Hz within 10 seconds following trip and restart of the largest load

- each of the above design commitments will have the capability of being periodically verified

GDC 17 of 10 CFR Part 50, Appendix A, requires that the Class 1E ac standby power supply (the diesel generator power supply) have sufficient (1) capacity and capability to permit safety systems to perform their required safety function and (2) testability to perform their safety functions assuming a single failure. GDC 18 of 10 CFR Part 50, Appendix A, requires that the Class 1E ac standby electric power system be designed (1) to permit appropriate periodic inspection and testing of important areas and features (such as wiring, insulation, connections, and switchboards) to assess the continuity of the systems and the condition of their components, (2) the capability to test periodically the operability and functional performance of the components of the systems, and (3) the capability to test periodically the operability of the systems as a whole and (under conditions as close to design as practical) the full operation sequence that brings the systems into operation. The staff concludes that a diesel generator power supply design that meets the above described commitments will have sufficient capacity and capability to supply required loads and will be testable. The design, therefore, meets the above defined requirements of GDC 17 and 18 of 10 CFR Part 50, Appendix A, and is acceptable. Verification that GE has revised the SSAR to include the above design commitments in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.3.8.4-1. GE has included these commitments in Sections 8.1.3.1.1.3, 8.3.1.1.8.2, 8.3.1.1.8.3, and 8.3.1.1.8.6 of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

GE indicated that each of the above design commitments will be verified periodically and that testing or analysis will be performed periodically to demonstrate the capability of the diesel generator to supply the actual full design basis load current for each sequenced load step. Verification that GE specified in a future SSAR amendment that the COL applicant will include these periodic tests and analysis in appropriate plant procedures was DFSER (SECY-92-349) COL Action Item 8.3.8.4-1. GE included this action item in 8.3.4.36 of SSAR Amendment 32, which is acceptable.

Start time for the diesel generators

As part of DSER (SECY-91-355) Open Item 67, the staff identified inconsistencies between Sections 8.3.1.1.8.2 and 8.3.1.1.8.3 of SSAR Amendment 10 with regard to the design capability of the diesel generator to start and attain rated voltage and frequency. Section 8.3.1.1.8.2 of SSAR Amendment 10 indicated a 13-second design capability, while Section 8.3.1.1.8.3 indicated a 20-second capability.

GE's SSAR Amendment 17 corrected the inconsistency by changing the 13-seconds start time for the diesel generator to 20 seconds, with the sequence start times for loads changing accordingly. GE indicated that this change was made to be consistent with EPRI/advanced light water reactor requirements.

GE indicated that the accident analysis requires the residual heat removal (RHR) and HPCF injection valves to be open 36 seconds after the receipt of a high drywell or low reactor vessel level signal. Since the motor operated valves are not tripped off the buses, they start to open (if requested to do so by their controls) when power is restored to the bus at 20 seconds. This gives them an allowable travel time of 16 seconds.

Changing the allowable start time for the diesel generator from 13 to 20 seconds has changed the allowable travel time for the RHR and HPCF injection valves to move from the close to open position from 23 to 16 seconds. GE indicated that this reduction to a 16 second travel or opening time is attainable for the RHR and HPCF injection valves. The staff therefore concluded that the longer start time for the diesel generator is within the accident analysis limits and is acceptable. GE included this information in Section 8.3.1.1.8.2 of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

Load Sequencing on the Diesel Generator following a LOCA with a delayed LOPP

This section addresses, in part, the staff's evaluation of GE's response to DSER (SECY-91-355) Open Item 28 and DFSER (SECY-92-349) Confirmatory Item 8.2.3.2-1.

As noted in Section 8.2.3.2 of this report, the diesel generator may be overloaded if there is a LOCA with a delayed LOPP. Section 8.3.1.1.7(3) of SSAR

Amendment 21, LOPP following LOCA, indicated that all loads that have been connected to the Class 1E bus and are operating from the offsite power supply in response to the LOCA will be connected, in one block load, to the standby diesel generator (operating at no load due to the LOCA) if there is a LOPP. The staff was concerned that the diesel generator may not have sufficient capacity to supply, in one block, the loads which may be operating on the Class 1E bus following a LOCA with a delayed LOPP.

Subsequently, GE indicated that the diesel generator breaker will not close until after large motors loads are tripped. Thus, loads will be reloaded onto the bus in a controlled sequence which will not overload the diesel generator power supplies.

If there is a LOCA with a delayed LOPP, if there is a coincident LOCA and LOPP, or if there is a LOPP with a subsequent LOCA, the staff concludes that a diesel generator power supply design that meets the above commitments (as clarified) will have sufficient capacity and capability to supply required loads for all modes of plant operation and will be testable. The design, therefore, meets the above defined requirements of GDC 17 and 18 of 10 CFR Part 50, Appendix A, and is acceptable. GE has included the above clarification in Section 8.3.1.1.7 of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.3.8.4-1 was related to the design description and the ITAAC for the capacity and capability of the Class 1E ac standby power system. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

8.3.8.5 Constant Voltage Constant Frequency Power Supply Capacity

Based on information presented in the draft SSAR revision dated April 3, 1992, and discussions; GE indicated that each of the four redundant Class 1E CVCF power supplies will have a capacity based on the largest combined demands of the various continuous loads, plus the largest combination of noncontinuous loads that would likely be connected to the power supply simultaneously during normal or accident plant operation, whichever is higher. The design will also permit periodic verification of this capacity for each of the CVCF power supplies.

GDC 17 of 10 CFR Part 50, Appendix A, requires that the Class 1E CVCF power supplies have sufficient (1) capacity and capability to permit safety systems to perform their required safety function and (2) testability to perform their safety functions assuming a single failure. GDC 18 of 10 CFR Part 50, Appendix A, requires that the Class 1E CVCF power supply systems be designed (1) to permit appropriate periodic inspection and testing of important areas and features (such as wiring, insulation, connections, and switchboards) to assess the continuity of the systems and the condition of their components, (2) the capability to test periodically the operability and functional performance of the components of the systems, and (3) the capability to test periodically the operability of the systems as a whole and (under conditions as close to design as practical) the full operation sequence that brings the systems into operation. The staff concludes that a CVCF power supply design that meets the above commitments will have sufficient capacity and capability to supply required loads and will be testable. The design, therefore, meets the above defined requirements of GDC 17 and 18 of 10 CFR Part 50, Appendix A, and is acceptable.

Verification that GE provided the above design commitments in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.3.8.5-1. GE has included these commitments in Section 8.3.1.1.4.2.1 of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

GE also indicated that the above capacity commitment for the CVCF power supplies will periodically be verified. Verification that GE specified in a future SSAR amendment that the COL applicant will include these periodic tests in appropriate plant procedures was DFSER (SECY-92-349) COL Action Item 8.3.8.5-1. GE included this action item in Section 8.3.4.34 of SSAR Amendment 32, which is acceptable.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.3.8.5-1 related to the design description and the ITAAC for the capacity of the CVCF power supplies. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

8.3.8.6 Battery Charger

Based on information presented in the draft SSAR revision dated April 3, 1992, and discussions, GE indicated that each of the four redundant Class 1E dc battery chargers will have sufficient capability and will be testable. The battery chargers will have a capacity based on the largest combined demands of the various continuous steady-state loads, plus charging capacity to restore the battery from the design minimum charge state to the fully charged state within the time stated in the design-basis, regardless of the status of the plant during which these demands occur. The dc battery chargers will have a disconnecting device in the incoming ac power feed and in its dc-power output circuit for isolating the charger. The dc battery chargers will be designed to prevent the ac power supply from becoming a load on the battery. The dc battery chargers will also have provisions to isolate transients from the ac system from affecting the dc system and; conversely, provisions will be included to isolate transients from the dc system from affecting the ac system. The dc battery charger system will be sized in accordance with the guidelines defined in Section 6.1 of IEEE 946-1985 for establishing the required rating for battery chargers. The design of the dc system will include the capability to periodically verify the required capacity for each of the battery charger power supplies.

GDC 17 of 10 CFR Part 50, Appendix A, requires that the Class 1E battery chargers have sufficient (1) capacity and capability to permit safety systems to perform their required safety function and (2) testability to perform their safety functions assuming a single failure. GDC 18 of 10 CFR Part 50, Appendix A, requires that the Class 1E dc system battery chargers be designed (1) to permit appropriate periodic inspection and testing of important areas and features (such as wiring, insulation, connections, and switchboards) to assess the continuity of the systems and the condition of their components, (2) the capability to test periodically the operability and functional performance of the components of the systems, and (3) the capability to test periodically the operability of the systems as a whole and (under conditions as close to design as practical) the full operation sequence that brings the systems into operation. The staff concludes that a dc system battery charger design that meets the above described commitments will have sufficient capacity and capability to supply required loads and will be testable. The design, therefore, meets the above defined requirements of GDC 17 and 18 of 10 CFR Part 50, Appendix A, and is acceptable.

Verification that GE provided the above design commitments in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.3.8.6-1. GE has included these commitments on Figure 8.3-4 and in Sections 8.3.2.1.1 of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

GE also indicated that the above capacity commitment for the battery charger power supplies will be periodically verified. Verification that GE specified in a future SSAR amendment that the COL applicant will include this periodic testing of the battery capacity in appropriate plant procedures was DFSER (SECY-92-349) COL Action Item 8.3.8.6-1. GE included this action item in Section 8.3.4.35 SSAR Amendment 32, which is acceptable.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.3.8.6-1 related to the design description and the ITAAC for the capacity of the battery charger. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

8.3.8.7 Distribution Systems

Based on information presented in the draft SSAR revision dated April 3, 1992, and discussions, GE indicated that each Class 1E distribution circuit will be capable of transmitting sufficient energy to start and operate all required loads in that circuit for all plant conditions described in the design basis. GE also indicated that the design will also permit periodic verification of this required capacity for each distribution circuit.

GDC 17 of 10 CFR Part 50, Appendix A, requires that the Class 1E distribution system have sufficient (1) capacity and capability to permit safety systems to perform their required safety function and (2) testability to perform their safety functions assuming a single failure. GDC 18 of 10 CFR Part 50, Appendix A, requires that the Class 1E distribution systems be designed (1) to permit appropriate periodic inspection and testing of important areas and features (such as wiring, insulation, connections, and switchboards) to assess the continuity of the systems and the condition of their components, (2) the capability to test periodically the operability and functional performance of the components of the systems, and (3) the capability to test periodically the operability of the systems as a whole and (under conditions as close to design as practical) the full operation sequence that brings the systems into operation.

The staff concludes that a Class 1E distribution system design that meets the above described commitments will be capable of supplying sufficient energy to Class 1E safety system loads for their operation, and will be capable of being tested. The design, therefore, meets the above defined requirements of GDC 17 and 18 of 10 CFR Part 50, Appendix A, and is acceptable.

Verification that GE provided the above design commitments in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.3.8.7-1. GE has included these commitments in Sections 8.3.1.1.5(2) and

8.3.3.2 of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

GE also indicated that the above capacity commitment for each distribution circuit will periodically be verified. Verification that GE has specified in a future SSAR amendment that the COL applicant will include this periodic testing in appropriate plant procedures was DFSER (SECY-92-349) COL Action Item 8.3.8.7-1. GE included this action item in Section 8.3.4.30 of SSAR Amendment 32, which is acceptable.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.3.8.7-1 related to the design description and the ITAAC for capacity of distributions systems. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

8.3.9 Station Blackout (SBO)

In SECY-90-016, dated January 12, 1990, the staff recommended that the Commission approve its position that a diverse, AAC source, be provided for SBO in evolutionary LWRs. In its June 26, 1990, SRM, the Commission approved the staff's position.

In response to the above staff position and DFSER (SECY-92-349) Confirmatory Items 8.3.9.1-1, 8.3.9.2-1, and 8.3.9.3-1 and open items 8.3.9.1-1, 8.3.9.2-1, and 8.3.9.3-1, GE indicated that they planned to change their commitment for meeting the SBO rule from a design that met the SBO rule through the use of coping to a design that meets the SBO rule through the use of an alternate ac (AAC) power source in accordance with NRC Policy Issue SECY-90-016.

In an SSAR draft version dated July 2, 1993, GE indicated in accordance with RG 1.115, that the AAC power source will

- be a CTG
- be capable of automatically starting, accelerating to rated speed, of reaching nominal voltage, and to begin accepting load within two minutes of receipt of its start signal
- be capable of being manually connected to Class 1E 6.9 kV buses and safety-related loads
- be capable of being manually reconfigured such that non safety investment protection loads can be shed and safety-related shutdown loads can be connected (via any one of the Class 1E 6.9 kV buses) from the main control room within ten minutes
- be a self-contained unit equipped with its own auxiliary control and support systems such that external ac power is not required for its operation
- be physically and electrically independent and diverse from the Class 1E standby diesel generators such that weather-related failures, common cause failures, or single point vulnerabilities are minimized to the extent practicable
- be electrically isolated from each 6.9 kV bus by two normally open circuit breakers in series (one Class 1E and one non-Class 1E)
- require DC control power from the Class 1E and non-Class 1E dc power systems so that the AAC power source can be connected to the Class 1E 6.9 kV bus (to close the two normally open circuit breakers) from the control room within the 10 minute time requirement
- be capable of operating during and after a station blackout without any ac support systems powered from the preferred power supply or the blacked-out units Class 1E power sources affected by the event
- have sufficient capacity and capability to power one safety-related bus within 10 minutes of the onset of a station blackout, such that the plant safety systems will be capable of maintaining core cooling and containment integrity
- not normally or automatically be connected to the off-site power sources or the on-site Class 1E 6.9 kV buses thus minimizing the possibility of a common cause failure
- be designed to assume non-safety plant investment protection (PIP) loads automatically, to shed loads manually, and assume safety system loads manually while maintaining voltage and frequency within design requirements of safety system loads
- be capable of powering required shutdown loads with margin
- be capable of powering HVAC systems, chillers, battery chargers, and other support/auxiliary equipment during the station blackout event such that the environment during and following a station blackout event will not exceed the environment for which the

equipment is designed and the habitable environment for personnel

- undergo factory testing, similar to the Class 1E standby diesel generator, to demonstrate its ability to reliably start, accelerate to rated speed, voltage, and supply power within two minutes

- be subject to site acceptance testing, periodic preventive maintenance, inspection, testing and operational reliability assurance program goals

- be designed to quality assurance requirements commensurate with its importance to safety

- be located above the maximum flood level in the turbine building

- be provided with an oil storage and transfer system that will be physically and mechanically independent of the Class 1E standby diesel generator oil storage and transfer system

- have its fuel sampled and analyzed consistent with applicable standards

- have sufficient fuel oil stored on-site to support 7 days operation

- be capable of being periodically inspected, tested, and maintained

In addition, GE indicated that required plant core cooling and containment integrity during the station blackout duration (10 minutes) will not depend on any ac power sources.

Section 50.63(c)(2) of 10 CFR Part 50 requires (1) the time required for startup and alignment of the AAC to required shutdown equipment be demonstrated by test, (2) the ACC source have sufficient capacity and capability to permit required shutdown equipment to perform their required function, and (3) the AAC can be started and aligned within 10 minutes to the buses that distribute power to required shutdown equipment.

The staff concludes that a design, which meets the above described commitments, will have sufficient capacity, capability, and testability to provide power to required shutdown loads within specified station blackout time limitations. The design, therefore, meets the above defined requirements of Section 50.63(c)(2) of 10 CFR Part 50 and is acceptable. GE has included the above design commitments in Section 8.3.1.1.7(9) of SSAR Amendment 32, which is acceptable.

GE also indicated that the COL applicant must include periodic testing and/or analysis to verify the adequacy of the CTG to meet AAC requirements for station blackout in appropriate plant operating procedures. GE included this requirement in Section 9.5.13.19 of SSAR Amendment 32, which is acceptable.

Certified Design Material

The adequacy and acceptability of the ABWR design descriptions and ITAAC for the AAC are evaluated in Section 14.3 of this report.

9 AUXILIARY SYSTEMS

9.1 Fuel Storage and Handling

9.1.1 New Fuel Storage

The staff reviewed the new fuel storage capability in accordance with Standard Review Plan (SRP) Section 9.1.1. Staff acceptance of the new fuel storage facility design is based on the design meeting the requirements of General Design Criteria (GDC) 2 as it relates to the ability of structures housing the facility and the facility itself to withstand the effects of natural phenomena such as earthquakes; the position in Regulatory Guide (RG) 1.29, "Seismic Design Classification," Revision 3, Section C, which lists the systems which need to be designed to operate following a safe shutdown earthquake (SSE); GDC 5 as it relates to shared structures, systems, and components (SSCs) important to safety being capable of performing required safety functions; GDC 61 as it relates to the facility design for fuel storage; and GDC 62 as it relates to the prevention of criticality.

New fuel storage is provided in the new fuel storage vault located in the reactor building. The vault will contain storage racks for up to 40 percent of one full core fuel load. New fuel normally will be in dry storage; however, the storage racks can be used in either a wet or dry mode. The storage racks, vault, and the reactor building that houses the facilities are designed to seismic Category I criteria. The reactor building also is designed to provide protection against flooding and tornado missiles as evaluated in Sections 3.4.1 and 3.5.2 of this report, respectively. Therefore, the design satisfies the requirements of GDC 2 and the guidelines of Position C.1 of RG 1.29, Revision 3 because the reactor building provides the required protection.

The ABWR is designed as a single-unit facility. Therefore, the requirements of GDC 5 regarding the sharing of SSCs are not applicable. An application for a multi-unit facility will require review of the design for compliance with GDC 5.

The facility is designed to store unirradiated, low-emission fuel assemblies. Accidental damage to the fuel would release only minor amounts of radioactivity. These amounts would be treated by the standby gas treatment system (SGTS) which will limit any possible release of radioactivity from a fuel- handling accident to below acceptable values. Thus, the design meets the requirements of GDC 61 regarding appropriate containment, confinement, and filtering systems for a fuel storage facility.

The new fuel racks are not located near any high-energy lines, rotating machinery, or non seismic Category I SSCs and thus are protected from internally-generated missiles and the effects of pipe breaks by physical separation. Applicants referencing the ABWR design will be required to provide features which prevent placement of a new fuel assembly in other than its prescribed location. Because the racks are freestanding (i.e., no supports above the base of the racks), the racks and rack support structure provide the necessary dynamic stability for the racks, thus preventing them from tipping.

The new fuel racks are designed to ensure that k_{eff} does not exceed 0.95 under all normal and abnormal conditions. The standard safety analysis report (SSAR) states that the new fuel storage racks are purchased equipment. In the draft safety evaluation report (DSER) (SECY-91-235), interface requirements were identified for the combined license (COL) applicant to provide:

- An analysis of design details that prevent inadvertent placement of a fuel assembly in other than prescribed locations.

- Confirmatory dynamic and impact analyses. The input excitation for these analyses should utilize the horizontal and vertical response spectra provided in SSAR Section 3A.10.2.

- An analysis showing that the design of the new fuel storage racks will be such that k_{eff} will not exceed 0.98 with a fuel load of the highest anticipated reactivity, assuming optimum moderator conditions (foam, small droplets, spray, or fogging), as described in SRP Section 9.1.1.

In the DSER, the staff identified these requirements as Interface Information Items 28 and 29. They were later reclassified as Open Item 9.1.1-1 in the draft final safety evaluation report (DFSER). Upon further evaluation, the staff determined that these requirements can be accomplished by COL actions and are not needed for design certification. SSAR Sections 9.1.6.1 and 9.1.6.2 state that the COL applicant shall provide a confirmatory criticality analysis and dynamic and impact analyses of the new fuel storage racks. This is acceptable as information which will be evaluated during the COL review. On the basis of the above, DFSER Open Item 9.1.1-1 and DSER Interface Information Items 28 and 29 are resolved. Therefore, the staff concludes that design commitments in the SSAR meet GDC 62 as it relates to prevention of inadvertent criticality for fuel storage facilities.

In the DFSER, the staff requested GE to provide adequate design description and inspections, tests, analyses, and acceptance criteria (ITAAC) for the new fuel storage facility. This was part of DFSER Open Item 9.1.2-2. GE submitted a revised set of design description 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 the above, the applicable part of DFSER Open Item 9.1.2-2 is resolved.

The new fuel facility includes the fuel assembly storage racks and the concrete storage vault that contains the storage racks. The staff reviewed the applicant's proposed design criteria, design bases, and seismic classification for the new fuel storage facility related to the provisions necessary to maintain a subcritical array. The staff concludes that the design of the new fuel storage facility and supporting systems is acceptable and meets the requirements of GDC 2 as it relates to the ability of structures housing the facility (the reactor building) and the facility itself (designed to seismic Category I) to withstand the effects of natural phenomena such as earthquakes and GDC 61 as it relates to the facility design for fuel storage. GE adequately incorporated a COL action item requiring the applicant referencing the ABWR to provide information to ensure compliance with GDC 62 during the COL review. Therefore, the staff concludes that the design and related commitments comply with the guidelines of SRP 9.1.1 and are acceptable.

9.1.2 Spent Fuel Storage

The staff reviewed the ABWR spent fuel storage facility in accordance with the guidelines of SRP Section 9.1.2 to verify that is meets the requirements of GDC 2 as it relates to structures housing the facility and the facility itself being capable to withstand the effects of natural phenomena; GDC 4 as it relates to structures housing the facility and the facility itself being capable of withstanding the effects of environmental conditions, missiles, pipe whip, and jet impingement forces associated with pipe breaks; GDC 5 as it relates to shared SSCs important to safety being capable of performing required safety functions, GDC 61 as it relates to the facility design for fuel storage and handling of radioactive materials, GDC 62 as it relates to prevention of criticality, and GDC 63 as it relates to monitoring systems to detect conditions that could result in the loss of decay heat removal capabilities, to detect excessive radiation levels, and to initiate appropriate safety actions.

The spent fuel storage facility consists of fuel storage racks contained in the spent fuel storage pool in the reactor building. The spent fuel pool can store 270 percent of one full core load, in accordance with American Nuclear Society (ANS) 57.2 for the capacity of a single-unit facility. As stated in SSAR Section 9.1.4.2.8, defective fuel assemblies are placed in special fuel storage containers which are stored in equipment storage racks. Both the containers and the racks are designed for defective fuel.

The reactor building housing the facility is designed to seismic Category I criteria, as are the storage racks and other fuel storage facilities, including the gates between the spent fuel pool and other areas. In the DSER (SECY-91-235), the staff commented on the lack of information in the SSAR regarding the seismic classification of the spent fuel pool liner. This was DSER Open Item 44. Table 3.2-1 of the SSAR states that the liner will be seismic Category I. This meets the guidelines of SRP Section 9.1.2 and resolved DSER Open Item 44.

Applicants referencing the ABWR design will be required to provide features which prevent placement of a spent fuel assembly in other than its prescribed location. Because the racks are freestanding (i.e., no supports above the base of the racks) the racks and rack support structure provide the necessary dynamic stability for the racks, thus preventing them from tipping.

The reactor building is also designed to provide protection against flooding and tornado missiles as discussed in SSAR Sections 3.4.1 and 3.5.2 and evaluated in Sections 3.4.1 and 3.5.2 of this report. Therefore, the design satisfies the requirements of GDC 2 for protection of safety-related SSCs from natural phenomena and the guidelines of Position C.3 of RG 1.13, "Spent Fuel Storage Facility Design Basis," Revision 1, Positions C.1 and C.2 of RG 1.29, and Positions C.1 through C.3 of RG 1.117, "Tornado Design Classification," Revision 1.

The spent fuel pool is not located near any high-energy lines, rotating machinery, or non seismic Category I SSCs, and thus is protected from internally-generated missiles and the effects of pipe breaks by physical separation. The reactor building provides protection against externally generated missiles. Therefore, the design of the spent fuel facility satisfies GDC 4 and Positions C.1 through C.3 of RG 1.117.

The ABWR is designed as a single-unit facility. Therefore, the requirements of GDC 5 regarding the sharing of SSCs are not applicable. An application for a multi-unit facility will require review of the design for compliance with GDC 5.

Accidental damage to the fuel could release radioactive material. These amounts would be treated by the SGTS to limit any possible release of radioactivity from a fuel-handling accident to below acceptable values. Thus, the design meets the requirements of GDC 61 regarding appropriate containment, confinement, and filtering systems.

The fuel storage racks are designed to meet design requirements specifying that the fuel assemblies will be stored in an array that limits k_{eff} to 0.95 or less under all normal and abnormal conditions, such as a dropped fuel assembly, earthquake, or stuck fuel assembly. The SSAR specifies the design requirements.

In the DSER (SECY-91-235), the staff included interface requirements, that the COL applicant's purchase specification for the spent fuel storage racks should require the vendor to submit:

- Confirmatory criticality analysis, including the uncertainty value and associated probability and confidence level for the k_{eff} value determined by the analysis.

- Confirmatory load drop analysis, including the free fall of a fuel assembly and its associated handling tool.

These were identified as Interface Information Item 30 in the DSER and later reclassified as Open Item 9.1.2-1 in the DFSER. Upon further evaluation, the staff determined that these requirements can be accomplished by COL actions and are not needed to be met for design certification. SSAR Sections 9.1.6.3 and 9.1.6.4 state that the COL applicant shall provide a confirmatory criticality analysis and load drop analysis of the spent fuel storage racks. This is acceptable. On the basis of the above, DFSER Open Item 9.1.2-1 and DSER Interface Information Item 30 are resolved. Therefore, the staff concludes that the design and associated commitments meet GDC 62 as it relates to prevention of inadvertent criticality.

In the DSER, the staff determined that the design of the storage pool includes the provisions for radiation monitoring systems described in SSAR Section 11.5 that satisfy, in part, the requirements of GDC 63 regarding the monitoring of spent fuel. In the DSER, the staff identified, as Open Item 45, a lack of information regarding pool level indication. Subsequently, GE provided additional information which states that the pool level will be monitored and alarmed (on high and low level) locally and in the control room. Leakage flow detectors in the pool drains and pool liners are also provided and alarmed in the control room. The staff concludes that these provisions will provide adequate monitoring capability and meet the requirements of GDC 63. Therefore, DSER Open Item 45 is resolved.

In the DFSER, the staff requested GE to provide adequate design description and the ITAAC relating to the spent fuel storage facility. This was part of DFSER Open Item 9.1.2-2. Subsequently, GE provided a revised set of design description 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 the above, the applicable part of DFSER Open Item 9.1.2-2 is resolved.

The spent fuel storage facility includes the spent fuel storage racks and the spent fuel storage pool that contains the racks. The staff reviewed the design criteria, design bases, and safety classification for the spent fuel storage facility and the provisions necessary to maintain a subcritical array. The staff concludes that the design and related commitments of the spent fuel storage facility and supporting systems conform with the Commission's regulations as set forth in GDC 2, 4, 5, 61, and 63. GE adequately incorporated a COL action item requiring the applicant referencing the ABWR to submit information to ensure compliance with GDC 62.

The staff concludes that the spent fuel storage design and commitments meet the guidelines of SRP 9.1.2 and the requirements of GDC 2 as it relates to structures housing the facility and the facility itself being capable of withstanding the effects of natural phenomena; GDC 4 as it relates to structures housing the facility and the facility itself being capable of withstanding the effects of environmental conditions, missiles, pipe whip, and jet impingement forces associated with pipe breaks; GDC 5 as it relates to shared SSCs important to safety being capable of performing required safety functions; GDC 61 as it relates to the facility design for fuel storage and handling of radioactive materials; GDC 62 as it relates to prevention of inadvertent criticality; and GDC 63 as it relates to monitoring systems to detect conditions that could result in the loss of decay heat removal capabilities, to detect excessive radiation levels, and to initiate appropriate safety actions. The design and related commitments, therefore, are acceptable.

9.1.3 Spent Fuel Pool Cooling and Cleanup System

The staff reviewed the spent fuel pool cooling and cleanup (FPC) system and its makeup system in accordance with SRP Section 9.1.3 to verify that they meet the requirements of GDC 2 as it relates to structures housing the system and the system itself being capable of withstanding the effects of natural phenomena; GDC 4 as it relates to structures housing the system and the system itself being capable of withstanding the effects of external missiles; GDC 5 as it relates to shared SSCs important to safety being capable of performing required safety functions; GDC 44 as it relates to the system's ability to reliably transfer heat loads from safety-related SSCs (including suitable redundancy and isolability); GDC 45 and 46 as they relate to inspection and functional testing, respectively; GDC 61 as it relates to the facility design for fuel storage and handling of radioactive materials; GDC 63 as it relates to monitoring systems provided to detect conditions that could result in the loss of decay heat removal capabilities, to detect excessive radiation levels, and to initiate appropriate safety actions; and 10 CFR Part 20 as it relates to keeping radiation doses as low as reasonably achievable (ALARA).

The FPC system is a non-safety-related system designed to remove decay heat generated by spent fuel assemblies in the spent fuel storage pool; to maintain water quality and clarity; and to remove corrosion products, fission products, and other impurities from the pool water. The system consists of all components and piping from the system inlet at the fuel pool to the system outlet at the fuel pool, piping used to carry the fuel pool makeup water, and the cleanup filter/demineralizers (shared with the suppression pool cleanup (SPCU) system) to the point of discharge to the radwaste system. Specifically, the system includes two 50-percent capacity circulating pumps, two 50-percent capacity heat exchangers, two filter/demineralizers, one postdemineralizer strainer, two skimmer surge tanks, piping, valves, controls, and instrumentation. The pool water is circulated by means of overflow through skimmers around the periphery of the pool and a scupper at the end of the transfer canal; the overflow is collected in the surge tanks. Normally, one of the two circulating pumps draws water from these tanks and discharges it through a common header to either of two filter/demineralizers connected in parallel. System flow then passes through two heat exchangers in parallel, cooled by water from the reactor building cooling water (RCW) system, and returns to the spent fuel storage pool. The system includes a bypass line around the cleanup portion of the system (the filter/demineralizers). Circulating pumps can be powered from the combustion turbine generators (CTGs) if normal power is not available.

The requirements of GDC 5 regarding the sharing of SSCs do not apply because the ABWR is designed as a single-unit facility. An application for a multi-unit facility will require review of the design for compliance with GDC 5.

Filtering and ion exchange together maintain clarity and purity of pool water. The filter/demineralizers maintain total corrosion product metals at 30 parts per billion (ppb) or less with a pH range of 5.6 to 8.6 at 25 °C (77 °F) for compatibility with fuel storage racks and other equipment. Conductivity is maintained at less than 1.2 micro siemens (μS)/cm (3.0 micro Mho/in.) at 25 °C and chlorides less than 20 ppb. Maintaining the pool water purity within these levels will ensure that there are no adverse chemical interactions with the fuel storage racks and other pool equipment supplied by the applicant. Each filter unit in the filter/demineralizer subsystem has adequate capacity to maintain the desired purity of the pool under normal operating conditions.

The FPC system is housed in the reactor building (secondary containment area), a seismic Category I structure designed to protect against flood and tornado missiles. Thus, the system meets Position C.2 of RG 1.13 for protection of this support system for the fuel storage facility from natural phenomena such as tornados, winds, and externally-generated missiles.

Normal makeup water is supplied to the storage pool by the non-safety-related makeup water (condensate) system (MUWC). Backup to the normal makeup will also be available from the nonsafety-related SPCU system. Additionally, an emergency safety-related, seismic Category I makeup water source to the spent fuel pool will be provided by FPC connections to the residual heat removal (RHR) system, which draws water from a safety-related water source, that is, the suppression pool. FPC system piping from the spent fuel pool to the RHR system is classified as safety related. In a letter dated July 23, 1990, GE stated that the FPC system, except for the filter demineralizers, is designed to seismic Category I, Quality Group (QG) C standards. Although the entire system is not classified as seismic Category I, QG C standards, the nonsafety-related portions of the system that could affect any SSCs important to safety if they failed during a seismic event, are designed to ensure their integrity under seismic loading. Thus, the system meets Positions C.1 and C.2 of RG 1.29 with respect to the seismic design of the safety-related and nonsafety-related portions of the system, respectively.

There are no connections to the spent fuel pool that would cause the pool water to be drained below a safe shielding level. All lines that connect to the pool and extend below the safe level of the pool are equipped with siphon breakers, check valves, or other suitable devices to prevent inadvertent pool draining. Drainage paths are interconnected behind the liner welds. Leakage through the pool liner is collected in a drain system and transferred to an equipment drain tank. No piping connections penetrate the fuel pool liner to the fuel storage pool. The FPC system is designed so that no single failure, malfunction, or misoperation of the active components will uncover the stored fuel. Thus, the system meets Position C.6 of RG 1.13.

As mentioned above, the system includes a seismic Category I makeup water source that uses a safety-related piping segment of the FPC system return line. Check valves in the return lines that lead to the submerged FPC water return diffusers protect against backflow. Also, the system includes redundant safety-related isolation devices to facilitate isolation of the safety-related makeup portion of the system from the nonsafety-related SPCU system. The SPCU system makeup uses the same FPC system safety-related piping segment that the RHR makeup uses, and is, therefore, equally vulnerable to a passive failure. In the DSER (SECY-91-235), the staff noted that such a failure will also affect the decay heat removal capability of both the FPC and the RHR systems, because both of these use the same safety-related piping segment of the FPC system return line. Subsequently, GE indicated that the pipe segments are protected from the effects of the failure of other components and systems. GE also submitted the results of an analysis that show that given a failure of this pipe segment, alternative cooling and makeup capabilities exist that will maintain the pool level and temperature within acceptable limits. The SSAR Section 9.1.6.10 clarifies the common piping through which FPC, RHR, and SPCU flow by stating that the COL applicant will ensure that the RHR system connections are adequately protected from the effects of pipe whip, internal flooding, internally-generated missiles, and the effects of a moderate energy pipe rupture in the vicinity. This is acceptable. Furthermore, responding to an early staff request (request for additional information (RAI) Question 410.37), GE stated that fire hoses will also be available to supply makeup water to the pool. This is also acceptable. The staff concludes that the FPC system design meets Position C.8 of RG 1.13.

Acceptance Criteria II.1.a and b of SRP Section 9.1.3 state that GDC 2 and 4 need not apply to the FPC system design if the design includes a safety-related makeup water system, including its source, and a safety-related fuel pool area ventilation and filtration system designed in accordance with the guidelines of RG 1.52. As noted above, a major portion of the FPC system is not safety-related and, therefore, does not meet Position C.1 of RG 1.13. The design relies instead on the SGTS, a safety-related ventilation and filtration system for the fuel pool area and relies on a safety-related makeup water system. In Section 6.5.1 of the DSER (SECY-91-235), the staff stated that the design of the SGTS had been identified as Open Item 30 in SECY-91-153 because GE had not demonstrated its compliance with all the applicable positions of RG 1.52 (e.g., redundancy of filters in the SGTS). The staff stated its concerns about safety-related makeup capability for the fuel pool and the SGTS design in the DSER (SECY-91-235) as part of Open Item 46 (Section 1.8 of the DSER (SECY-91-235) grouped four concerns of DSER Section 9.1.3 as Open Item 46.).

GE subsequently committed to revise the design of the SGTS to include a second redundant filter train. Because the safety-related makeup for the pool is not totally independent, but relies on the safety-related portion of the FPC system, GDC 2 and 4 are applicable for that portion of the FPC system. As stated above, the system is protected against the effects of adverse natural phenomena by virtue of its location in the reactor building. Also, SSAR Sections 3.5.1 and 3.6.1, and SSAR Table 3.6-2, include spent fuel pool cooling for protection against the effects of missiles and piping failures. The staff noted that GE's submittal dated June 2, 1989, further stated that the major components of the FPC system are located in separately-shielded rooms and that barriers, restraints, and equipment compartments protect fuel pool cooling components against failure of high-energy piping systems. In the DSER (SECY-91-235), the staff stated, as part of Open Item 46, the need for GE to confirm that the safety-related portion of the system is protected from moderate-energy piping failures in the vicinity. The SSAR states that the safety-related portions of the system are protected from moderate-energy pipe breaks. This resolved this part of DSER (SECY-91-235) Open Item 46. GE also committed to modify the design of the SGTS to resolve the part of DSER (SECY-91-235) Open Item 46 associated with the SGTS design, and include the required ventilation and filtration system required as an alternative to designing the entire FPC system as a safety-grade system. Incorporation of the identified changes to the SGTS into the SSAR was DFSER Confirmatory Item 9.1.3-1. The staff reviewed the SSAR and concluded that the SGTS has been modified to include two redundant safety-grade filter trains and that the system complies with RG 1.52. This modification allows the ABWR FPC system to meet the alternate design guidance provided in SRP Sections 9.1.3.II.1.a and b. The SSAR clarifies the common piping through which FPC, RHR, and SPCU flow. SSAR Section 9.1.6.10 notes that the COL applicant will provide the protective features for this section of piping. The staff finds this approach and commitment acceptable. Therefore, DFSER Confirmatory Item 9.1.3-1 is resolved.

Based on the discussions above, the staff concludes that the design of the FPC system and the structures housing the system meet the criteria in GDC 2 and 4.

The staff based its evaluation of the capability of the FPC system to handle the decay heat load in the fuel pool on the guidelines in SRP Section 9.1.2, Paragraphs II.1.d.(4) and III.1.h. Each of the two FPC system heat exchangers is rated at 6.91E+9 J/h (6.55E+6 BTU/hr) at the design temperature of 52 °C (125 °F). Based on an independent staff calculation of the spent fuel heat load using Branch Technical Position (BTP) ASB 9-2, the maximum normal heat load at 150 hours after shutdown is approximately 18E+9 J/h (1.7E+7 BTU/hr). This heat load is based on the decay heat generated by one refueling load (35 percent of the core) at equilibrium conditions 150 hours after shutdown, plus a refueling load at 365 days, and one at 400 days after shutdown in accordance with SRP Section 9.1.3.III.1.h.iv for pools with storage capacity greater than 1.3 cores. The FPC heat exchangers are inadequately sized to handle the decay heat load in the fuel pool at 150 hours after shutdown as outlined by SRP Section 9.1.3, Paragraphs II.1.d.(4) and III.1.h, to meet GDC 44. The heat load beyond the FPC heat exchanger capacity is to be handled by the RHR system, which includes a segment of the non-safety-related suction portion of the FPC system and a safety-related piping segment of the FPC system return line for performing its decay heat removal function. The RHR system also supplements the FPC system capabilities to maintain pool water temperature below 66 °C (150 °F) under the maximum abnormal heat load, that is, a full core offload. GE has based the FPC heat exchanger design on the heat load at 21 days after shutdown, the time at which the fuel transfer canal can be closed without requiring supplemental cooling. Independent calculations of the heat load at this time confirm the GE-calculated value of 12.0E+9 J/h (1.14 x 10^7 BTU/hr). At this time after shutdown, the FPC system is adequately sized to handle the decay heat load of the fuel pool.

In the DSER (SECY-91-235), the staff stated that the FPC system design does not accommodate any single active failures. A single heat exchanger has insufficient heat removal capability to maintain the fuel pool temperature at an acceptable level at all times. GE also indicated that for some active single failures, using the RHR system may be necessary to limit the temperature of the fuel pool to less than 60 °C (140 °F). As a result, in addition to this concern, the DSER (SECY-91-235) identified as part of Open Item 46, the apparent undersizing of the FPC heat exchangers for the ABWR. The staff also identified as a concern the sole reliance on the RHR system to supplement the FPC system's normal maximum spent fuel heat load removal capability during certain situations (e.g., all times preceding 21 days after shutdown and after 21 days when there is a single active failure coincident with a loss of all offsite power). To address this concern, the SSAR provides additional commitments regarding the design of the RHR connection and alternative makeup capabilities so that the connection will be protected from the effects of high- energy and moderate-energy pipe breaks, and any other hazards to limit the likelihood of failure of the SPCU and RHR spent fuel pool makeup and cooling capability. The SSAR also addresses the capability to provide alternative makeup within an acceptable time limit. Fire water makeup to the spent fuel pool will be available within 30 minutes after the failure of the fuel pool cooling capability. GE's analysis showed that the spent fuel pool would not reach 60 °C (140 °F) under the maximum normal heat load at 21 days in less than this time. It concluded that over 6 hours would be required for the pool to heat up to this level. Under conditions of a full core off-load at 21 days after shutdown, the time to reach boiling was estimated to be over 16 hours. The staff's independent calculations verify that the results are representative of the times available to provide additional makeup. The available time to realign the fire water system appears to be sufficient to make the configuration changes. Incorporation of this information into the SSAR was DFSER Confirmatory Item 9.1.3-2. The SSAR provides this information which adequately addresses the part of DSER Open Item 46 (SECY-91-235) regarding the use of the RHR system as an integral part of the FPC system from 150 hours after shutdown to 21 days after shutdown, and under the conditions of a loss of offsite power (LOOP) and a single active failure. This resolved the applicable part of DSER Open Item 46 and DFSER Confirmatory Item 9.1.3-2. Therefore, the staff concludes that the FPC system design complies with GDC 44 regarding the system's ability to transfer heat from systems important to safety under both normal and accident conditions and also complies with the RHR capability and coolant inventory maintenance requirements of GDC 61.

The pumps, heat exchangers, and the filter/demineralizers are each located in separate shielded rooms within the secondary containment portion of the reactor building. Individual components can be isolated from the rest of the system during operation. The system design allows for isolation and accessibility of the components. This complies with the requirements of GDC 61 for shielding, containment, and confinement.

One division of the FPC system (pump, heat exchanger, and filter-demineralizer) is normally in operation. The second pump will be operated periodically to ensure its operability or to allow the operating pump to be removed from service. Periodically, the FPC system components will be visually inspected. From this discussion, the staff concludes that the requirements of GDC 45 and 46 regarding the inspection and testing of cooling water systems are satisfied for the components in the safety-related portion of the FPC system. The design also complies with the testing and inspection requirements of GDC 61.

The components of the FPC system are accessible and can be isolated to allow for periodic testing. The FPC system is located within the secondary containment portion of the reactor building, thus providing adequate containment. The FPC system contains design features to prevent reduction in fuel storage coolant inventory under accident conditions in accordance with Position C.6 of RG 1.13. The decay heat removal capabilities of the FPC system are designed to meet the heat load requirements of the fuel pool at 21 days after shutdown, not the requirements at 150 hours after shutdown. Use of the RHR system is required to meet the heat load requirements of the FPC system during the period between 150 hours and 21 days after shutdown. Additionally, as identified in the DSER, the FPC system is not capable of handling the decay heat load assuming both a LOOP and a single active failure. Again the RHR system, in conjunction with the operable portions of the FPC system, is the system used to meet the heat load requirements. As indicated previously, GE has provided information justifying this system arrangement. The design includes measures to ensure RHR system operability in the spent fuel cooling mode and alternative makeup and cooling capabilities. Therefore, the staff concludes that the FPC system meets all requirements of GDC 61.

The control room includes instrumentation to monitor the fuel pool level and temperature and area radiation for the fuel storage and handling areas. The control room also includes high and low fuel pool level and high-radiation alarms. Interconnected drainage paths behind the liner welds include instrumentation to detect and measure liner leaks. Several FPC system parameters are also displayed and recorded in the control room and locally. These include pump suction pressure and pump flow and system water discharge temperature. The pump suction pressure and flow are also alarmed in the control room. These monitoring and alarming devices enable plant personnel to quickly begin appropriate safety actions such as erecting temporary shields to reduce radiation, supplying makeup water to the pool through a remotely operated valve, and actuating the SGTS on a high radiation signal on the refueling floor). Indications of high system temperature allow the operator to manually bypass system flow from the filter-demineralizers to protect the resins from damage. Thus, the system instrumentation meets the requirements of GDC 63 regarding the monitoring of fuel storage facilities and the initiation of appropriate safety actions.

Based on its review as discussed above, the staff concludes the following:

1. Designing the FPC system for the ABWR as a non-safety-related system, except as stated above, meets the applicable acceptance criteria of SRP Section 9.1.3.

2. The cooling portion of the FPC system complies with the requirements of GDC 44 and 61 for decay heat removal, inspection and testing, containment and confinement of stored fuel, maintenance of fuel storage coolant inventory, and shielding requirements.

3. The safety-related portion of the FPC system complies with the requirements of GDC 4 for its protection from the effects of missiles and high- and moderate-energy piping failures in the vicinity.

4. The safety-related portion of the FPC system complies with the requirements of GDC 2 for its protection from the effects of adverse natural phenomena.

5. The cooling portion of the FPC system complies with the requirements of GDC 45, 46, 61, and 63, for inspection, testing, and monitoring.

In the DSER (SECY-91-235), the staff stated that conclusions 1, 2, and 3 are subject to:

1. GE's confirmation of protection of the system's safety-related portions from the effects of moderate-energy piping failures in the vicinity. The SSAR includes a commitment that the COL applicant will include features to protect the common portion of FPC piping from pipe and equipment failures. This commitment is acceptable.

2. Resolution of the staff's concerns relating to (a) the system's decay heat removal capability, and (b) the SGTS design as it pertains to the requirement for redundant filters. The information provided in the SSAR regarding the capability of the operators to align the fire water system to supply pool makeup and cooling and the temperature response of the pool water after the loss of all cooling, adequately addresses the staff's concern about the system's decay heat removal capability. The staff's concern regarding the design of the SGTS was identified as Confirmatory Item 9.1.3-3 in the DFSER. Subsequently, the SGTS design was modified (see SSAR Section 6.5.1 for a full discussion of the SGTS and Section 6.5.1 of this report for the staff evaluation of the SGTS). As a result of the design modifications, the staff concludes that the SGTS provides adequate redundancy. Therefore, Confirmatory Item 9.1.3-3 is resolved.

In the DSER, the staff raised the issue of the technical specification (TS) implications of using the RHR system as an integral part of the FPC system, as part of Open Item 46. In the DFSER, the staff stated that GE's response did not explain how the RHR system can be used for the maximum normal heat load removal without violating the TS requirements for availability of the system for other purposes in Mode 5 (refueling mode). This was DFSER Open Item 9.1.3-1. Subsequently, GE submitted information in which it explained that, during refueling, containment is not required to be operable unless fuel is being moved. Therefore, suppression pool cooling and wetwell/drywell spray is not required while in the refueling mode. In a situation where augmented fuel pool cooling may be required, the shutdown cooling function is not affected since in this mode, the pool gates are removed and the reactor and spent fuel pool are common. RHR in the low-pressure flooder (LPFL) mode may be needed during refueling. In this case, TS require only two emergency core cooling system (ECCS) divisions to be operable. Five divisions are available in the ABWR design to meet this need (two divisions of high-pressure core flooder (HPCF) and three divisions of RHR). TS consider an RHR division aligned in the shutdown cooling mode to be considered operable if it can be manually realigned to the LPFL mode. The staff notes that the ABWR design allows this reconfiguration. Therefore, based on the evaluation above, the staff concludes that the ABWR design is capable of providing adequate availability of RHR for all required heat removal functions during all modes. This resolved DFSER Open Item 9.1.3-1.

The ACRS inquired about the ability of the ABWR design to initiate and/or maintain spent fuel pool cooling during and following an accident, including a LOCA. This question arose as a result of design concerns identified at an operating reactor in a 10 CFR Part 21 notification. Specifically, it was postulated that if a LOCA occurred, the resulting environmental conditions in the secondary containment would render the fuel pool cooling and cleanup system (FPC) equipment inoperable while, at the same time, preventing implementation of recovery actions. Should this occur, water in the spent fuel pool would boil, resulting in excessive condensation leading to flooded conditions. In addition, boiling conditions could render the standby gas treatment system (SGTS) inoperable, so that any radioactive material resulting from failed fuel in the pool (due to lack of pool cooling) could be released to the environment without first being processed by the SGTS.

The staff discussed the postulated scenario with GE and asked whether the ABWR design would be able to initiate and/or maintain spent fuel pool cooling during and following an accident, including a LOCA, in accordance with GDC 44 and 61. Subsequently, GE stated that the ABWR design can accommodate this accident for the following reasons:

(1) GE determined that a break in the reactor water cleanup system (CUW) presents the most limiting environmental conditions inside secondary containment for all design-basis accidents, including a LOCA. Safety-related equipment located inside the secondary containment is environmentally qualified to remain functional given a CUW pipe failure inside the secondary containment. Therefore, this equipment would remain available to perform its safety-related functions following an accident.

(2) In Section 9.1.3 of the SSAR, GE provided the results of a pool heatup analysis. The results showed that, should pool cooling fail with the maximum abnormal heat load in the pool, it would take approximately 16 hours for the pool to reach boiling conditions. This allows sufficient time for operators to implement manual recovery actions (manipulation of manual valves, etc.) well before boiling conditions develop. The system connections to each of the two residual heat removal (RHR) system divisions connected to the FPC utilizes a motor-operated valve (MOV) on the FPC inlet from the RHR system and a manual valve on the FPC outlet to the RHR system. The inlet valves are remotely operated; the outlet valves, however, require the operator to manually manipulate the valves. GE states in the SSAR that these manual valves will be accessible following an accident in sufficient time to permit an operator to align the RHR system to prevent the SFP from boiling.

(3) Two of the three safety-related RHR divisions are available to provide cooling water to the spent fuel pool. Either division is sufficient to cool the spent fuel pool given the worst-case heat load in the pool. Therefore, given a single failure of an RHR division, one remaining division can provide spent fuel pool cooling while the other provides cooling for the reactor.

(4) In the unlikely event that pool boiling were to occur, safety-related makeup water can be provided from the suppression pool through either of the MOVs on the FPC inlet from RHR. The resulting environmental conditions are bounded by the worst-case environmental conditions postulated in the CUW line break discussed earlier. Each division of safety-related equipment inside the secondary containment is physically and electrically separated to prevent flood conditions from affecting more than one division of safety-related equipment.

(5) The system is not safety related except for those portions from RHR to the FPC system and from the FPC system to the RHR system. However, the FPC system is designed to seismic Category I standards (with the exception of the filter-demineralizers) to ensure it remains functional during seismic events. Alternative means of power to the system is available through the combustion turbine generator (CTG) which is the alternate ac (AAC) power source for the ABWR design. Should a loss of preferred power (LOPP) or station blackout (SBO) occur, the CTG can be used to provide power to the FPC system. Therefore, the system will be available to provide spent fuel pool cooling under both normal and accident conditions.

On the basis of this information, the staff concludes that the ABWR design ensures that cooling can be initiated and/or maintained during both normal and accident conditions, including a LOCA.

In the DFSER, the staff requested GE to submit adequate design description and the ITAAC relating to the FPC system. This was DFSER Open Item 9.1.3-2. The adequacy and acceptability of the design description and the ITAAC are evaluated in Section 14.3 of this report. On the basis of the above, DFSER Open Item 9.1.3-2 is resolved.

The staff's review of the FPC system included all components and piping of the system from the inlet to and the exit from the storage pool, the seismic Category I makeup source and piping used for the fuel pool makeup, the cleanup system filter/demineralizers, and the regenerative process to the point of discharge to the radwaste system. The scope of the FPC system review included layout drawings, process flow diagrams, piping and instrumentation diagrams, and descriptive information for the system and supporting systems that are essential to safe operation. The cooling portion of the system and the emergency primary makeup system are designed to seismic Category I, QG C requirements, since they are necessary to remove decay heat from the spent fuel and to prevent fuel damage that could lead to unacceptable releases of radioactivity. The staff concludes that the design of the FPC system and its makeup systems meet the guidelines of SRP 9.1.3 and the requirements of GDC 2, 4, 5, 44, 45, 46, 61, and 63, subject to incorporation of information regarding protection of the filter-demineralizer resins.

9.1.4 Light Load-Handling System (Related to Refueling)

The light load-handling system (LLHS) provides the means of transporting, handling and storing fuel (both new and spent fuel) in the reactor building. The staff reviewed the fuel-handling system, in accordance with the guidelines of SRP Section 9.1.4. Staff acceptance of the LLHS is based on meeting the requirements of 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 SSCs important to safety being capable of performing required safety functions; GDC 61 as it relates to radioactivity release as a result of fuel damage, and the avoidance of excessive personnel radiation exposure; and GDC 62 as it relates to prevention of inadvertent criticality.

The transfer of new fuel assemblies between the uncrating area and the new fuel inspection stand and/or the new fuel storage vault is accomplished using a 4500-kg (5-ton) auxiliary hook on the reactor building crane equipped with a suitable grapple. A 450-kg (1000-lb) auxiliary hoist on the reactor building crane is used with an auxiliary fuel grapple to transfer new fuel from the new fuel vault to the fuel storage pool. From there, the fuel is handled by the telescoping grapples on the refueling platform or auxiliary hoists.

The refueling machine is a gantry crane used to transport fuel and reactor components to and from pool storage and the reactor vessel. The platform spans the fuel storage and vessel pools. A telescoping mast and grapple suspended from a trolley system is used to lift and orient fuel bundles for placement in the core or storage rack. The fuel grapple hoist has a redundant load path so that no single component failure can result in a fuel bundle drop. Interlocks on the platform (1) prevent hoisting a fuel assembly over the vessel with a control rod removed, (2) prevent collision with fuel pool walls or other structures, (3) limit travel on the fuel grapple, (4) interlock grapple hook engagement with hoist load and hoist up power, and (5) ensure correct sequencing of the transfer operation in the automatic or manual mode.

The refueling machine also has two (4.71 kN (1060 lbf) and 9.87 kN (2200 lbf)) auxiliary hoists. The hoists can be used normally with appropriate grapples to handle control rods, guide tubes, fuel support pieces, sources, and other internals of the core.

The refueling machine is designed structurally to seismic Category I standards and the entire system is housed within the reactor building, which is a seismic Category I structure designed to protect against flood and tornados.

In the DSER (SECY-91-235), the staff identified as Open Item 47, the impact of the failure of the fuel inspection stand during a safe-shutdown earthquake (SSE) because the stand was not classified as seismic Category I, and the radiological and criticality impact of the fall of the new fuel stand was not satisfactorily addressed in the SSAR. The SSAR commits the new fuel inspection stand to be firmly attached to a wall so that it does not fall or dump personnel into the fuel pool during an earthquake. This is acceptable and resolved DSER Open Item 47.

The final inspection stand design meets the requirements of GDC 2 and the guidelines of Positions C.1 and C.6 of RG 1.13, and Positions C.1 and C.2 of RG 1.29, relating to the protection of safety-related equipment and spent fuel from the effects of earthquakes, because it is housed and attached to a seismic Category I structure.

The reactor building crane main hook is used to move the spent fuel cask, and the auxiliary hook is used to move new fuel from the new fuel vault to the spent fuel storage pool. Interlocks and procedures prevent the main hook of the reactor building crane from traversing over the spent fuel pool or the new fuel storage vault while carrying a heavy load. This is acceptable.

The requirements of GDC 5 for sharing SSCs do not apply because the ABWR is designed as a single-unit facility. An application for a multi-unit facility will require review of the design for compliance with GDC 5.

In Section 9.1.2 of the DSER (SECY-91-235), the staff stated that the COL applicant's purchase specification should include an interface requirement for the vendor to submit a confirmatory analysis of the consequences of dropping a fuel assembly and its associated handling tool from above the spent fuel storage racks. The staff determined that this requirement can be accomplished by COL actions as resolved in Section 9.1.2 of this report.

This additional information will ensure compliance with GDC 62 regarding protection of the fuel from inadvertent criticality.

The fuel-handling machine grapple retracts to a maximum position which ensures that the fuel will not be lifted beyond a safe-shielding height. Various electrical and mechanical interlocks prevent excessive cable load, movement over the core with a control rod removed, withdrawal of a control rod with fuel of the core, and collisions of fuel with surrounding structures. The refueling machine has redundant load paths and is designed to retain its load during a seismic event. Finally, the ABWR design includes instrumentation to monitor radiation levels on the refueling floor and initiated protective actions on a high-radiation level (realignment of heating, ventilation, and air conditioning (HVAC) systems and SGTS initiation). The staff concludes that the design of the light load-handling system meets the guidelines of Position C.3 or RG 1.13 and ANS 57.1, and the requirements of GDC 61 for the release of radioactivity resulting from fuel damage and protection of personnel from excessive radiation exposure.

In the DFSER, the staff requested GE to submit adequate design description and the ITAAC for the LLHS. This was DFSER Open Item 9.1.4-1. GE submitted a revised set of design description 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 the above, DFSER Open Item 9.1.4-1 is resolved.

The LLHS includes all components and equipment used in moving fuel and other related light loads between the receiving area, storage areas, and reactor vessel. Based on the review of the applicant's proposed design criteria and design bases for the LLHS, and the requirements for the safe operation of the LLHS, as discussed above, the staff concludes that the design of the LLHS and supporting systems are in conformance with the guidelines of SRP 9.1.4, and the Commission's regulations as set forth in GDC 2 (protection from natural phenomena), GDC 5 (sharing of SSCs), and GDC 61 (fuel storage handling and radioactivity control) and will conform to GDC 62 (protection of criticality in fuel storage handling), and are acceptable.

9.1.5 Overhead Heavy Load-Handling System

Inadvertent operations and equipment malfunctions caused by a load drop during critical load-handling operations could cause (1) a release of radioactivity to the environment above acceptable limits, (2) a criticality accident, (3) the inability to cool fuel within the reactor vessel or spent fuel pool, or (4) the inability to achieve safe shutdown of the reactor when needed. Therefore, critical load-handling equipment and operations are required to prevent these problems by built-in design features and operating procedures. Additionally, safe handling of loads includes design considerations for maintaining occupational radiation exposures as low as practicable during transportation and handling. The overhead heavy load-handling system (OHLHS) consists of all components and equipment used in moving all heavy loads, that is, loads weighing more than one fuel assembly and its associated handling device.

The staff reviewed the system in accordance with SRP Section 9.1.5. Staff acceptance of the OHLHS design was based on meeting the requirements of GDC 2 as it relates to the ability of structures, equipment, and mechanisms to withstand the effects of natural phenomena; GDC 4 as it relates to protection of safety-related equipment from the effects of dropped loads; GDC 5 as it relates to sharing of equipment and components important to safety; and GDC 61 as it relates to the safe handling and storage of fuel. In addition, the acceptance criteria for the OHLHS includes meeting the guidelines of ANS 57.1, "Design Requirements for Light Water Reactor Fuel Handling System," and 57.2, "Design Objectives for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Plants." The staff used the guidelines contained in the SRP Section III titled "Review Procedures" and in NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants."

The OHLHS includes equipment necessary for the safe disassembly and reassembly of the reactor vessel head and internals during refueling operations and for the safe handling of the spent fuel cask. The reactor building crane will handle heavy loads in the containment and fuel-handling area of the reactor building. This single-failure-proof crane will handle the reactor vessel head, shroud head, steam separator, steam dryer, pool gates, new fuel from the reactor building entry hatch to new fuel storage, spent fuel shipping cask, some reactor internal pump (RIP) components during pump servicing, and other miscellaneous loads during operation and outage. Also, the OHLHS includes reactor servicing equipment, upper and lower drywell servicing equipment, and main steam tunnel (MST) area servicing equipment, which are used for safe handling of main steam isolation valves (MSIVs), safety/relief valves (SRVs), RIP motors, heat exchangers and pump components, control rod blades and guide tubes, and fine motion control rod drive (FMCRD) components during their removal and reinstallation or replacement. This servicing equipment includes among other things, monorail and its hoist, transportation carts, equipment hatchway hoist, steam tunnel crane and its hoist, servicing platform, refueling platform, equipment platform, and lower drywell RIP hoist.

The OHLHS equipment described above is housed in the reactor building in which the associated heavy load-handling operations are performed. The building is a flood-protected and tornado-protected structure and thus, the OHLHS described above is protected against the effects of adverse natural phenomena. The spent fuel storage facility and the new fuel storage vault are seismic Category I. Also, as stated in Section 9.1.3 of this report, the spent fuel storage facility meets Position C.6 of RG 1.13, "Spent Fuel Storage Facility Design Basis." The OHLHS equipment and components are not seismic Category I except for the upper and lower drywell servicing equipment, refueling platform, equipment platform, and reactor service platform. Though the load- handling equipment need not be seismic Category I, the non seismic Category I equipment that can adversely impact SSCs important to safety should they fail during a seismic event, have to meet Position C.2 of RG 1.29. SSAR Table 3.2-1 states that the reactor building and refueling platform cranes are designed to hold their positions during an SSE. The table further indicates that the non seismic Category I system equipment that can adversely impact safety-related SSCs during an SSE, are designed to ensure their integrity under seismic loading resulting from an SSE. The staff identified in the DSER (SECY-91-235), as part of Open Item 48, (Section 1.8 of the DSER grouped eight concerns of DSER Section 9.1.5 as Open Item 48. Five are addressed in the items that follow. The remaining three are addressed in subsequent paragraphs.) several concerns pertaining to compliance of the OHLHS with GDC 2 as it relates to protection of safety-related equipment from the effects of adverse natural phenomena:

1. In the DSER (SECY-91-235), the staff noted that Note x for SSAR Table 3.2-1 needed to be clarified to show that the reactor building and refueling platform cranes will hold their loads under the dynamic effects of an SSE.

   Note x of Table 3.2-1, SSAR Amendment 20, stated that the cranes are capable of holding their loads under operating-basis earthquake (OBE) conditions but did not address the SSE. This part of DSER Open Item 48 was not resolved and was reclassified as Open Item 9.1.5-1 in the DFSER. In the SSAR, Note x of Table 3.2-1, states that the cranes are designed to hold their load and maintain their position during an SSE. This is acceptable and resolved DFSER Open Item 9.1.5-1.

2. In the DSER (SECY-91-235), the staff noted that the above criterion should be applied for all other applicable (i.e., affecting safe-shutdown equipment) non seismic Category I load-handling equipment.

   In response, GE indicated that the use of the steam tunnel handling equipment is limited to times when the reactor is shut down. Failure of the handling equipment at this time would not result in the loss of safety functions (identified in SSAR Amendment 17). The rationale for not requiring this equipment to have the capability of holding its loads during an SSE is acceptable. However, sufficient justification for the lack of the application of a seismic criterion, such as that identified in item 1 above, has not been provided for the remaining devices in the control building, secondary containment, and clean zone of the reactor building. This part of DSER Open Item 48 was not resolved and was reclassified as Open Item 9.1.5-2 in the DFSER. GE later added Amendment 24 to the SSAR which stated that, with the exception of containment, the MST, and the refueling floor, no safety-related components of one division shall be routed over another safety-related division. Therefore, a dropped load cannot adversely effect more than one safety-related division.

   GE also indicated that the amount of safety-related equipment present inside primary containment, and the divisional equipment is widely separated (120 degrees) around the containment. Therefore, the likelihood of a dropped load damaging more than one train of safety-related equipment is minimized. Load-handling equipment in the MST is operated only during shutdown conditions. During this condition, systems located in the tunnel are isolated and not required to be operable. In addition, isolation valves inside the tunnel have redundancy with valves inside containment.

Therefore, a dropped load in the MST could damage components inside the tunnel, but the redundant components inside the containment will maintain the safety function. Finally, regarding the refueling floor, the reactor building crane meets the single-failure-proof design requirements of NUREG-0554 and is interlocked to prevent movement over the spent fuel storage pool. The SSAR incorporates the above information. The staff reviewed the information and concluded that the design features for the load-handling equipment in all certified buildings ensure that both new and spent fuel, as well as safety-related equipment, will not be damaged to such an extent as to inhibit or prevent safety functions. This is acceptable and resolved DFSER Open Item 9.1.5-2.

3. In the DSER (SECY-91-235), the staff noted that terms such as "refueling bridge crane," "automatic refueling machine," and "spent fuel handling crane" were used in the SSAR to represent possibly the same load-handling device. Also, it was not clear whether the ABWR used a jib crane and whether the fuel-handling platform was different from the refueling platform. Further, SSAR Tables 3.2-1 and 9.1-2 gave different seismic classifications for the refueling platform crane (automatic refueling machine) and the jib crane.

   In the DFSER, the staff stated that many of the discrepancies had been corrected. However, neither the jib crane nor the refueling platform crane were referred to in the text of SSAR Section 9.1, although both were listed in SSAR Table 3.2-1. While the term "Refueling Bridge," was used only in Table 3.2-1, the term "Refueling Platform," was used throughout Section 9.1. The text of Section 9.1 appeared to be internally consistent. References to a refueling platform consistently referred to the fuel-handling equipment. The staff requested GE to clarify the remaining discrepancies in the nomenclature in SSAR Table 3.2-1 and Section 9.1. The staff reclassified this part of DSER Open Item 48 as Open Item 9.1.5-3 in the DFSER. Subsequently, GE provided clarifying information which corrected the remaining discrepancies and changed "refueling platform" to "refueling machine." This is acceptable and resolved DFSER Open Item 9.1.5-3.

4. The staff noted in the DSER (SECY-91-235) that SSAR Section 9.1.5.2.1 implies that the load-handling equipment for steam tunnel servicing (MSIVs and SRVs) is housed in the reactor building, while SSAR Table 3.2-1 shows the equipment to be located in "any other location." The staff requested GE to correct the location of the subject equipment, as appropriate and, if the location is "any other location," to state why housing the equipment in a non seismic Category I structure is acceptable.

   In response, SSAR Section 9.1.5.2.2.3, Amendment 17, stated that the MST servicing equipment is a permanently installed monorail system. Note c in SSAR Table 3.2-1, which indicated the MST tunnel servicing cranes and hoists are in "any other location," is interpreted to mean that the monorail is not in any of the locations specifically identified in Note c, and does not contradict the statement in SSAR Section 9.1.5. The clarification shows that the equipment is housed in a seismic Category I structure. This is acceptable and resolved this part of DSER Open Item 48.

5. The staff stated in the DSER (SECY-91-235) that SSAR Table 3.2-1 showed load-handling equipment for special service rooms being housed in the non seismic Category I radwaste and turbine buildings, and requested GE to clarify why such locations are acceptable.

   In response, GE indicated that the load-handling equipment in the turbine and radwaste buildings will handle only the equipment in those structures. In the DFSER, the staff requested GE to incorporate this information into the SSAR by reclassifying it as Confirmatory Item 9.1.5-1 in the DFSER. The SSAR adds information in Sections 9.1.5.3 and 9.1.6.6 clarifying that all load-handling devices, including those in the radwaste building and turbine buildings, must be designed and procedures must be used which ensure that the loads they carry will not damage more than one division of a safety-related system. Further, the load-handling devices in the turbine and radwaste buildings are intended to handle only equipment in these buildings and that the loads will not be moved over safety-related equipment. Therefore, failure of this equipment or dropping of any loads moved by this equipment will not adversely affect safety-related SSCs. The staff finds this additional clarification acceptable. Therefore, DFSER Confirmatory Item 9.1.5-1 is resolved.

The OHLHS meets Positions C.1 and C.2 of RG 1.29, and Positions C.1 and C.6 of RG 1.13 for the spent fuel storage facility and therefore, complies with the requirements of GDC 2 with regard to protection of the system from the effects of adverse natural phenomena.

The spent fuel cask pool is separated from the spent fuel storage pool by a water-tight gate. Redundant safety interlocks and limit switches in the reactor building crane prevent transport of any heavy load, including the spent fuel cask, over the spent fuel pool. Administrative control and a coverage area prevent transport of any heavy load over the new fuel storage vault. A dropped cask cannot, therefore, result in any fuel damage. Additionally, as mentioned above, the reactor building crane and the refueling platform crane are single-failure-proof cranes. The staff in the DSER (SECY-91-235) identified several concerns, as part of Open Item 48, pertaining to compliance of the OHLHS with the requirements of GDC 4 as it relates to protection of safety-related equipment from the effects of internally-generated missiles (i.e., dropped loads) during load-handling operations:

1. The staff stated in the DSER (SECY-91-235) that SSAR Section 9.1.4.3 incorrectly stated that the spent fuel-handling crane's lifting height for the spent fuel cask is limited to 30 feet on the operating floor, and that this should be corrected to refer to the reactor building crane.

GE subsequently modified the SSAR to eliminate inconsistent terms. This modification resolved this part of DSER Open Item 48.

2. The staff noted in the DSER (SECY-91-235) that SSAR Amendment 7 (response to RAI 410.43) gives contradictory statements: "While carrying heavy loads, such as the spent fuel cask, the reactor building crane is prohibited from moving the heavy load over the spent fuel pool," and "No other heavy loads other than the spent fuel cask, need to be carried above the top of the spent fuel storage pool." The staff requested GE to correct this as appropriate.

   In response, GE committed to revise the response to RAI 410.43 to eliminate the contradictory statements regarding movement of the spent fuel cask. Incorporation of the revisions to the response to RAI 410.43 was identified as Confirmatory Item 9.1.5-2 in the DFSER. SSAR Section 9.1.5.2.1 states that the reactor building crane will be interlocked to prevent movement over the spent fuel pool while carrying heavy loads. This is acceptable and resolved DFSER Confirmatory Item 9.1.5-2.

3. The staff noted in the DSER (SECY-91-235), as part of DSER Open Item 48, that GE had not identified how safety-related equipment would be protected during all heavy load-handling operations involving non-single-failure-proof load lifting devices (e.g., MSIVs, SRVs, RIP motors and hoists, load-handling operations in control, radwaste, and turbine buildings).

   In response, SSAR Amendment 20 identified the criteria to assess the safety impact of the failure of the load-handling devices. The lower drywell load-lifting device does not carry loads over safety-related equipment. The failure of the upper drywell load-lifting device, used only during maintenance at shutdown, can affect only the one division of an ECCS. Safe-shutdown conditions can be maintained with the failure of the one ECCS division. The MST equipment is used only during shutdown, and its failure will not impact any safety-related equipment needed to maintain safe-shutdown conditions. Load-handling equipment in the turbine and radwaste buildings is not used near safety-related equipment because such equipment is restricted to use in these structures. However, the use of heavy load-handling equipment in the control building was not addressed. As discussed in the DFSER, all items identified in this portion of DSER Open Item 48 are resolved with the information provided by SSAR Amendment 20 except for the concern regarding the control building. Identification of how safety-related equipment would be protected during all heavy load-handling operations involving non-single-failure-proof load lifting devices in the control building was identified as Open Item 9.1.5-4 in the DFSER. Subsequently, GE provided clarifying information in SSAR Sections 9.1.5.3, 9.1.5.4, 9.1.5.5, and 9.1.6.6. The design of all heavy load-handling equipment will meet the guidelines of NUREG-0612, including the guidelines of NUREG-0554 regarding single-failure-proof cranes. Based on the additional information provided in these SSAR subsections, the staff concludes that safety-related equipment will be adequately protected during all heavy load-handling operations. Therefore, DFSER Open Item 9.1.5-4 is resolved.

Based on the above information, the staff concludes that the OHLHS design will meet the guidelines of Positions C.3 and C.5 of RG 1.13, with respect to protection of the spent fuel storage facility from the effects of internally-generated missiles and for the safe handling and storage of fuel, and further concludes that safety-related equipment is protected from the effects of internally-generated missiles during load-handling operations. Therefore, the OHLHS design for the ABWR will comply with GDC 4 as it relates to protection of safety-related equipment from internally-generated missiles.

The requirements of GDC 5 regarding the sharing of SSCs do not apply because the ABWR is designed as a single-unit facility. An application for a multi-unit facility will require review of the design for compliance with GDC 5.

As mentioned above, the reactor building and refueling platform cranes are single-failure-proof cranes. SSAR Section 9.1.5.5 further states that all the cranes, hoists, and related lifting devices for handling heavy loads either satisfy the single-failure-proof guidelines of NUREG-0612, Section 5.1.6, including NUREG-0554, or evaluations are made to demonstrate compliance with guidelines of NUREG-0612, Section 5.1, including Sections 5.1.4 and 5.1.5. GE stated that the applicable components of the OHLHS will comply with the guidelines of the following industry standards: American National Standards Institute (ANSI) N14.6, ANSI B30.9, ANSI B30.10, ANSI B30.2, ANSI B30.16, ANSI B30.11, and Crane Manufacturers Association of America (CMAA) Specification 70. NUREG-0554 and NUREG-0612 recommend the standards specified in ANSI N14.6, ANSI B30.2, ANSI B30.9, and CMAA 70 for the design and performance of a nuclear power plant OHLHS. Additionally, GE states that the design of special lifting devices and slings will comply with the guidelines of NUREG-0612, Sections 5.1.1(4) and 5.1.1(5).

GE submitted a general description of the inspection, operation, maintenance, service, and test requirements for the OHLHS equipment. The COL applicant is responsible for supplying operating, maintenance, and test procedures and instruction manuals that will comply with NUREG-0612, Sections 5.1.1(2) and 5.1.1(6) for heavy load-handling equipment components (e.g., cranes, hoists, refueling platform). In accordance with the requirements of 10 CFR Part 50, Appendix B, the COL applicant will subject the OHLHS equipment to qualification load and performance tests, dimensional inspection, and nondestructive examination before the equipment is accepted by the applicant's quality assurance group. The applicant will also ensure that lifting components are appropriately inspected and tested before shipment, after receipt at the site, before use, and at periodic intervals. The tests will be conducted in accordance with the requirements of ANSI B30.2 and NUREG-0612, Section 5.1.1(6). For each item of equipment requiring servicing, the applicant will develop an interface control diagram (ICD) delineating the space around the equipment required for servicing. The ICD will include pull space for internal parts and access space for tools, handling equipment, and alignment requirements. The ICD will specify the weights of large removable parts, show the locations of their centers of gravity, and describe installed lifting accommodations such as eyes and trunnions. The COL applicant will develop safe load paths and routing plans for each heavy load to be handled, which will show, among other things, frequency of transportation and usage of the route. The safe load paths/routing will comply with NUREG-0612, Section 5.1.1(1), guidelines. These items are DFSER COL Action Item 9.1.5-1.

The staff stated in the DSER (SECY-91-235) that the information provided in SSAR Section 9.1.5 (also Section 9.1.4, which is cross-referenced in Section 9.1.5), lacked details with respect to OHLHS compliance with the applicable guidelines of NUREG-0612. The staff's concerns, identified as DSER Open Item 49 (Section 1.8 of the DSER grouped four concerns of DSER Section 9.1.5 as Open Item 49.) in this regard, are as follows:

1. In the DSER (SECY-91-235), the staff noted that GE had not identified all the hoists for the reactor building, refueling platform, and steam tunnel cranes and any other crane, nor the load-handling capacity for all the hoists, including the monorail hoist, equipment hatchway hoist, equipment platform/lower drywell RIP hoist. In the DFSER, the staff further noted that GE had not provided this information, therefore, this part of DSER Open Item 49 was reclassified as Open Item 9.1.5-5 in the DFSER. Upon further review, the staff determined that the issue can be addressed by COL actions. The SSAR, Section 9.1.6.6, states that the COL applicant will provide this information. This is acceptable and resolved DFSER Open Item 9.1.5-5.

2. In the DSER (SECY-91-235), the staff noted that GE had not identified which of the hoists and related lifting devices associated with all heavy load-handling systems meet single-failure-proof criteria and which of these components meet the alternative criteria specified in applicable sections of NUREG-0612. The staff indicated that GE should identify the specific alternative criterion applicable to the chosen heavy load- handling device, or if neither of the above criteria is applicable for some heavy load-handling devices (because they do not affect required safety-related equipment). GE should include justification for such a conclusion for the applicable devices.

   As previously noted, SSAR Amendment 20 identified the criteria to be applied to each heavy load-handling system except for those in the control building. With the exception of this area as discussed separately in item (4) below, this information resolved the applicable part of DSER Open Item 49.

3. In the DSER (SECY-91-235), the staff noted that GE did not identify the specific limit and safety devices (e.g., interlocks and limit switches) provided for automatic and manual operation of all the heavy load-handling equipment (within ABWR scope) under both normal and emergency conditions. Further, the staff noted that GE did not submit a failure modes and effects analysis for the OHLHS instrument and control system to demonstrate that the control system will adequately limit the loads or limit the crane load movement, assuming a single failure, without affecting the function of safety-related equipment or causing the release of radioactivity.

Section 9.1.5.5 of SSAR Amendment 20 states that limit switches and safety interlocks are provided to prevent transporting heavy loads over spent fuel. However, the amendment does not include the level of detail requested by the staff in the DSER (SECY-91-235) and does not specify this information as the action of the COL applicant. In the DFSER, the staff stated that GE should either submit this information or designate this as the action of the COL applicant. The staff reclassified this part of DSER Open Item 49 as DFSER Open Item 9.1.5-6. SSAR Section 9.1.6.6 states that the COL applicant will submit this information. This is acceptable and DFSER Open Item 9.1.5-6 is resolved.

4. The staff stated in the DSER (SECY-91-235) that GE had not indicated the heavy load-handling operations which may involve areas other than the reactor building which are within ABWR scope (e.g., control building). Specifically, GE had not identified the heavy load-handling equipment, its handling capacity, the load required to be handled, when such operations have to be performed, and how safety-related equipment is protected during such operations.

   In the DFSER, the staff stated that GE did not submit the information requested in the DSER (SECY-91-235) and did not state guidelines for load- lifting systems outside the scope of the ABWR (e.g., the reactor service water (RSW) pump house). The staff further noted that this information should include the identification of the load-handling systems, the system characteristics, and the ability to meet the guidelines of NUREG-0612. The staff reclassified as DFSER Open Item 9.1.5-7 the submission of the previously requested information and the identification of the requirements for load lifting systems outside the ABWR scope. Upon further evaluation, the staff determined that the issue can be resolved by COL actions. SSAR Section 9.1.6.6 clarifies that the COL applicant will submit design and operational details for all load-handling devices both in and out of the design scope. The staff concludes that this clarification will ensure that the design and operational details associated with the movement of heavy loads will be provided. This is acceptable and resolved DFSER Open Item 9.1.5-7.

The staff finds that the OHLHS design and commitments meet the guidelines of NUREG-0612 and requirements of GDC 61 for the safe handling and storage of fuel.

In the DFSER, the staff requested GE to submit adequate design description and the ITAAC for the OHLHS. This was DFSER Open Item 9.1.5-8. GE submitted a revised set of design description 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 the above, DFSER Open Item 9.1.5-8 is resolved.

The staff concludes that the design of the OHLHS for the ABWR conforms with the requirements of GDC 2, 4, 5, and 61 for protection against natural phenomena, protection of safety-related equipment from the effects of internal missiles, the sharing of important systems, and the safe handling and storage of the fuel. The design also conforms with the guidelines of Positions C.1, C.3, C.5, and C.6 of RG 1.13 and Positions C.1 and C.2 of RG 1.29. The staff further concludes that the OHLHS equipment within the ABWR scope conforms with the guidelines of NUREG-0612. However, since a major portion of NUREG-0612 is outside the scope of the ABWR standard design (the part which deals with crane operation, operator training, operating and maintenance procedures, and physical marking of safe load paths), the staff stated in the DFSER that the COL applicant will provide the specifics of compliance and final implementation of NUREG-0612 guidelines on a plant-specific basis. This was DFSER COL Action Item 9.1.5-2. SSAR Section 9.1.6 states that the COL applicant will submit this information.

The OHLHS includes all components and equipment used to handle all heavy loads at the plant site over the lifetime of the facility. Based on the review of the SSAR design criteria and design bases for the OHLHS, the staff concludes that the design of the OHLHS, as discussed in detail above, is in conformance with the guidelines of SRP Section 9.1.5 and the Commission's regulations as set forth in GDC 2, 4, 5, and 61, and is acceptable.

9.2 Water Systems

For some systems listed in the SRP, the functions in the ABWR design will be performed by one or more different systems. For example, the functions of the closed cooling water system also will be performed by the RCW system, the heating, ventilation, and air conditioning normal cooling water (HNCW) system, the HVAC emergency cooling water (HECW) system, and the turbine building cooling water (TCW) system.

9.2.1 Station Service Water System

See Section 9.2.15, "Reactor Service Water."

9.2.2 Reactor Auxiliary Cooling Water System

See Sections 9.2.11 "Reactor Building Cooling Water System," and 9.2.12, "HVAC Normal Cooling Water System" of this report.

9.2.3 Demineralized Water Makeup System

See Sections 9.2.8, "Makeup Water System (Preparation)," 9.2.9, "Makeup Water System (Condensate)," and 9.2.10 "Makeup Water System (Purified)" of this report.

9.2.4 Potable and Sanitary Water System

The staff reviewed the design requirements for the potable and sanitary water (PSW) system in accordance with SRP Section 9.2.4. Staff's acceptance of the design is based on meeting the requirements of GDC 60 as it relates to preventing the release of liquid effluents containing radioactive material into the PSW system. Compliance is met if there are no interconnections between the PSW system and any potentially contaminated systems and if the PSW system is protected by an air gap, if necessary.

GE stated that only the portion of the PSW system located in the buildings of the certified standard plant is within the design scope, while the remainder of the system is not. The out-of-scope portion of the system will be designed by the applicant referencing the ABWR design. The DFSER noted that the staff's review was in progress. This was DFSER Open Item 9.2.4-1. SSAR Section 9.2.4 includes a conceptual design and interface requirements for the out-of-scope portion of the system. The staff reviewed the conceptual design and interface requirements and concluded that sufficient guidance has been provided to allow an applicant referencing the ABWR to design the out-of-scope portion to meet all applicable regulatory requirements. This is acceptable and resolved DFSER Open Item 9.2.4-1.

The PSW system is a non-safety-related system designed to provide a minimum of approximately 45 m^3/hr (200 gpm) of potable water during peak demand periods. The system is composed of a potable water subsystem, a sanitary drainage subsystem, and a sewage treatment subsystem.

Water is supplied to the potable water subsystem from the makeup water system (preparation) (MWP) to a potable water storage tank. The water is chemically treated, pressurized, heated, and distributed throughout the plant. According to SSAR Table 3.2-1, the system serves all areas of the plant within the ABWR scope except the primary containment and the MST.

Liquid wastes (including those from the nonradioactive drain system) are collected in the sanitary drainage subsystem and sent to the sewage treatment subsystem. The sewage treatment subsystem uses the activated sludge biological treatment process. The subsystem contains a comminutor, aeration tanks, aerobic digesters, air blowers, clarifiers, a froth spray pump, a hypochlorite pump, and associated equipment. This subsystem can be operated in the extended aeration mode or the contact stabilization mode (used during high demand periods (e.g., refueling outages) when additional personnel are on site).

The PSW system does not include any connections to systems which may contain radioactive material. Where necessary, additional protection is provided through the use of air gaps.

The system contains adequate controls, instrumentation, and alarms to ensure adequate operation during normal conditions and to alert operators to abnormal conditions. Drainage piping will be hydrostatically tested.

In the DFSER, the staff requested GE to submit adequate design description, ITAAC, and interface requirements for the PSW system. This was DFSER Open Item 9.2.4-2. Subsequently, GE provided a revised set of design description, ITAAC, and interface requirements. The adequacy and acceptability of the design description, the ITAAC, and the interface requirements are evaluated in Section 14.3 of this report. On the basis of the above, DFSER Open Item 9.2.4-2 is resolved.

The PSW system includes all components and piping from the supply connection to the municipal or other water source to all points of discharge to sewage facilities or other plant systems. Based on the review above, the staff determined that adequate design provisions have been made to prevent the inadvertent contamination of the system with radioactive material. The staff concludes that the design of the PSW system meets the guidelines of SRP Section 9.2.4, and since there are no interconnections between the PSW system and any contamination systems it also meets the requirements of GDC 60, and is acceptable.

9.2.5 Ultimate Heat Sink

The staff reviewed the design requirements for the ultimate heat sink (UHS) in accordance with SRP Section 9.2.5. Staff acceptance of a UHS 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 the capability of shared SSCs to perform required safety functions; 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, 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 inservice inspection and operational functional testing, respectively, of safety-related systems and components.

The SSAR states that the design of the UHS is outside the scope of the ABWR design. The SSAR includes a conceptual design and interface requirements, as required by 10 CFR Part 52, to allow an applicant referencing the ABWR design to provide a plant-specific UHS design that is capable of dissipating reactor decay heat and essential cooling loads after a normal reactor shutdown or a shutdown after an accident, including a loss-of-coolant accident (LOCA). The UHS must be designed to accept the heat loads of the RSW system (Section 9.2.15 of this report), which in turn accepts the heat loads of the RCW system (Section 9.2.11 of this report) under both normal and accident conditions.

The conceptual design for the UHS consists of a seismic Category I spray pond from which the RSW system will receive cooling water. The spray pond will be excavated below grade and contain adequate water volume to supply cooling for 30 days under design-basis conditions. Six spray networks, three functioning during normal operation, will cool the RSW return water. The spray nozzles may be bypassed during cold weather conditions allowing RSW return water to be returned directly to the pond. RSW pumps will be located in the spray pond pump structure, each pump in its own bay. The pond will also have a seismic Category I overflow weir to accommodate normal level fluctuations. The spray pond will receive makeup from a power cycle heat sink makeup line.

The structures and components of the UHS will be designed to seismic Category I requirements and will be designed to withstand the effects of natural phenomena such as floods, earthquakes, and tornados. The system design should ensure that the UHS can perform its safety function given the occurrence of any of the following: (1) the most severe natural phenomena appropriate with site conditions, (2) site-related events that have historically occurred, (3) reasonable combinations of natural phenomena and site-related events, and (4) a single failure of man-made structures. These interface requirements will allow an applicant to design a UHS that meets the requirements of GDC 2 regarding protection from natural phenomena and RG 1.27, "Ultimate Heat Sink for Nuclear Power Plants," Revision 2 and RG 1.29, "Seismic Design Classification," Revision 3.

The UHS design will ensure that safety-related portions of the system are protected from spraying, steam impingement, pipe whip, jet forces, missiles, fire, internal flooding, and the effects of failure of non seismic Category I equipment. Thus, the interface requirements meet the requirements of GDC 4 for protection of systems from the environmental and dynamic effects associated with equipment failures.

The requirements of GDC 5 for the sharing of SSCs do not apply because the ABWR is designed as a single-unit facility. An application for a multi-unit facility will require review of the design for compliance with GDC 5.

The UHS must have the capability to transfer heat loads from safety-related structures and systems during normal and accident conditions, and suitable redundancy so that it will function given a single-failure coincident with a LOOP. It must also be capable of isolating portions of the system in such a way as to not interfere with the system's safety function. The UHS will provide cooling capability for 30 days.

The requirements of RG 1.72, "Spray Pond Piping Made from Fiberglass-Reinforced Thermosetting Resin," Revision 2, apply to the design of a spray pond as an UHS. The staff established Open Item 50 in the DSER because these requirements were not referenced in the SSAR submitted before the staff completed the DSER (SECY-91-235). SSAR Section 9.2.5.8 states that the COL applicant will submit information to show that all applicable requirements of RG 1.72 are met. This is acceptable and resolved Open Item 50.

The SSAR requires the COL applicant to prepare a preoperational test program in accordance with the requirements of Chapter 14 and shall perform inspections and tests during normal operations. Based on these commitments, the staff believes that the applicant referencing the ABWR design can perform inspections and tests which meet the requirements of GDC 45 and 46, respectively.

In the DFSER, the staff requested GE to include interface requirements for the UHS system in the Tier 1 information. This was DFSER Open Item 9.2.5-1. The adequacy and acceptability of the Tier 1 information is evaluated in Section 14.3 of this report. On the basis of the above, DFSER Open Item 9.2.5-1 is resolved.

The staff found the conceptual design and interface requirements for the UHS to be acceptable. They include 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, sharing of SSCs and heat transfer, redundancy, isolation capabilities, and ability to transfer heat loads. The system must be designed by the COL applicants to allow periodic inspections and tests and must, therefore, meet the requirements of GDC 45 and 46 with respect to inspection and testing requirements for cooling water systems. The interface requirement in the ABWR SSAR should enable an applicant referencing the ABWR to design an acceptable UHS.

9.2.6 Condensate Storage Facility

See Section 9.2.9, "Makeup Water System (Condensate)," of this report.

9.2.7 Chilled Water Systems

See Sections 9.2.12, "HVAC Normal Cooling Water System," and 9.2.13, "HVAC Emergency Cooling Water System," of this report.

9.2.8 Makeup Water System (Preparation)

The staff reviewed the requirements for the makeup water system (preparation) (MWP) in accordance with SRP Section 9.2.3. Staff acceptance of the MWP system design is based on meeting GDC 2 as it relates to safety-related portions of the system being capable of withstanding the effects of natural phenomena, GDC 5 as it relates to the capability of shared SSCs to perform required safety function, and position C.2 of RG 1.29, Revision 3 (September 1978) relative to protection of safety-related SSCs from failure of the system following an SSE.

The MWP system is not within the certified design scope of the ABWR. As required by 10 CFR Part 52, the SSAR provides a conceptual design and interface requirements for the MWP system. This design information and commitments provide sufficient detail to ensure that the system can be designed and built to meet the applicable regulatory requirements.

The MWP system is a non-safety-related system that supplies water for the makeup water system (purified) distribution system (MUWP) and the portable and sanitary water system (PSW). The MWP system design includes a requirement for two system divisions, each capable of producing approximately 45 m^3/hr (200 gpm) of demineralized water and each with a storage capacity of at least 760 m^3 (200,000 gallons).

The system will consist of both a permanently installed water treatment system and a mobile water treatment system. The permanently installed system will consist of a well, filters, reverse osmosis modules, demineralizers, storage tanks, and pumps to treat and store well water for use as demineralized water. The mobile water treatment system will be used before the permanent system is operable and if needed to supplement the permanent system. Therefore, the system does not require a seismic Category I makeup source.

The MWP system will be located in a building that does not contain any safety-related components, systems, or structures. Any failure of the system (including failures that could cause flooding) will not result in the failure of any safety-related SSCs and consequently will not adversely affect any safety-related function of the ABWR design. Therefore, the staff concludes that the conceptual design meets the requirements of GDC 2 with regard to protection of safety-related equipment from natural phenomena, and Position C.2 of RG 1.29, Revision 3 (September 1978).

The requirements of GDC 5 for the sharing of SSCs do not apply because the ABWR is designed as a single-unit facility. An application for a multi-unit facility will require review of the design for compliance with GDC 5.

In the DFSER, the staff requested GE to submit adequate design description, ITAAC, and interface requirements for the MWP system. This was DFSER Open Item 9.2.8-1. GE submitted a revised set of design description, ITAAC and interface requirements. The adequacy and acceptability of the design description, ITAAC, and interface requirements are evaluated in Section 14.3 of this report. On the basis of the above, DFSER Open Item 9.2.8-1 is resolved.

Based on the SSAR commitments discussed above, the staff concludes that GE has described an acceptable conceptual design and interface requirements sufficient to ensure that an applicant referencing the ABWR can design a MWP system that will meet the requirements of Position C.2 of RG 1.29, Revision 3 (September 1978); GDC 2 with regard to the ability of the non-safety-related portions of the system to withstand the effects of natural phenomena without affecting safety-related systems; and GDC 5 with regard to the sharing of SSCs important to safety. The interface requirements in the SSAR will enable an applicant referencing the ABWR to design an acceptable MWP system in compliance with the guidelines of SRP 9.2.3. The conceptual design is acceptable.

9.2.9 Makeup Water System (Condensate)

The staff reviewed the MUWC system (the condensate storage and transfer system) in accordance with SRP Section 9.2.6. Staff acceptance of the MUWC system design is based on meeting the requirements of GDC 2 as it relates to the system being able to withstand the effects of natural phenomena; GDC 5 as it relates to the capability of shared systems and components to perform required safety functions; GDC 44 as it relates to redundancy and isolability of components as well as the capability to provide makeup to safety-related systems; and GDC 45 and 46 as they relate to inservice inspection and testing, respectively, of safety-related systems and components. The design of the entire MUWC system is within the scope of the ABWR certified design.

The MUWC system will supply condensate quality water and will include a piping distribution system from the source to the components that require this water during normal and emergency operations. MUWC system water will be stored in the condensate storage tank (CST) with a capacity of at least 2,100,000 L (555,000 gallons). The CST is located outdoors adjacent to the turbine building. The CST will reserve approximately 570,000 L (150,000 gallons) of this capacity to remove decay heat for up to 8 hours after a station blackout (SBO). Water will be supplied to the CST from the MUWP system. Level-sensing instrumentation and transmitters will automatically switch over the HPCF and reactor core isolation cooling (RCIC) pumps from the preferred CST to the safety-related suppression pool when the CST water level is low. The tank will also supply water for the control rod drive (CRD) supply pump (the preferred water source being the condensate treatment system) and the SPCU pump, which will be used for fuel pool makeup, when required. The MUWC system will normally supply water through three system transfer pumps for charging, flushing, pump sealing, surveillance testing, room decontamination, and makeup, as appropriate, for systems including RHR, HPCF, RCIC, fuel pool skimmer surge tanks, and the main condenser hotwell.

The MUWC system is not safety-related, except as noted below, because it will not affect the integrity of the reactor coolant system pressure boundary, prevent achieving and maintaining safe shutdown, or affect the capability to prevent or mitigate the consequences of accidents, which could result in unacceptable offsite radiological exposures. Therefore, GDC 44, 45, and 46 do not apply to the non-safety-related portions of the system.

The safety-related portions of the system discussed below meet the redundancy, inservice inspection, and inservice testing requirements for safety-related equipment and therefore meet the requirements of GDC 44, 45, and 46.

As stated in SSAR Table 3.2-1, for the MUWC, RCIC, and HPCF systems, certain parts of MUWC system piping, including supports and valves, will be designed to seismic Category I and QG B standards and will be located in seismic Category I, flood-protected and tornado-missile-protected structures. The safety-related portions include those forming part of the containment boundary and those system piping portions that interface with the safety-related RCIC and HPCF systems, up to and including the isolation/suction valves for the systems from the MUWC. The non-safety-related portions of the system that could affect any SSCs important to safety if they fail during a seismic event are designed to ensure their integrity under seismic loading conditions resulting from an SSE. The level instrumentation in the MUWC system that facilitate the automatic switchover of the HPCF and RCIC pumps suction from the CST to the suppression pool and their power supplies are safety-related. The SPCU pumps will be switched over manually. The staff concludes that the safety-related and non-safety-related portions of this system meet Positions C.1 and C.2, respectively, of RG 1.29.

The staff stated in the DSER (SECY-91-235) that GE did not submit an analysis for flooding that could result from a possible failure of the non-safety-related portion of the MUWC system, including the CST, and how safety-related SSCs are protected from such flooding. This was DSER Open Item 51. SSAR Section 3.4.1.1.1 reports the results of a flood analysis and describes the safety features that will protect safety-related SSCs from external floods (including the failure of the CST) as discussed in Section 3.4.1 of this report. The instrumentation used to initiate the automatic switchover of HPCF and RCIC suction from the CST to the suppression pool is safety-related and is housed in a safety-grade standpipe in the reactor building, a seismic Category I structure designed to withstand tornadic winds and missiles, flooding, hurricanes, and an SSE. On the basis of this information, the staff concludes that the system meets the requirements of GDC 2 as it relates to the protection of safety-related portions of the system from the effects of natural phenomena. Therefore, DSER Open Item 51 is resolved.

The ABWR is designed as a single-unit facility. Therefore, the requirements of GDC 5 regarding the sharing of SSCs are not applicable. An application for a multi-unit facility will require review of the design for compliance with GDC 5.

Normal alignment for removal of decay heat is with the CST. Water for RCIC operation is taken from either the CST or the suppression pool as described in the emergency procedure guidelines of SSAR Appendix 18A. The volume of water in these two sources is sufficient to permit core cooling during SBO for a duration of 8 hours. The switchover from the CST to the suppression pool (or the reverse) is performed using station dc power and is not dependent upon either offsite ac power and systems or onsite emergency power systems. Therefore, the MUWC system complies with the applicable guidance of RG 1.155, "Station Blackout," Revision 0 (August 1988).

MUWC is demonstrated to be operable by normal system operation. Those portions of the system normally closed to flow can be tested to ensure system operability and integrity.

Based on this information, the staff concludes that MUWC meets the applicable requirements of GDC 44, 45, and 46.

In the DFSER, the staff requested GE to provide adequate design description and the ITAAC relating to the MUWC system. This was DFSER Open Item 9.2.9-1. Subsequently, GE provided a revised set of design description 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 the above, DFSER Open Item 9.2.9-1 is resolved.

The MUWC system includes all components and piping associated with the system to the points of connection with other systems. The staff determined that the SSAR design criteria and bases for the MUWC and the requirements for a sufficient water supply to safety-related systems during normal and emergency conditions are acceptable.

The staff concludes, as discussed above, that the MUWC system design complies with Positions C.1 and C.2 of RG 1.29, GDC 2, 5, 44, 45, 46, and, therefore, with the applicable acceptance criteria of SRP Section 9.2.6 and is acceptable.

9.2.10 Makeup Water System (Purified) Distribution System

The staff reviewed the MUWP system in accordance with SRP Section 9.2.3. The design is acceptable if it complies with the requirements of GDC 2 as it relates to protection of the safety-related portions of the system from natural phenomena, and GDC 5 as it relates to the capability of shared SSCs to perform required safety functions.

The design of the MUWP system is fully within the scope of the ABWR design. The staff evaluated the ABWR capability to prepare, store, and transport demineralized water to this system as described in Section 9.2.8 of this report. The MUWP system is not safety-related, except as noted below.

The MUWP system will supply demineralized makeup quality water and will include a piping distribution system from the source to the components that require this water. The MUWP system will receive makeup water from the MWP system and will normally supply demineralized water for flushing, sealing, surveillance testing, area decontamination, sampling, and makeup as appropriate. The MUWP system will supply makeup water to systems such as the MUWC system, the RCW system, the TCW system, the diesel generator cooling water (DGCW) system, the liquid waste management system (LWMS), the standby liquid control system (SLCS), and other plant auxiliary systems. Protection from flooding for safety-related SSCs is discussed in Section 3.4.1 of this report.

The MUWP system is not safety-related, except as noted below, because it will not affect the capability of the reactor coolant system pressure boundary, the capability to achieve and maintain safe shutdown, or the capability to prevent or mitigate the consequences of accidents that could result in unacceptable offsite radiological exposures. The MUWP system enters the primary containment through one penetration. The system piping through this penetration has a locked-closed manual valve outside the containment and a check valve inside the containment. The portions of the system penetrating the containment (including these two valves) are designed to seismic Category I, QG B requirements in accordance with Position C.1 of RG 1.29.

Although it is not safety-related, this system is designed to prevent any radioactive contamination of the purified water. SSAR Table 9.2-2a presents chemistry requirements for the purified makeup water.

In the DSER (SECY-91-235), the staff stated, as part of Open Item 52, that it was not clear whether the non-safety-related portions of the system, which upon their failure during a seismic event can adversely impact SSCs important to safety, will be designed to ensure their integrity under seismic loading resulting from an SSE. SSAR Section 9.2.10.1 states that the portions of the MUWP whose failure can impact safety-related SSCs will be designed to ensure their integrity under seismic loading conditions resulting from an SSE. This is acceptable. Therefore, the design of the MUWP system complies with Positions C.1 and C.2 of RG 1.29 for the safety-related and non-safety-related portions of the system, and GDC 2 for protection against natural phenomena. This resolved the applicable part of DSER Open Item 52.

In the DSER (SECY-91-235), the staff also stated, as part of Open Item 52, that it was not clear whether the portions of the MUWP system in buildings other than the reactor building (e.g., turbine building) and the transport of the demineralized water to these buildings were within GE's scope or the scope of the applicant referencing the ABWR design. SSAR Section 9.2.10.1 states that the interfaces between the MUWP system and safety-related systems are located in the control building or reactor building which are seismic Category I, tornado-missile resistant and flood-protected structures. Therefore, the MUWP system complies with Positions C.1 and C.2 of RG 1.29 and with GDC 2 and, therefore, with the applicable acceptance criteria of SRP Section 9.2.3. This is acceptable and the applicable part of DSER Open Item 52 is resolved.

In the DFSER, the staff stated that the applicant referencing the ABWR design should supply the following: testing capability for air-operated valves; adequate pump net positive suction head (NPSH), purified water storage tank overflow/drainage diversion to the radwaste system; material corrosion resistance; adequate distribution piping, valves, instrumentation, and controls; control room instrumentation that indicate the water level in the purified water storage tank; outdoor piping freeze protection; and adequate diking and other means to control spill and leakage from the demineralized water storage tank that will be located outdoors. This was DFSER COL Action Item 9.2.10-1. Upon further review, the staff has determined that the listing of this information as a COL action item is not needed in the SSAR because it will be provided as a normal part of the licensing process. Therefore, COL action item 9.2.10-1 is appropriately deleted from the SSAR.

The requirements of GDC 5 for the sharing of SSCs do not apply because the ABWR is designed as a single-unit facility. An application for a multi-unit facility will require review of the design for compliance with GDC 5.

In the DFSER, the staff requested GE to supply adequate design description and ITAAC for the MUWP system. This was incorrectly identified in the DFSER as Open Item 9.2.10-2. This designation was later corrected to DFSER Open Item 9.2.10-1. Subsequently, GE provided a revised set of design description 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 the above, DFSER Open Item 9.2.10-1 is resolved.

Based on information in the SSAR and that discussed above, the staff concludes that the MUWP system complies with Positions C.1 and C.2 of RG 1.29, GDC 2, and GDC 5, and therefore, with the applicable acceptance criteria of SRP Section 9.2.3 and is acceptable.

9.2.11 Reactor Building Cooling Water System

The staff reviewed the RCW system in accordance with SRP Section 9.2.2. Staff acceptance of the design of the RCW system is based on meeting the requirements of GDC 2 as it relates to the system withstanding the effects of natural phenomena; GDC 4 as it relates to the system withstanding the effects of failed equipment and piping during both normal and accident conditions; GDC 5 as it relates to the capability of shared SSCs to perform required safety functions; 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, providing suitable redundancy for components given a single active component failure, and the capability to isolate part of the system so that the safety function is not compromised; and GDC 45 and 46 as they relate to permitting inservice inspection and testing, respectively, for safety-related equipment.

The requirements of GDC 5 for the sharing of SSCs do not apply because the ABWR is designed as a single-unit facility. An application for a multi-unit facility will require review of the design for compliance with GDC 5.

The function of the RCW system is to remove heat from plant auxiliaries (some of which are required for safe shutdown) during normal operation and after a LOCA. The RCW system is required to operate, with and without preferred ac power available, at normal power, reactor shutdown, hot standby, and after a postulated LOCA has occurred. The RCW system is a closed cooling water system that provides cooling water to the following essential systems and components: RHR and fuel pool cooling heat exchangers; mechanical seals and motor bearings for RHR and HPCF pumps; air conditioning units (ACUs) for pump rooms (RHR, HPCF, FPC, and RCIC) and system rooms (SGTS, containment atmospheric monitoring system (CAMS), and flammability control system (FCS)); jacket water coolers and filtered water and lubricating oil coolers for diesel generators; and HECW system refrigerators. The RCW system supplies cooling water to the non-essential RIP pump motor coolers and motor generator sets, drywell coolers, reactor water cleanup (CUW) pump coolers, instrument air (IA) and service air (SA) system coolers, CUW non-regenerative heat exchangers, CRD pump oil coolers, and other nonessential auxiliary components in the reactor, turbine, and radwaste buildings (e.g., radwaste components, condenser offgas, and reactor building and turbine building sampling coolers).

The RCW system supplies cooling water, which absorbs heat from the plant auxiliaries it serves, and rejects the heat through the RCW system heat exchangers to the reactor service water system. The RSW system, in turn, rejects the heat to an UHS that will be designed by the COL applicants. The GE scope of the RCW system includes all the piping, valves, pumps, heat exchangers, instrumentation, and controls from the RCW system heat exchangers to their loads in the reactor, turbine, and radwaste buildings. GE specified the total heat removal rate, total flow rate, temperature drop, and pressure drop at the RCW system heat exchangers for all modes of operation identified above. These parameters provide the heat removal requirements for the referencing applicant to design the plant-specific UHS system that would be connected to the RSW system.

The RCW system consists of three mechanically and electrically independent divisions, each consisting of its own separate piping (including supply and return headers), two pumps, three heat exchangers, valves, and instrumentation. Each division of the RCW system is powered by a different division of the engineered safety features (ESF) power system. Each division of the RCW system supplies cooling water to the auxiliaries of a separate emergency diesel generator, RHR heat exchanger, RHR pump room ACU, RHR pump motor and seal coolers and HECW refrigerators. Other safety loads and non-essential cooling loads are distributed among the three divisions; two divisions share the loads for systems with redundant components (e.g., HPCF and SGTS). Each division has one isolable train for nonessential loads.

Each division of the RCW system is equipped with a surge tank (shared with the HECW system) which the SSAR states is designed to accommodate 30 days of system design leakage without makeup water (GE response, dated March 7, 1989). Also, the system is designed to detect system leakage by associated level monitors, to provide adequate pressure for pump suction, and to allow for changes in system water volume without significant pressure variations. The system is initially filled with demineralized water from the MUWP system. Each division is further equipped with a chemical addition tank to add chemicals to the RCW system to protect it from corrosion or organic fouling.

The system is protected from water-hammer by high point vents in isolable portions of the system and operational procedures requiring filling and venting of any sections of the system before operation. During a LOCA, nonessential RCW system cooling loads are automatically isolated by the closure of valves except for system cooling loads to IA and SA system coolers, CRD pump oil coolers, and CUW pump coolers; these are isolated by the operator, if necessary. Level switches for the surge tank enable the automatic isolation of nonessential cooling loads in the event of significant system leakage results from piping failures in the non-safety-related portions of the system. One valve on each supply and discharge line, with suitable power and controls from applicable divisional sources, ensures isolation if a single active component fails. GE described the methods for determining whether the system leakage occurs in the nonessential portion of the system as indicated by a falling surge tank level. Radiation monitors located downstream of the RCW system pumps and heat exchangers indicate that radiation has leaked into that division. The ABWR design includes remote manual isolation capability for any division. The two remaining operable divisions will be sufficient to meet the total essential cooling load.

The RCW system consists of safety-related and non-safety-related portions. Portions of the system piping (including valves forming part of the primary containment boundary and other safety-related portions of the system piping up to and including the isolation valves that isolate the system from its non-safety-related portions) are designed to the requirements of seismic Category I, QG B or C, and 10 CFR Part 50, Appendix B. The safety-related portions include the RCW system pumps, heat exchangers, surge tanks, and the division isolation valves. Instrumentation and controls performing safety-related functions (e.g., surge tank level switches) are located in the safety-related portions of the system. Electric modules (e.g., sensors, power supplies, signal processors) and cables performing safety-related functions are all designed to seismic Category I and QG B requirements. Non-safety-related portions of the system that can adversely impact safety-related SSCs if they fail during a seismic event, are designed to ensure their integrity under seismic loads resulting from an SSE. The safety-related portions are located in seismic Category I structures designed to protect against flood and tornado missiles. In the DSER (SECY-91-235), staff requested GE to submit confirmatory information that the safety-related electric modules and safety-related cables are located in seismic Category I, flood-protected, and tornado-protected structures. The SSAR incorporates this information. This is acceptable and resolved the unnumbered DSER (SECY-91-235) confirmatory item. Therefore, the staff concludes that the design of the RCW system complies with GDC 2 with respect to protection from natural phenomena, and meets Positions C.1 and C.2 of RG 1.29 with respect to its seismic requirements for the safety-related and non-safety-related portions.

The SSAR states that both the mechanical equipment and piping and electrical equipment, including instrumentation and controls of the redundant divisions of the RCW system, are sufficiently separated and protected to ensure availability of the needed equipment to shut the reactor down in the event of any of the following occurrences: pipe rupture or equipment-failure-induced flooding, spraying or steam release; pipe whip and jet forces from a postulated nearby high-energy pipeline break; missiles from equipment failure; fire; non seismic Category I equipment failure; or a single active component failure in the system. The amendments included the results of a failure analysis of the RCW system to demonstrate that a single active or applicable passive component failure will not compromise the ability of the RCW system to transfer heat loads from safety-related components to the RSW during all modes of operation. GE submitted design characteristics for RCW system components (e.g., pump design flow rate; heat exchanger heat removal capacity) to show that the system will transfer the expected heat loads to the RSW system under all operating conditions.

The staff stated, as Open Item 53 in DSER (SECY-91-235), the following three concerns:

1. The heat removal design capacity of the RCW system heat exchanger may be inadequate for the heat load required to be removed during suppression pool cooling, when the pool temperature reaches 97 °C (207 °F) after a LOCA. The staff noted that a reactor shutdown at 4 hours after a blowdown to the main condenser may be the bounding case, which may require a greater heat removal rate and, consequently, a higher design capacity than that stated for the heat exchangers (see GE's response to RAI 440.73).

   In the DFSER, the staff reclassified this concern as Open Item 9.2.11-1 because it had not yet been addressed by GE. The staff reviewed the SSAR and found that the RCW heat exchanger capacity design had been modified to ensure that the RCW system will be able to remove the worst anticipated heat loads as stated in SSAR Table 9.2-4. This is acceptable and resolved DFSER Open Item 9.2.11-1.

2. The staff stated in the DSER (SECY-91-235) that the projected heat loads and flow rates for hot standby conditions with a loss of ac power indicate that both RCW pumps and all three heat exchangers in a division are required. The staff also stated that this will also be the case for shutdown at 4 hours.

   Data in SSAR Table 9.2-4 clearly indicate that this equipment is required for successful operation of each division for this mode of operation. This is acceptable and resolved the applicable part of DSER (SECY-91-235) Open Item 53.

3. The staff questioned in the DSER (SECY-91-235) whether the loss of an RCW system division during normal operation would result in plant shutdown or operation at reduced power.

   In response, the SSAR states that the loss of one division of RCW to the drywell coolers will not affect plant operation and the loss of cooling to the RIP coolers will reduce plant power output but will not result in a reactor trip. This clarification is acceptable and resolved the applicable part of DSER Open Item 53.

All three divisions of the RCW system will have at least one RCW system pump operating. This configuration ensures the immediate availability of the RCW system for plant shutdown in the event of a LOCA. A LOOP concurrent with a LOCA will result in a temporary loss of pumping until the automatically sequenced restart of RCW system pumps from the emergency diesel generator loading sequence. A LOCA will result in the automatic isolation of most non-safety-related RCW system loads, the starting of the second RCW system pump, and the placing of the third heat exchanger in each division in service.

The staff concludes that the safety-related portions of the RCW system comply with the requirements of GDC 4, with respect to protection against the dynamic effects of postulated piping failures and internally- and externally-generated missiles, and with GDC 44 for the provisions of a system to transfer heat from SSCs important to safety to an UHS.

As stated in Section 9.2.5 of this report, GE submitted interface requirements for an applicant to design the UHS. The RCW system water quality requirements are established by the MUWP because this system supplies makeup water for the RCW system as discussed in Section 9.2.10 of this report. All three divisions of the RCW system are designed to allow periodic inservice inspection of all the system components. The inspections consist of structural and leak-tightness visual inspection, inspection of the entire system for operability, and inspection of the system components for operability and performance. Testing will be conducted to simulate as closely as possible the entire operational sequence of the RCW system for reactor shutdown and LOCA. The system design incorporates provisions for accessibility to permit inservice inspection as required. The staff finds that the system complies with GDC 45 and 46 for inspection and testing requirements for cooling water systems.

In the DFSER, the staff requested GE to submit adequate design description and the ITAAC for the RCW system. This was DFSER Open Item 9.2.11-2. Subsequently, GE provided a revised set of design description 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 the above, DFSER Open Item 9.2.11-2 is resolved.

The RCW system includes pumps, heat exchangers, valves and piping, surge tanks, makeup piping, and the points of connection or interfaces with other systems. Portions of the RCW system that are necessary for safe shutdown, accident prevention, or accident mitigation are designed to seismic Category I and QG B or C requirements. The staff reviewed the SSAR design criteria, design bases, and safety classification for the RCW system against the requirements for supplying adequate cooling water for the safety-related ECCS components and reactor auxiliary equipment for all conditions of plant operation.

Based on the above the staff concludes that the design of the RCW system is acceptable and meets the requirements of GDC 2, 4, 5, 44, 45, and 46 and the guidelines of SRP Section 9.2.2.

9.2.12 HVAC Normal Cooling Water System

The staff reviewed the HNCW system in accordance with SRP Section 9.2.2. Staff acceptance of the HNCW design is based on meeting GDC 2 as it relates to the system withstanding the effects of natural phenomena; GDC 4 as it relates to the system withstanding the effects of failed equipment and piping during both normal and accident conditions; GDC 5 as it relates to the capability of shared SSCs to perform required safety functions; 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, providing suitable redundancy for components given a single active component failure, and the capability to isolate part of the system so that the safety function is not compromised; and GDC 45 and 46 as they relate to permitting inservice inspection and testing, respectively, for safety-related equipment.

The entire HNCW system is within the scope of the ABWR. The HNCW system is not safety related except for portions of the system that penetrate the primary containment, the portions of the system that are part of the secondary containment boundary, and the associated isolation valves.

The major components of the HNCW system are five 25-percent-capacity chillers (one standby), each with an HNCW pump (one standby), a surge tank (shared with the corresponding division of the TCW system), and the associated piping, valves, and instrumentation. Cooling water to the chiller-condenser is supplied by the TCW system.

The requirements of GDC 5 for the sharing of SSCs do not apply because the ABWR is designed as a single-unit facility. An application for a multi-unit facility will require review of the design for compliance with GDC 5.

The function of the HNCW system is to provide chilled water to the drywell cooler cooling coils and cooling coils of other non-safety-related air conditioners, primarily in the reactor, control, radwaste, and service buildings. The HNCW system is not safety-related because it is not required to ensure (1) integrity of the RCS pressure boundary, (2) capability to achieve and maintain safe shutdown, or (3) the ability to prevent or mitigate offsite radiological exposures during accidents. Therefore, GDC 44, 45, and 46, identified as acceptance criteria in SRP Section 9.2.2 do not apply to the non-safety-related portion of the HNCW system. The HNCW system joins the primary containment through two penetrations: one for the supply line and the other for the return line. The supply line penetration has one motor-operated isolation valve outside the containment and a check (isolation) valve inside the containment. The return line penetration has two motor-operated isolation valves, one inside and one outside the containment. Isolation valves and piping for the primary containment penetrations are safety-related and are designed to seismic Category I, QG B, and 10 CFR Part 50, Appendix B, standards. Piping for penetrations for secondary containment is designed to seismic Category I and 10 CFR Part 50, Appendix B, standards. Based on this information, the staff concludes that the safety-related portions of the HNCW system meet the requirements of GDC 44 regarding the provision for reliable systems for transferring heat loads to a heat sink.

The rest of the HNCW system is not safety-related, as stated above, and is designed to non seismic Category I standards. However, the non-safety-related portions of the system whose failure during a seismic event could affect any structure, system, or component important to safety, are designed to ensure their integrity under seismic loads resulting from an SSE. In the DSER (SECY-91-325), the staff stated that, subject to GE's confirmation that the safety-related portions include the isolation valves for the primary containment penetrations, the design of the HNCW system meets Positions C.1 and C.2 of RG 1.29, as addressed by the SRP Section 9.2.2 acceptance criterion with respect to the seismic requirements for the safety-related and non-safety-related portions of the system. This was DSER Confirmatory Item 6 and DFSER Confirmatory Item 9.2.12-1. SSAR Section 9.2.12.3 clarifies the safety classification of the containment penetrations, which is acceptable. This resolved DSER Confirmatory Item 6 and DFSER Confirmatory Item 9.2.12-1.

By virtue of their location in seismic Category I, tornado-missile-protected and flood-protected structures, the safety-related portions of the system are protected against damage from adverse natural phenomena. Further, as concluded in Section 3.4.1 of this report, all safety-related systems are protected against flooding that may result from system failure. Therefore, the system design complies with GDC 2 with respect to protection of its safety-related portions against natural phenomena and protection of other safety-related systems against the consequences of failure of the non seismic portions of the system, as required by the SRP Section 9.2.2 acceptance criterion.

The major components of the HNCW system are located in the turbine building. Therefore, failure of any of these components will not adversely effect safety-related SSCs. Safety-related portions of the system are located such that the adverse consequences of a pipe or other component failure will not prevent the safety-related portions of the system from performing their safety function. Thus the system design meets the requirements of GDC 4 as it relates to the system's capability to withstand the effects of adverse environmental and dynamic effects.

This system is designed to allow periodic testing and inspection of major components. Appropriate American Society of Heating, Refrigeration, and Air Conditioning (ASHRAE), American Society of Mechanical Engineers (ASME), Tank Equipment Manufacturers Association (TEMA), and Hydraulic Institute (HI) standards are used for all tests. Based on this inspection and test information, the staff concludes that the HNCW systems meets the inspection and testing requirements of GDC 45 and 46, respectively.

Makeup water to the system is supplied by the TCW system surge tank which, in turn, receives water from the MUWP system. The MUWP system and the TCW systems are evaluated in Sections 9.2.10 and 9.2.14 of this report, respectively. The SSAR states design characteristics for the system (e.g., cooling capacity of the chillers, pump design flow rate, chilled water supply temperature) and the heat loads required to be removed from the components served by the system. These characteristics indicate that the system is capable of meeting the cooling water needs of the components it serves during normal plant operation and refueling shutdown. The chiller units are controlled individually by remote manual switches. The containment isolation valves for the system close automatically on a LOCA signal. These valves can also be operated manually by remote means. In the DSER (SECY-91-235), the staff stated, as Open Item 54, a discrepancy regarding the number of HNCW pumps and chillers. The SSAR piping and instrumentation diagram (P&ID) correctly shows five HNCW pumps and associated chillers. This resolved DSER Open Item 54.

In the DFSER, the staff requested GE to provide adequate design description and the ITAAC relating to the HNCW system. This was DFSER Open Item 9.2.12-1. Subsequently, GE provided a revised set of design description 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 the above, DFSER Open Item 9.2.12-1 is resolved.

The HNCW system includes pumps, chillers, valves and piping, surge tanks, makeup piping, and the points of connection or interfaces with other systems. The staff reviewed the SSAR design criteria, design bases, and safety classification for the HNCW system against the requirements for supplying adequate cooling water for auxiliary equipment for all conditions of plant operation. The staff concludes, as discussed above, that the design of the HNCW system is acceptable and meets the applicable requirements of GDC 2, 4, 5, 45, and 46, and the guidelines of SRP Section 9.2.2.

9.2.13 HVAC Emergency Cooling Water System

The staff reviewed the HECW system in accordance with SRP Section 9.2.2. Staff acceptance of the HECW design is based on meeting GDC 2 as it relates to the system withstanding the effects of natural phenomena; GDC 4 as it relates to the system withstanding the effects of failed equipment and piping during both normal and accident conditions; GDC 5 as it relates to the capability of shared SSCS to perform required safety functions; 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, providing suitable redundancy for components given a single active component failure, and the capability to isolate part of the system so that the safety function is not compromised; and GDC 45 and 46 as they relate to permitting inservice inspection and testing, respectively, for safety-related equipment.

The requirements of GDC 5 regarding the sharing of SSCs do not apply because the ABWR is designed as a single-unit facility. An application for a multi-unit facility will require review of the design for compliance with GDC 5.

The HECW is a closed cooling water system whose function is to provide cooling water to the main control room (MCR) air conditioners, reactor building essential electrical equipment room (diesel generator zone) coolers, and control building essential electrical equipment room coolers. The HECW system is required to operate at normal power, reactor shutdown, and after any postulated abnormal reactor conditions, including a LOCA. The HECW system has no primary or secondary containment penetrations. GE states that the entire HECW system is safety-related and is within the scope of the ABWR design. The GE scope of the HECW system includes all piping, valves, pumps, chillers, instrumentation, and controls from the HECW system chillers to their cooling loads.

The HECW system consists of three mechanically and electrically independent and completely redundant divisions. HECW Division "A" serves Division "A" diesel generator (DG) zone coolers and control building essential electrical equipment room coolers. Division "A" consists of one refrigerator and pump, a surge tank (shared with the corresponding division of RCW), piping, valves, and instrumentation. Divisions "B" and "C" serve their respective DG zone coolers, control building essential electrical equipment room coolers, and MCR air conditioners. Each division consists of two refrigerators and two pumps, a surge tank, and separate piping, valves, and instrumentation. Each refrigeration unit includes a condenser, an evaporator, a centrifugal compressor, refrigerant, piping, and package chiller controls. Cooling water is supplied to the condensers by the corresponding RCW system divisions. Each HECW system division is powered by a different division of the ESF power system. The system also has a chemical feed tank to add chemicals to each division to protect the system components from fouling.

The HECW system and the cooling water lines from the RCW system are designed to the requirements of seismic Category I, 10 CFR Part 50, Appendix B, and QG C. Thus, the system meets Position C.1 of RG 1.29 for seismic classification for safety-related systems. The system is located in the control building, a seismic Category I structure that protects against flood and tornado-missiles. Therefore, the system complies with GDC 2 with regard to protection of safety-related systems against adverse natural phenomena.

Each HECW system division is equipped with a surge tank that GE states is designed to accommodate more than 100 days' system leakage without makeup water during an emergency. The surge tank is connected to the MUWP system, which supplies normal makeup water. The tank includes level switches to detect system leakage and to allow makeup water to be supplied to the tank when required. These switches actuate the makeup water supply valves (open or closed on low or high tank water level, respectively) and annunciate control room alarms for high-high or low-low tank water levels.

The design of the HECW system includes sufficient separation and independence for both mechanical and electrical components of the redundant trains and protection for the system to perform its function under all reactor conditions, including a LOCA, loss of normal ac power, or a single active component failure in the system, or any combination of the above. GE performed a failure analysis of the HECW system and presented the results in the SSAR to demonstrate that failure of a single active component, failure of all power to a single Class 1E power system bus, or a failure-of-refrigerator signal will not compromise the ability of the system to perform its function. With the system controls set for automatic operation, the system is automatically initiated whenever the HVAC systems in the control building or diesel generator areas are started. The system can also be manually started from the control room. Interlocks for the chillers automatically start the redundant division whenever the operating division fails (e.g., high temperature of the returned cooling water or inadequate chilled water flow). The system flow switches prevent the chiller from operating unless sufficient water is flowing through both the evaporator and the condenser. The chiller units can be controlled individually from the control room by remote-manual switches. The system includes instrumentation and controls for monitoring and controlling system parameters, such as chilled water flow and temperature, condenser water flow, and evaporator discharge flow and temperature. Since the system is not expected to contain any significant level of radioactivity, it has no radiation monitors. GE has provided the design characteristics for the system components (e.g., capacity of the HECW system refrigeration units, chilled water pump flow rate, chilled water and condenser water supply temperatures). GE also submitted the heat removal and flow requirements for the individual system components. This information indicates that any single division of the system is by itself capable of rejecting the total heat from the components the system serves via the refrigerant to the RCW cooling water under all reactor conditions.

From its initial review, the staff stated in the DSER (SECY-91-235) the following three concerns of DSER Section 9.2.12 as Open Item 55:

1. SSAR Table 9.2-9 did not indicate that a single HECW system pump by itself could deliver the required total chilled water flow rate. The staff noted that the statement in SSAR Section 9.2.13.2 that each division contained two 100-percent capacity pumps was contradictory.

   In response, GE committed to provide information to reflect that Divisions "B" and "C" consist of two 50-percent capacity pumps and that Division "A" consists of one pump capable of meeting the hydraulic requirements of the division, which would be consistent with the information provided in revised SSAR Table 9.2-9. Incorporation of this information into the SSAR was Confirmatory Item 9.2.13-1 in the DFSER. The SSAR states that each HECW pump has a capacity of approximately 950 L/M (250 gpm). This capacity is sufficient to supply 100 percent of the needs of Division "A" and 50 percent of the needs for Divisions "B" and "C," which is acceptable. Therefore, the applicable part of DSER Open Item 55 and DFSER Confirmatory Item 9.2.13-1 are resolved.

2. It was not clear whether the single chemical feed tank provided for the system is safety-related. The staff also noted that GE had not indicated whether the associated isolation valve and its piping are safety-related, or whether any non-safety-related portions of the system are isolated from the safety-related portions of the system if isolation is warranted.

   In response, GE indicated that the only non-safety-related portions of the HECW divisions are the chemical addition tank and the piping from the tank to the safety-related valves which isolate the tank from the safety-related portions of the system. This addressed the applicable part of DSER (SECY-91-235) Open Item 55. In the DFSER, the staff requested that the piping and instrumentation diagrams (P&IDs) be appropriately updated and that the above information be incorporated into the system description. This was DFSER Confirmatory Item 9.2.13-2. The staff reviewed Figure 9.2-3 of the SSAR and concluded that the surge tank isolation valves are included in the safety-related portion of the system. This is acceptable and resolved DFSER Confirmatory Item 9.2.13-2 and the applicable part of DSER (SECY-91-235) Open Item 55.

3. The staff noted that it was not clear which division of the HECW supplies chilled water to the DG in Zone C. The staff also noted that SSAR Figure 9.4-4 and Section 9.2.13 were inconsistent in showing the number of divisions for the system.

   SSAR Section 9.2.13 currently shows the three divisions of the HECW system. Divisions A, B, and C support diesel generator areas A, B, and C, respectively. This eliminates the discrepancy in the DSER (SECY-91-235) and is acceptable. Therefore, the applicable part of DSER Open Item 55 is resolved.

Based on the above information, the staff finds that the HECW system will comply with GDC 4 regarding protection for the system against dynamic effects resulting from postulated piping failures and internally- and externally-generated missiles, and with GDC 44 regarding system reliability.

The HECW system water quality requirements are established by the MUWP system, as this system is the source of the water for the HECW system surge tanks. (See Section 9.2.10 of this report.)

The design of the HECW system includes provisions to allow periodic inservice inspection of all the system components to ensure the integrity of the system and its capability to perform its intended function. Local display devices indicate vital parameters required in testing and inspections. For example, chilled water flow rate and temperature of the system can be checked by viewing the display of locally-mounted pressure and temperature gauges at the main control panel. The staff reviewed the SSAR and concluded that the system also includes provisions to permit periodic testing of system components as well as the system as a whole. The SSAR states that this testing capability includes structural and leak-tightness visual inspection, tests of entire system operability, and tests of system component operability and performance. This is acceptable. The staff finds that the design of the system complies with the requirements of GDC 45 and 46 for inspection and testing of safety-related cooling water systems.

As a result of further staff review, an additional concern was identified regarding the HECW system. Because of the properties of the refrigerant used in the HECW chiller units, unique problems may arise in recovering the units following a SBO. In the DFSER, the staff requested GE to provide an analysis regarding the HECW system's ability to recover following a postulated SBO condition. This was DFSER Open Item 9.2.13-1. As described below, the SSAR, Appendix 1C, provides information regarding the HECW system's response during an SBO. The system can be connected to an alternate ac (AAC) power source within 10 minutes after the initiation of an SBO. The CTG serves as the AAC source. During the SBO, little heat will be generated in the areas served by the HECW since only battery-powered equipment will be running. The temperature increase in the rooms over the 10-minute period will not exceed the qualification limits for the equipment in these areas. Once AAC is available, fans (which are normally powered from Class 1E sources) will be available to remove heat from the areas served by the HECW. As the chillers become available, normal area temperatures will be restored. Furthermore, the SSAR states that the applicant referencing the ABWR will provide the necessary means for restarting the system on an SBO after the AAC is available. The staff concludes that the design provisions in the SSAR give adequate assurance that, during an SBO condition, safety-related equipment served by the HECW system will not exceed its environmental operability limits. This is acceptable and resolved DFSER Open Item 9.2.13-1.

In the DFSER, the staff requested GE to provide adequate design description and the ITAAC relating to the HECW system. This was DFSER Open Item 9.2.13-2. Subsequently, GE provided a revised set of design description 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 the above, DFSER Open Item 9.2.13-2 is resolved.

The HECW system includes pumps, chillers, valves, piping, surge tanks, makeup piping, and the points of connection with other systems. Portions of the HECW system that are necessary for safe shutdown, accident prevention, or accident mitigation are designed to seismic Category I and QG B requirements. The staff reviewed the SSAR design criteria, design bases, and safety classification for the HECW system against the requirements for supplying adequate cooling water for the safety-related ECCS components and reactor auxiliary equipment for all conditions of plant operation. The staff concludes, as discussed above, that the design of the HECW system is acceptable and meets the requirements of GDC 2, 4, 5, 44, 45, and 46, the applicable criteria of SRP Section 9.2.2.

9.2.14 Turbine Building Cooling Water System

The staff reviewed the non-safety-related TCW system in accordance with applicable portions of SRP Section 9.2.2. Staff acceptance of the TCW system design is based on meeting GDC 2 as it relates to the system withstanding the effects of natural phenomena; GDC 4 as it relates to the system withstanding the effects of failed equipment and piping during both normal and accident conditions; GDC 5 as it relates to the capability of shared SSCs to perform requires safety functions; 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, providing suitable redundancy for components given a single active component failure, and the capability to isolate part of the system so that the safety function is not compromised; and GDC 45 and 46 as they relate to permitting inservice inspection and testing, respectively, for safety-related equipment.

The TCW system is a non-safety-related system designed to remove heat for various turbine island auxiliary equipment. The TCW is a closed-loop system consisting of three 50-percent capacity pumps, three 50-percent capacity heat exchangers, a surge tank (shared with the corresponding division of the HNCW system), and associated piping, valves, and instrumentation. In Section 9.2.14 of the DSER (SECY-91-235), the staff noted, as part of Open Item 56, that the description of the TCW in SSAR Section 9.2.14 contained several inconsistencies. The component and system descriptions in SSAR Section 9.2.14.2.3 and in SSAR Figure 9.2-6a did not agree with the descriptions of SSAR Sections 9.2.14.2.1 and 9.2.14.2.2 and GE's responses to RAIs. The discrepancies (in particular, the number of heat exchangers and pump capacities) required correction. GE submitted information and committed to update the SSAR to eliminate the discrepancies between the figures and the text. GE stated that the system will be a single-loop system consisting of three pumps and three heat exchangers, each of 50-percent-capacity as stated above. Incorporation of the corrected information into the SSAR was Confirmatory Item 9.2.14-1 in the DFSER. The staff reviewed the SSAR and concluded that the inconsistencies had been corrected. This is acceptable and resolved the applicable part of DSER Open Item 56 and DFSER Confirmatory Item 9.2.14-1.

The SSAR includes pump flow requirements, heat exchanger capacity, and turbine service water temperature limits for TCW operation. Demineralized water is added automatically based on surge tank level indications. During normal operation, two pumps are in operation and the third is in standby. The third pump automatically starts on low pump discharge pressure. There are no connections between the TCW and safety-related water systems, and the TCW system is designed in accordance with QG D standards.

The TCW system is located in and near the turbine building, away from safety-related systems. In response to RAIs 430.206 and 430.207, GE indicated that failure of any TCW components, including the atmospheric surge tank, would not cause any safety-related equipment to fail. In the DSER (SECY-91-235), the staff noted that, from equipment layout diagrams reviewed, this statement appears to be true for all equipment shown on the diagrams. The staff also noted that the atmospheric surge tank did not appear on these diagrams and the staff verification that failure of this component would not affect safety-related systems was not possible. The SSAR identifies the surge tank as being located above the TCW pumps in the turbine building. This location would place the tank in an area away from safety-related components, and failure of the tank would not affect any safety-related components, which is acceptable. This description meets Position C.2 of RG 1.29, "Seismic Design Classification," Revision 3, and resolved the remaining part of DSER Open Item 56.

Based on this information, the staff concludes that the TCW system design meets the guidelines of Position C.2 of RG 1.29, pertaining to seismic requirements for non-safety-related systems and components. Therefore, the design meets the requirements of GDC 2 in accordance with SRP 9.2.2.

The requirements of GDC 5 regarding the sharing of SSCs do not apply because the ABWR is designed as a single-unit facility. An application for a multi-unit facility will require review of the design for compliance with GDC 5.

Because the TCW system is non-safety-related and does not interface with a safety system, the remaining requirements (GDC 4, 44, 45, and 46) of SRP Section 9.2.2 do not apply since they address requirements of safety-related systems.

In the DFSER, the staff requested GE to provide adequate design description and the ITAAC relating to the TCW system. This was DFSER Open Item 9.2.14-1. Subsequently, GE provided a revised set of design description 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 the above, DFSER Open Item 9.2.14-1 is resolved.

The TCW system includes pumps, chillers, valves, piping, surge tanks, makeup piping, and the points of connection with other systems. The staff reviewed the applicant's proposed design criteria, design bases, and safety classification for the TCW system against the requirements for supplying adequate cooling water for the auxiliary equipment for all conditions of plant operation. The staff concludes, as discussed above, that the design of the TCW system is acceptable and meets the applicable requirements of GDC 2, 4, 5, 44, 45, and 46 and the guidelines of SRP Section 9.2.2.

9.2.15 Reactor Service Water

The staff reviewed the reactor service water (RSW) system in accordance with SRP Section 9.2.1. Staff acceptance of the RSW system design is based on meeting the requirements of GDC 2 as it relates to protecting SSCs important to safety from the effects of natural phenomena; GDC 4 as it relates to protecting SSCs important to safety from the effects of piping and equipment failures during both normal and accident conditions; GDC 5 as it relates to the capability of shared SSCs to perform required safety functions; 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, providing suitable redundancy for components given a single active component failure, and the capability to isolate part of the system so that the safety function is not compromised; and GDC 45 and 46 as they relate to inservice inspection and testing of safety-related systems and components.

The portion of the RSW system within the ABWR design scope includes all the piping, valves, instrumentation, and controls within the control building. All other equipment outside the control building, including the RSW pumps, are outside the ABWR design scope and are the responsibility of the COL applicant.

The SSAR includes a conceptual design and interface requirements for that portion of the RSW system outside the scope of the ABWR design as required by 10 CFR Part 52.

The function of the RSW system is to provide cooling water to the RCW system (reviewed in Section 9.2.11 of this report) for distribution to several safety-related and non-safety-related loads. The RSW is required to operate at normal power, reactor shutdown, hot standby, and after a postulated LOCA. Under each of these conditions, the RSW is required to function both with and without preferred ac power available and with a single active failure.

The RSW system is an open-cycle system that provides cooling water to the RCW heat exchangers. The RSW system supports no other heat loads. The RSW system picks up heat from the RCW heat exchangers and rejects the heat to the UHS, which is to be designed by COL applicants referencing the ABWR design as discussed in Section 9.2.5 of this report. Although earlier SSAR amendments discussed the total heat rate, total flow rate, temperature drop, and pressure drop at the RCW heat exchangers for all identified modes of operation for the RCW system, the staff completed the DSER (SECY-91-235) before receiving in an amendment similar parameters for the RSW system (including identification of sufficient NPSH at pump suction locations for low water levels). This was identified in Section 1.8 of the DSER (SECY-91-235) as Open Item 57. GE submitted information and committed to update the SSAR, identifying these parameters as actions by the COL applicant. This was DFSER COL Action Item 9.2.15-1. SSAR Section 9.2.15.2 requires the COL applicant referencing the ABWR design will submit sufficient information to allow the staff to perform a plant-specific safety evaluation on that portion of the RSW system outside of the ABWR design scope. This is acceptable. On the basis of the above, the applicable part of DSER Open Item 57 is resolved.

The RSW system is composed of three mechanically and electrically independent divisions. Each division consists of its own separate piping from intake to discharge, two pumps, two strainers, valves, and instrumentation. Each RSW division supplies cooling water to one division of the RCW system.

The SSAR states that the RSW system will be able to function during abnormally low or high water levels and that steps are taken to prevent organic fouling that may degrade system performance. These steps include installing trash racks, biocide treatment (or non-biocide treatment where biocide treatment is not allowed), and thermal backwash capabilities. In the DFSER, the staff stated that selection of appropriate measures is site-specific and, therefore, the responsibility of the COL applicant. This was DFSER COL Action Item 9.2.15-2. SSAR Section 9.2.15.2.2 states that the COL applicant will design the out-of-scope portion of the RSW system to prevent excessive organic fouling, erosion, and corrosion of the RSW piping. This is acceptable and resolved the applicable portion of DSER Open Item 57.

In the DFSER, staff review of the interface requirements in the SSAR and the certified design material (CDM) was in progress. This was DFSER Open Item 9.2.15-1. Subsequently, GE submitted the design description, the ITAAC, and the interface requirements relating to the RSW system. The adequacy and acceptability of the design description, the ITAAC, and the interface requirements are evaluated in Section 14.3 of this report. On the basis of the above, DFSER Open Item 9.2.15-1 is resolved.

System protection from water-hammer is achieved through the use of high point vents and operational procedures requiring filling and venting of any sections of the system before operation. In the DFSER, the staff stated that the COL applicant should supply these procedures. This was DFSER COL Action Item 9.2.15-3. SSAR Section 9.2.15.2.1(6) states that the COL applicant will submit the above information. This is acceptable.

All portions of the RSW system are designed to seismic Category I, QG C, requirements. SSAR Table 3.2-1 states that the RSW pumps are located in the structures associated with the UHS; therefore, all portions of the system will be located in seismic Category I, flood- and missile-protected structures. The design of the RSW system complies with the requirements of GDC 2 with respect to its protection from natural phenomena, and meets Position C.1 of RG 1.29 with respect to its seismic requirements.

The SSAR states that both the mechanical equipment and piping and electrical equipment, including instrumentation and controls, of the redundant divisions of the RSW system are sufficiently separated and protected to ensure availability of the needed equipment to shut down the reactor in the event of any of the following occurrences: flooding or spraying steam release induced by pipe rupture or equipment failure, pipe whip and jet forces from a postulated nearby high-energy line break, missiles from equipment failure; fire; non seismic Category I equipment failure, or a single active component failure in the system.

In the earlier SSAR amendments, insufficient detail was provided to ensure that this design criteria can be met. Specifically, location and design features for the RSW pump and associated equipment were not specified prior to DSER (SECY-91-235) completion. This was identified in Section 1.8 of the DSER (SECY-91-235) as part of Open Item 57. The SSAR provides this information as an interface requirement and establishes interface criteria to assure an appropriate design. The staff reviewed this information and found it acceptable. This resolved the applicable part of DSER Open Item 57.

A portion of each division of the RSW system is located in separate divisional areas in the control building basement. An RSW pipe break in this area would expose the safety-related RCW heat exchangers to flood water from the UHS through the RSW system. GE incorporated a flood protection feature to isolate the affected division of the RSW system on a high water level signal in the divisional space within the control building. In this way, only the division experiencing the break and subsequent flooding will be affected. The redundant divisions of RCW and RSW will still be available to perform their safety functions. Based on this information, the staff concludes that the ABWR RSW design and related commitments are adequate to ensure compliance with GDC 4 as it relates to protection of safety-related equipment from the dynamic effects resulting from postulated pipe failures, floods, and internally- and externally-generated missiles. This resolved the second part of DSER Open Item 57. Incorporation of this information into the SSAR was identified as Confirmatory Item 9.2.15-1 in the DFSER. SSAR Section 9.2.15.2.1 states that the COL applicant referencing the ABWR will design the out-of-scope portion of the RSW system to withstand the effects of piping and equipment failures. This is acceptable and resolved DFSER Confirmatory Item 9.2.15-1.

The staff notes that, in addition to the interface criteria identified above, the portion of the RSW system within the scope of the COL applicant's action must meet the following:

- the system will be sized to remove the heat associated with the worst-case condition listed in Table 9.2-4 in the SSAR.

- RSW piping length will not exceed 2000 m (~6600 ft) and redundant system isolation capability will be designed to ensure that flooding in the control building resulting from the failure of RSW piping is within the bounds of the flood analysis. This is discussed in Section 3.4.1 of this report.

The requirements of GDC 5 for the sharing of SSCs do not apply because the ABWR is designed as a single-unit facility. An application for a multi-unit facility will require review of the design for compliance with GDC 5.

At least one RSW pump in each of the three divisions of the RSW system will be operating. This configuration ensures immediate availability of the RSW system for plant shutdown in the event of a LOCA. A LOPP concurrent with a LOCA will result in a temporary loss of pumping until the automatically sequenced restart of RSW pumps from the emergency diesel generator loading sequence. Upon the occurrence of a LOCA, the second RSW pump starts and the third heat exchanger in each division is placed in service. The design criteria for the RSW include the requirement that a single active or applicable passive component failure will not compromise the ability of the RSW system to transfer heat loads from the RCW system to the UHS.

GE discussed protection from adverse environmental conditions, such as freezing, icing, and biofouling in its response to Unresolved Safety Issues B-29 and B-32 and Generic Issue (GI) 51. The staff discusses the adequacy of the ABWR design provisions and interface requirements with regard to these issues in Sections 20.1 and 20.2 of this report.

The staff reviewed design information and interface criteria supplied by GE and concludes in view of the above that the RSW system design ensures that the heat removal requirements of GDC 44 will be met and are acceptable.

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, tests for entire system operability, and tests for system component operability and performance. Testing will be conducted to simulate as closely as possible the entire operational sequence of the RSW system from reactor shutdown to LOCA. The system design also incorporates provisions for accessibility to permit inservice inspection as required. Therefore, the system complies with the requirements of GDC 45 and 46 with respect to inspection and testing requirements for cooling water systems.

In the DFSER, the staff requested GE to provide adequate design description, the ITAAC, and interface requirements relating to the RSW system. This was DFSER Open Item 9.2.15-2. Subsequently, GE provided a revised set of design description and ITAAC. The adequacy and acceptability of the design description, the ITAAC, and the interface requirements are evaluated in Section 14.3 of this report. On the basis of the above, DFSER Open Item 9.2.15-2 is resolved.

The design and related commitments of the RSW system comply with the applicable requirements of GDC 2, 4, 5, 44, 45, and 46, and the guidelines of SRP Section 9.2.1 discussed above, and are acceptable.

9.2.16 Turbine Service Water System

The staff reviewed the non-safety-related turbine service water (TSW) system in accordance with applicable portions of SRP Section 9.2.1. The TSW system design is acceptable if it meets the requirements of GDC 2 as it relates to protecting SSCs important to safety from the effects of natural phenomena; GDC 4 as it relates to protecting SSCs important to safety from the effects of piping and equipment failures during normal and accident conditions; GDC 5 as it relates to the capability of shared SSCs important to perform safety functions; 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, providing suitable redundancy for components if a single active component fails, and the capability to isolate part of the system so that the safety function is not compromised; and GDC 45 and 46 as they relate to inservice inspection and testing of safety-related systems and components.

The portion of the TSW system design within the ABWR design scope includes all the piping, valves, instrumentation, and controls within the turbine building. All other equipment outside the turbine building, including the TSW pumps, are outside the ABWR design scope and are the responsibility of the COL applicant.

The SSAR includes a conceptual design and interface requirements for that portion of the TSW system outside the scope of the ABWR design as required by 10 CFR Part 52.

The TSW system is a non-safety-related system designed to transfer heat from the TCW system heat exchangers to the power cycle heat sink. The TSW includes three 50-percent capacity pumps, three duplex strainers, and associated piping, valves, and instrumentation. In the DSER (SECY-91-235), the staff stated that GE did not provide system parameters, (pump flow requirements, system design pressure) but submitted a requirement that water supplied to the TCW heat exchangers be at a temperature not to exceed approximately 40 °C (100 °F). In response, GE committed to update the SSAR concerning the design of the TSW system. SSAR Table 9.2-16 provides the specific system parameters described above.

This is acceptable and resolved DFSER Confirmatory Item 9.2.16-1.

In the DFSER, the staff noted contradictory information concerning the number of pumps in the system. The text of SSAR Section 9.2.16 referred to two pumps and two duplex strainers. However, SSAR Figure 9.2-8 and Table 9.2.17 both showed three 50-percent capacity pumps. Resolution of this discrepancy was identified as Open Item 9.2.16-1 in the DFSER. The SSAR clarifies that the system consists of a single loop containing three vertical wet pit pumps. During normal operation, two pumps are operating and the third is in standby. The standby pump starts automatically if an operating pump trips or if the pump discharge pressure drops below a preselected limit. This clarification is acceptable and resolved DFSER Open Item 9.2.16-1.

The TSW system is located in the intake structure (the power cycle heat sink pump house) and the turbine building. The system does not have any connections with safety-related systems. The applicant must demonstrate that all safety-related components, systems, and structures are protected from flooding in the event of a pipeline break in the TSW system in order to meet Position C.2 of RG 1.29, and thus, comply with GDC 2. The staff recognized the site-specific location of some TSW components and stated in the DSER (SECY-91-235) that this requirement may need to be specified as an interface requirement. This was identified in Section 1.8 of the DSER as Open Item 58. Incorporation of this requirement in the SSAR was identified as Confirmatory Item 9.2.16-2 in the DFSER. GE stated that the COL applicant should submit an interface requirement for flood protection of safety-related SSCs in case of TSW component failures. The SSAR includes sufficient information to ensure that an applicant referencing the ABWR can design the out-of-scope portion of the TSW system to prevent flood damage to safety-related equipment as a result of a break in the TSW piping. The staff notes that in the SSAR, Section 3.4.1 includes a flood analysis which indicated that any flooding resulting from a break in the TSW system will be prevented from affecting safety-related equipment by several means. First, a break in the TSW line will result in a high water level alarm in the condenser pit. The operator can then isolate the system. If the operator is unsuccessful in isolating the system, flood waters would rise to plant grade where it would flow out of the truck door and onto the ground at this elevation. Second, the below-grade tunnel which connects the turbine, radwaste, and reactor buildings is sealed at all ends to prevent water from entering any of the buildings. On the basis of this information, the staff concludes that flooding as a result of a break in the TSW line will not adversely affect any safety-related equipment. Therefore, the guidelines of Position C.2 of RG 1.29 and the requirements of GDC 2 can be met, and DFSER Confirmatory Item 9.2.16-2 is resolved. The CDM aspects are addressed in the discussion of DFSER Open Item 9.2.16-2 below.

The requirements of GDC 5 for the sharing of SSCs do not apply because the ABWR is designed as a single-unit facility. An application for a multi-unit facility will require review of the design for compliance with GDC 5.

The remaining requirements of SRP Section 9.2.1 (GDC 4, 44, 45, and 46) do not apply because the TSW system is non-safety-related and has no connections to safety-related systems.

In the DFSER, the staff requested GE to provide adequate design description, interface requirements, and the ITAAC relating to the TSW system. This was DFSER Open Item 9.2.16-2. Subsequently, GE provided a revised set of design description 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 the above, DFSER Open Item 9.2.16-2 is resolved.

The design and related commitments of the TSW system comply with the requirements of GDC 2 as discussed above. The requirements of GDC 4, 5, 44, 45, and 46 do not apply. In view of the above the staff concludes that the system design and related commitments meet the applicable acceptance criteria of SRP Section 9.2.1 and are acceptable.

9.3 Process Auxiliaries

9.3.1 Compressed Air Systems

The design of the compressed air (CA) systems are discussed in SSAR Sections 6.2.5 (Atmospheric Control System), 6.7 (High Pressure Nitrogen Gas Supply), 9.3.6 (Instrument Air), and 9.3.7 (Service Air) and were reviewed in accordance with SRP Section 9.3.1. The review of the CA systems involved a review of information in the SSAR and GE's responses to staff RAIs. The acceptance criteria for the safety-related portions of the CA systems is provided in SRP Section 9.3.1 and includes compliance with GDC 1 as it relates to systems and components important to safety being designed, fabricated, and tested to quality standards in accordance with the importance of the safety functions to be performed; GDC 2 as it relates to safety-related CA systems being capable of withstanding the effects of natural phenomena; and GDC 5 as it relates to the capability of shared systems, and components to perform required safety functions. The staff reviewed the CA systems for compliance with GDC 1 as discussed in Section 3.2 of this report.

The requirements of GDC 5 for the sharing of SSCs do not apply because the ABWR is designed as a single-unit facility. An application for a multi-unit facility will require review of the design for compliance with GDC 5.

The CA system consists of four subsystems: (1) the atmospheric control system (ACS), (2) the high-pressure nitrogen gas supply (HPIN) system, (3) the instrument air (IA) system, and (4) the service air (SA) system. These systems provide compressed gas (either air or nitrogen) to operate safety-related equipment relied upon to mitigate the consequences of design-basis events and plant equipment used for normal facility operation. The non-safety-related portions of the system are interconnected with the safety-related portions of the system; the designs of the four subsystems are summarized below.

The ACS system establishes and maintains an inert atmosphere within the primary containment during all plant operating modes except (1) during shutdown for refueling or equipment maintenance, and (2) during limited periods of time to permit access for inspection at low reactor power (15 percent). The ACS is non-safety-related except as necessary to ensure primary containment integrity (e.g., penetrations and isolation valves). The ACS includes nitrogen storage tanks, vaporizers, valves and piping carrying nitrogen to the containment, valves and piping from the containment to the SGTS and reactor building HVAC exhaust line, non-safety oxygen-monitoring, and all related instrumentation and controls. The ACS provides nitrogen from the nitrogen evaporator to the HPIN system during normal operation.

The HPIN system consists of both safety-related and non-safety-related portions. A single non-safety-related line provides a continuous nitrogen supply to all pneumatically operated components in the primary containment during normal operation. During normal operation, the HPIN system is supplied from the nitrogen gas evaporator/storage tank through the makeup line from the ACS.

The safety-related portion of the system consists of two independent divisions; each division contains a safety-related nitrogen supply capable of supplying 100-percent of the requirements of the division being serviced. High-pressure nitrogen gas storage bottles supply nitrogen gas for the safety-related portion of the system. Tielines connect the non-safety-related portion of the system to each division of the safety-related portion of the system. Each tieline has a motor-operated isolation valve.

The IA system provides dry, oil-free, CA for valve actuators, non-safety-related instrument control functions, and general instrumentation and valve services outside the containment. (All instrumentation and control systems inside the containment are supplied with nitrogen gas during normal plant operation.) The primary containment penetrations of the IA system are of seismic Category I, QG B, design and are equipped with sufficient isolation valves to satisfy single-failure criteria. In GE's response to RAI 430.215, the staff noted in the DSER (SECY-91-355), as an unnumbered confirmatory issue, that the reference to ". . . containment penetrations and drywell penetrations of the instrument air system . . ." in SSAR Section 9.3.6.1.1 should be revised to reference primary containment penetrations only. The correction of the SSAR text was identified as DFSER Confirmatory Item 9.3.1-1. GE subsequently made this correction in Amendment 24 of the SSAR. DFSER Confirmatory Item 9.3.1-1 is resolved.

The SA system is designed to provide CA of suitable quality for non-safety-related functions. The SA system provides CA for services requiring air of lower quality than that provided by the IA system. The containment penetrations and drywell penetrations of the SA system are of seismic Category I, QG B, design and are equipped with sufficient isolation valves to satisfy single-failure criteria. The SA system does not directly interface with the HPIN system and does not perform any safety-related function.

As noted earlier, only the HPIN system provides compressed gas to safety-related components. However, the ACS and the IA system directly interface with the HPIN system and could affect the reliability of safety-related components relied upon to mitigate the consequences of design-basis events. Therefore, the staff reviewed these three systems in assessing the adequacy of the CA systems in accordance with acceptance criteria in SRP Section 9.3.1.

The staff reviewed the SSAR to determine the safety-related portions of the CA system and stated in the DSER (SECY-91-355) that the text and figures in SSAR Section 6.7 did not clearly state which portions of the HPIN system are safety-related. However, the response to an RAI, and the valve and instrument numbers on a revision to SSAR Figure 6.7-1 specify which portions of the system are safety-related. These portions include the nitrogen storage bottles and their headers up to and including valves F002A through F002D, piping and valves from F002 (A through D) to the accumulators for the automatic depressurization system (ADS) valves, and piping from the cross-tie valves F012A and F012B to the piping leading to the accumulators (identified previously). Additional safety-related piping in the IA system and ACS include piping and valves from F200 to F208, inclusive.

The staff noted an inconsistency in the designation of the inboard isolation valve. GE referred to this valve as F209 in all drawings and as F208 in the responses to RAIs. In DSER (SECY-91-355) Table 9.3.1-1, the staff identified (DSER Open Item 69) additional numbering inconsistencies, including valve numbers and valve operator types. The staff also stated [DSER (SECY-91-355) Open Item 70] that the SSAR text and figures did not include all information on the safety classification of components before the staff completed DSER (SECY-91-355). While the information in response to requests for information is acceptable, the staff stated in the DSER (SECY-91-355) that this information should be fully incorporated into the SSAR. The inconsistencies and lack of information noted in the DSER (SECY-91-235) were addressed in revised figures to be incorporated into the SSAR. This information resolved DSER Open Items 69 and 70; however, the staff identified incorporation of this information into the SSAR as Confirmatory Item 9.3.1-2 in the DFSER. GE submitted Amendment 20 of the SSAR which corrected the inconsistencies and included the additional information. This is acceptable, and resolved DFSER Confirmatory Item 9.3.1-2.

Contrary to the guidelines of the SRP, the SSAR does not indicate the failure mode for the valves in the HPIN system. Except as noted below, and assuming that the MOVs fail "as-is," the system configuration is acceptable. In the DSER (SECY-91-355), the staff stated as Open Item 71 that information related to the failure mode of components should be incorporated into the SSAR text and drawings. GE indicated in a letter dated March 11, 1992, that the motor-operated valves in the air systems fail as-is; air-operated valves fail open (unless otherwise indicated) in the HPIN, IA, and SA systems and fail closed in the ACS. This information resolved DSER Open Item 71; however, identification of the failure modes of the valves in the CA systems (i.e., confirmation of the failure states) in the SSAR was identified as DFSER Confirmatory Item 9.3.1-3. GE modified SSAR Figures 6.7-1, 6.2-39, 9.3-6, and 9.3-7 to include the failure modes of all pneumatically-operated valves in the CA systems. All motor-operated valves fail as-is. This is acceptable to the staff and resolved DFSER Confirmatory Item 9.3.1-3.

In the DSER (SECY-91-355), the staff stated that Valves AO F018A and B in SSAR Figure 6.7-1 were identified as "NC, FO" (i.e., normally closed and fail open) and that this was inconsistent with Section 19E.2.1.2.2.2(2)(b), which states that upon a loss of power, the operator will have to manually open these valves. The staff noted that this is also inconsistent with a design which should protect the storage bottles from inadvertent depressurization due to a postulated line break. Confirmation that these valves do not fail in the open position was identified as DSER (SECY-91-355) Open Item 72. In a letter dated March 11, 1992, GE changed these valves to motor-operated valves that are normally locked closed and as such they will fail as-is, closed. This resolved the DSER Open Item 72. Incorporation of this information into the SSAR was identified as DFSER Confirmatory Item 9.3.1-4. In resolving DFSER Confirmatory Item 9.3.1-2, GE changed the designation of valves F018A and B to F003A and B in SSAR Amendment 20 and changed these valves from pneumatically-operated valves to motor-operated valves which are normally closed. The staff found the modifications acceptable. DFSER Confirmatory Item 9.3.1-4 is resolved.

The vessels, piping, and fittings of the safety-related portions of the HPIN system (except penetrations) are designed to the requirements of seismic Category I, ASME Code Section III, Class 3, QG C, and Quality Assurance B. The SSAR states that the cross-tie valves (i.e., F012A and B) that connect the safety-related portions of the HPIN system to the non-safety-related portions are safety-related. While two isolation valves are not provided on the non-safety-related/safety-related interface within the HPIN system, the system is deemed adequate in accordance with information GE provided in the response to RAI 430.211. One isolation valve is provided between each division of the safety-related HPIN system and the non-safety-related portion of the system. Additionally, check valves are provided to prevent backflow of nitrogen from the accumulators through any possible break in the non-safety-related portion of the system. The inboard isolation valves are check valves and each safety-related accumulator is downstream of a separate check valve. The accumulators are sized to perform their function. The combination of isolation valves, check valves, and the sizing of the accumulators ensures that the system would fulfill its function in the event of a rupture in the non-safety-related portion of the piping.

The piping and valves for the containment and drywell penetrations for the HPIN and IA systems are designed to the requirements of seismic Category I, ASME Code Section III, Safety Class 2, QG B, and Quality Assurance B. The isolation provisions for the ACS primary containment penetrations include two isolation valves that are both located outside the primary containment, which is not strictly in conformance with GDC 56. However, the penetrations do not extend inside the containment, and an inboard isolation valve would not be practical (as described by GE in the response to RAI 430.209). An inboard isolation valve for the ACS would be exposed to a more severe environment and would not be easily accessible for inspection, surveillance, and maintenance. The staff approved a similar design for the GESSAR II BWR/6 design. Therefore, the staff determined that the isolation design is acceptable. The containment isolation valve provisions are reviewed in detail in Section 6.2.6 of this report.

Based on this information, the staff concluded that the CA system complies with Positions C.1 and C.2 of RG 1.29 regarding the ability of the system to withstand the effects of earthquakes.

The safety-related portions of the HPIN system are located within the reactor building. The reactor building is designed to withstand and protect equipment from tornados, externally-generated missiles, floods, and other natural phenomena. In addition, the safety-related portions of the HPIN system will retain their function during a LOCA and seismic events in which non-safety-related portions may be damaged. SSAR Section 6.7.3 states that the space separating the pipe routing of Divisions I and II of nitrogen gas is sufficient to prevent a strike by a single high-energy whipping pipe, the jet force from a single broken pipe, or an internally-generated missile from preventing the other division from accomplishing its safety function. Thus, the system satisfies GDC 2 regarding protection of the system from natural phenomena.

The ADS valves will perform their safety-related functions using compressed gas that will be provided by the HPIN under most conditions. This nitrogen is supplied during normal operation from the ACS nitrogen storage tank. During design-basis events, nitrogen is supplied by accumulators charged by either the ACS, during normal operation, or by nitrogen bottles during periods when the ACS is unavailable. In addition, stored nitrogen can be used to replenish the accumulators or to supplement their operation. The ABWR design uses nitrogen containing particles up to 5 microns. As discussed in DSER Open Item 73 (SECY-91-355), this does not comply with the guidance of ANSI MC 11.1-1976, which recommends that nitrogen used by safety-related components contain particles no larger than 3 microns. In response to this issue, GE indicated that the filters in the HPIN system will be able to remove particles larger than 5 microns from the system, and the nitrogen supply subsystem will supply oil-free nitrogen with a moisture content of less than 2.5-ppm. The 5-micron capacity of the filters is not in compliance with the requirements of ANSI MC 11.1-1976, and GE had not provided sufficient justification for the staff to allow deviation from the 3-micron criterion. In the DFSER, the staff stated that GE should either commit to the 3-micron requirement of the standard or provide a commitment to use equipment demonstrated to be unaffected by the use of nitrogen containing particulates of the larger size. DSER Open Item 73 (SECY-91-355) was identified as DFSER Open Item 9.3.1-1. Subsequently, GE modified the SSAR to clarify the design requirements for components using this nitrogen source. GE submitted Amendment 24 to the SSAR. The staff reviewed Section 6.7.2 of this amendment and concludes that GE had clarified that any equipment using this nitrogen will be capable of operating with nitrogen containing 5-micron particulates. The staff finds the modification acceptable. DFSER Open Item 9.3.1-1 is resolved.

IE Bulletin No. 80-01 concerns the operability of the ADS pneumatic supply. The bulletin states that the ADS pneumatic supply may not be operable for all possible events because of a combination of misapplication of check valves, a lack of testing of the accumulator system backing up each ADS valve operator, and questions about the continued operability of the pneumatic supply in a seismic event.

The bulletin requires licensees of GE BWR facilities which use a pneumatic operator for ADS function to:

(1) Determine if the facility has installed hard-seat check valves to isolate the ADS accumulator system from the pneumatic supply system.

(2) Determine if periodic leak tests have been performed in the ADS accumulator systems to assure emergency pneumatic supply for the FSAR-required number and duration of valve operations.

(3) Review seismic qualifications of the ADS pneumatic supply system:

   (a) from accumulator system isolation check valve to ADS valve operator,

   (b) from isolation valve outside containment up to ADS accumulator check valve.

(4) Based upon determination of Items 1, 2, and 3 above, evaluate the operability of the ADS for the conditions under which it is required to be operable, including a seismic event. If operability cannot be established, adhere to appropriate technical specification action statement.

Operational experience has shown that check valves with hard seats may lead to excessive valve leakage, thus undermining the ability of the accumulators to provide the required pneumatic fluid to ensure that the ADS valves actuate. The ABWR design may use check valves with a hard seat. GE states that accumulator operability will be assured by ensuring that system leakage does not exceed 28 L/h (1 scfh) per valve. SSAR Sections 6.7.2 and 6.7.4 state that periodic leakage testing will be performed to ensure that the leakage rate does not exceed 28 L/h (1 scfh) for each valve. Section 6.7.3 states that the safety-related portions of the ADS system are seismic Category I. The section of system piping addressed in the bulletin is classified as safety-related for the HPIN system. Thus, these sections of the system will be able to withstand the conditions associated with a seismic event. Based on this information, the staff concludes that the HPIN design is sufficient to ensure that the ADS SRV accumulators will supply adequate nitrogen to the ADS SRVs to ensure proper valve actuation. The system design adequately addresses the issues in IE Bulletin No. 80-01.

The staff stated in the DSER (SECY-91-355) that SSAR Section 9.3.6.1.2 indicated that the non-safety-related IA system is also used as a backup to the nitrogen system when, during normal operation, the nitrogen gas supply pressure drops below a specified setpoint. The staff noted that this conflicted with GE's earlier response to RAI 430.218, dated March 11, 1992, which states: "Instrument air system does not serve as a backup to HPIN system during normal operation . . . ." The resolution of this discrepancy was identified as Open Item 74. Subsequently, GE committed to revise the response to RAI 430.218 to indicate that the IA system can be used as a backup to the HPIN system when nitrogen pressure drops below the system low pressure setpoint and recovery efforts have failed. IA would be used until repairs to the HPIN system are completed. This information resolved DSER Open Item 74, however, incorporation of this information into the SSAR was identified as DFSER Confirmatory Item 9.3.1-5. GE supplied the required information in Amendment 22 which is acceptable to the staff. DFSER Confirmatory Item 9.3.1-5 is resolved.

In evaluating the IA system as a potential backup to the HPIN system, the staff stated in the DSER (SECY-91-355) that it had found that the system complies with all aspects of the ANSI MC 11.1-1976 criteria except for particulate size. The ABWR design proposes a 5-micron criterion for particulate size that is contrary to the 3-micron criterion of the ANSI standard. This was identified as DSER Open Item 75. GE had not justified this aspect of the CA system design, therefore, the IA system's compliance with the requirements of GDC 1 remained an outstanding issue. The staff reclassified DSER Open Item 75 as Open Item 9.3.1-2 in the DFSER. The staff reviewed Section 9.3.6.2 of Amendment 24 of the SSAR and concludes that GE has clarified that any equipment using this air will be capable of operating with air containing 5-micron particulates. The staff finds the modification acceptable. DFSER Open Item 9.3.1-2 is resolved.

The staff reviewed the preoperational testing of the CA systems and compliance with RG 1.68.3 as addressed in Section 14 of this report.

GE submitted the design description and the ITAAC relating to the CA systems. This was identified as DFSER Open Item 9.3.1-3. 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, DFSER Open Item 9.3.1-3 is resolved.

The CA systems include all components and piping and the points of connection with other systems. The safety-related HPIN system provides a continuous nitrogen supply to safety-related components and is classified as seismic Category I and QG C. The staff reviewed the applicant's design and design criteria for the safety-related CA systems to verify that they conform to the Commission's regulations in the GDC, and to applicable regulatory guides and industry standards. As discussed in detail above, the staff concludes that the design of the CA systems is acceptable and conforms to the requirements of GDC 1, 2, and 5 for quality standards, protection from natural phenomena, and sharing of systems and components, and meets the guidelines of SRP Section 9.3.1.

9.3.2 Process and Post-Accident Sampling Systems

9.3.2.1 Process Sampling System

The process sampling system (PSS) is designed to collect water and gaseous samples contained in the reactor coolant system and associated auxiliary system process streams during all normal modes of operation. Provisions are made to ensure that representative samples (except from gaseous streams) are obtained from well-mixed streams or volumes of effluent by the proper selection of sampling equipment, sampling points, and sampling procedures. Additionally, grab samples are obtained for confirmatory analyses and to test for other chemicals. The reactor coolant sample lines penetrating the containment are each equipped with two normally closed, isolation valves which if open, automatically close on a containment isolation actuation signal.

In DSER (SECY-91-355) Open Item 98, the NRC staff determined that SSAR Section 9.3.2.1 contained insufficient information for the staff to evaluate conformance with SRP Section 9.3.2 in the following areas:

1. Under SRP Section 9.3.2, the PSS should include the capability to obtain samples from at least the following points: main condenser evacuation system off gas, SLCS tank, and sumps inside containment and other

locations given in SRP Section 11.5 and those specified in the SSAR.

2. The guidelines in Position C.2 of RG 1.21, "Measuring, Evaluating and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluent from Light-Water-Cooled Nuclear Power Plants" and positions of RG 1.56, "Maintenance of Water Purity in Boiling Water Reactors," Revision 1, should be used to meet the requirements of GDC 13, 14, 26, 63, and 64 unless otherwise acceptable alternatives were proposed. The SSAR did not indicate conformance with these guidelines.

3. In accordance with ANSI N13.1-1969 provisions should be made to ensure that representative samples are taken from gaseous process streams and tanks. This needed to be addressed in the design.

4. To meet 10 CFR 20.1003 in keeping radiation exposures ALARA, and the requirement of GDC 60 to control the release of radioactive materials to the environment, passive flow restrictions to limit reactor coolant loss from a rupture of the sample line should be provided.

5. To meet the requirements of GDC 1 and 2, the seismic design and quality group classification of sampling lines for the PSS should conform to the classification of the system to which each sampling line and component is connected.

GE responded in its March 11, 1992, letter entitled "GE Responses to the Resolution of Issues Related to ABWR DSER Chapters 1, 2, 3, 4, 5, 6, 9, 10, 12, 13, 14, and 15 (SECY-91-355)." In an attachment to this letter (Response to Open Item 98), GE stated that a capability is provided to obtain samples from the main condenser evacuation off gas, SLCS tank, sumps inside containment, liquid radwaste system process lines, and liquid radwaste system collection and sampling tanks. They said that the guidelines of RGs 1.21 and 1.56, and ANSI N13.1-1969 will be used, except that passive flow restrictors are not provided in reactor sampling lines to control the release of radioactive materials from a ruptured sample line. These devices become crud traps during normal operation and overly restrict sampling flow rate during shutdowns when the reactor is at low pressure. They stated that each reactor water sampling line is provided with two remotely operable isolation valves to limit reactor water loss from a sample line rupture. GE also indicated that the seismic design and group classification of sampling lines and their components will conform to the classification of the system into which they are connected. The staff finds this response acceptable and, therefore, DSER Open Item 98 was resolved. The staff concludes that the system meets the cited requirements and guidance and is, therefore, acceptable.

GE has submitted ITAAC and design description in Section 2.11.20 of the CDM for the PSS. This was DFSER Open Item 9.3.2.1-1. GE provided a revised set of design description 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 the above, this item is resolved.

9.3.2.2 Post-Accident Sampling System

After the accident at Three Mile Island Unit 2, the staff recognized the need for an improved post-accident sampling system (PASS) to determine the extent of core degradation following a severe reactor accident. Criteria for an acceptable sampling and analysis system are specified in 10 CFR 50.34(f)(2)(viii) and Item II.B.3 of NUREG-0737. According to these documents, the PASS should have the capability to obtain and quantitatively analyze reactor coolant and containment atmosphere samples without exposing any individual to radiation exceeding 5-rem to the whole body or 75-rem to the extremities (GDC 19) during and following an accident in which there is core degradation. Materials to be analyzed and quantified include certain radionuclides that are indicators of the severity of core damage (e.g., noble gases, isotopes of iodine and cesium, and nonvolatile isotopes), hydrogen in the containment atmosphere, and total dissolved gases, boron, and chloride in reactor coolant samples.

In DSER (SECY-91-355) Open Item 99, the NRC staff determined that the PASS design as described in SSAR Section 9.3.2 was not adequate. The staff stated that GE needed to address the Item II.B.3 of NUREG-0737 and indicate the PASS provisions required to satisfy each of the 11 specified criteria. The staff said that the upper limit for activity levels in liquid samples of 1 Ci/cm^3 in GE's PASS design (SSAR Section 9.3.2.3.1) was not justified. Item II.B.3 of NUREG-0737, Criterion 9, and RG 1.97, "Instrumentation for Light Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," Revision 3, specify reactor coolant and sump gross activity sampling capability in the 1 μCi/ml to 10 Ci/ml range. Delaying sampling until the PASS sample radioactivity decays to 1 Ci/ml is unacceptable because inordinate and unjustified delay in obtaining sample radioactivity results. All PASS systems in operating reactor plants are presently designed with the capability to sample liquids with up to 10 Ci/ml radioactivity.

GE responded (March 11, 1992, letter) explaining that the ABWR PASS has been designed to meet the intent of the 11 requirements listed in NUREG-0737. GE also identified several exceptions to some of the requirements of NUREG-0737. The NRC staff reviewed this information and found it acceptable, subject to revising the SSAR to include the following design features:

1. SSAR Section 9.3.2 indicates that PASS is designed for upper limit for activity samples of 1 Ci/cm^3. All PASS systems in operating reactor plants are designed with capability of sampling liquids up to 10 Ci/ml radioactivity. GE needs to design PASS with the capability of sampling liquids up to 10 Ci/ml. This is DFSER Open Item 9.3.2.2-1.

2. The ABWR PASS does not have capability to obtain pressurized or unpressurized reactor water samples for dissolved gas analysis. GE needs to design PASS to be capable of obtaining reactor water samples 24 hours after the end of power generation in order to evaluate concentrations of dissolved gases and chlorides in the reactor coolant. The information on the amounts of dissolved hydrogen, oxygen and chlorides in the reactor coolant is an important factor in evaluating post-accident conditions existing in the reactor vessel. This is DFSER Open Item 9.3.2.2-2.

The staff concluded in the DFSER that, subject to the resolution of DFSER Open Items 9.3.2.2-1 and 9.3.2.2-2 described above, GE will have adequately described the ABWR PASS.

GE addressed DFSER Open Item 9.3.2.2-1 in its SSAR, Section 9.3.2.3.1, which states that the area radiation is considered safe when the sample radioactivity is about 1 Ci/cm^3. When the sample radioactivity level is higher than 1 Ci/cm^3, abnormal or emergency conditions will be used to assess the situation. This is acceptable.

GE responded to DFSER Open Item 9.3.2.2-2 in its letter of January 28, 1993. The staff also discussed this issue in a telephone conference with GE on January 28, 1993. GE explained that whenever core uncovering is suspected, the reactor vessel is rapidly depressurized, and thus pressurized reactor water samples are not necessary. In a letter dated April 26, 1993, GE revised SSAR Section 9.3.2.3.1 to document this explanation. GE has also included this information in the SSAR. On the basis of the actions above, DFSER Open Item 9.3.2.2-2 is resolved.

In SECY-93-087, the staff recommended that the Commission approve its position that for evolutionary and passive ALWRs of boiling water reactor design, there would be no need for the PASS to analyze dissolved gases in accordance with the requirements of 10 CFR 50.34(f)(2)(viii) and Item III.B.3 of NUREG-0737. In its April 2, 1993, SRM, the Commission approved the staff position to exempt the PASS for the evolutionary and passive ALWRs of boiling water reactor design from analyzing dissolved gases in accordance with the requirements of 10 CFR 50.34(f)(2)(viii) and Item III.B.3 of NUREG-0737.

In SECY-93-087, the staff also recommended that the Commission approve the deviation from the requirements of Item II.B.3 of NUREG-0737 with regard to the requirements for sampling reactor coolant for boron concentration and activity measurements using the PASS in evolutionary and passive ALWRs. The modified requirement would require the capability to take boron concentration samples and activity measurements 8 hours and 24 hours, respectively, following the accident. In its April 2, 1993, SRM, the Commission approved the staff position to require the capability to take boron concentration samples and activities measurements 8 hours and 24 hours, respectively, following the accident.

The ABWR design will have PASS which meets the requirements of 10 CFR 50.34(f)(2)(viii) and Item II.B.3 of NUREG-0737 with the modifications described in SECY-93-087. The system will have the capability to sample and analyze for activity in the reactor coolant and containment atmosphere 24 hours following the accident. This information is needed for evaluating the conditions of the core and will be provided during the accident management phase by the containment high-range area monitor, the containment hydrogen monitor and the reactor vessel water level indicator. The need for PASS activity measurements will arise only during the accident recovery phase and 24 hours sampling time is, therefore, adequate. PASS will also be able to determine boron concentration in the reactor coolant. It will be capable to make this determination within 8 hours following the accident. The concentration of boron is required for providing insights for accident mitigation measures. Immediately after the accident this information will be obtained by the neutron flux monitoring instrumentation which is designed to comply with the criteria of RG 1.97, and which has fully qualified redundant channels capable to monitor over the full power range. Boron concentration measurements will not be, therefore, required for the first 8 hours after the accident.

In order to approve an exemption from the requirements of 10 CFR 50.34(f)(2)(viii) concerning the elimination of analyses for dissolved gases and chlorides in the reactor coolant, special circumstances must exist. For the ABWR, whenever core uncovering is suspected, the reactor vessel is depressurized to approximately the pressure within the wetwell and the drywell which results in partial release of the dissolved gases. Under these conditions, pressurized samples would not be meaningful. Therefore, application of the regulation in this particular circumstance would not serve the underlying purpose of the rule. During accidents when the reactor vessel has not been depressurized (such as when a small amount of cladding damage has occurred), reactor coolant samples could be obtained by the process sampling system.

With regards to the need for chloride analysis, determination of chloride concentrations is of a secondary importance because it is needed only for determining the likelihood of accelerated primary system corrosion which is a slow-occurring phenomenon. Chloride analyses could be performed on the samples taken by the process sampling system. In this case, therefore, the intended purpose of the rule could be achieved without the need for the PASS to have chloride sampling capabilities.

Accordingly, special circumstances required by 10 CFR 50.12(2)(ii) exist for the ABWR in that the regulation would not serve the underlying purpose of the rule in one circumstance and is not necessary in the other circumstance to achieve the underlying purpose of the rule because the intent of rule could be met with alternate design requirements proposed by the applicant. On this basis, the staff concludes that the exemption from analyzing dissolved gases and chlorides in the reactor coolant sample is justified.

GE has submitted ITAAC and design description of the post-accident sampling system as a part of the process sampling system in ITAAC Section 2.11.20. This was DFSER Open Item 9.3.2.2-3. GE provided a revised set of design description and ITAAC. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.1 of this report. On the basis of the above, DFSER Open Item 9.3.2.2-3 is resolved.

9.3.3 Non-Radioactive Drain System

The staff reviewed the non-radioactive drain (NRD) system in accordance with SRP Section 9.3.3. The NRD system is acceptable if it meets the requirements of GDC 2 as it relates to safety-related portions of the system being capable of withstanding the effects of natural phenomena; GDC 4 as it relates to the capability of the system to withstand the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents; and GDC 60 as it relates to providing a means to control the release of radioactive materials in liquid effluent, including anticipated operational occurrences.

SSAR Section 9.3.3 discusses the NRD, which is a non-safety-related system designed to transfer effluents that are not radioactive or potentially radioactive. Radioactive effluents are transferred by the radioactive drain transfer system (DTS), which is discussed in SSAR Section 9.3.8 and reviewed in Section 9.3.8 of this report.

Initially the NRD was not considered part of the ABWR design in the DFSER. Subsequently, GE brought portions of this system within the ABWR design scope. The portions of the system within standard plant buildings are within the ABWR design scope, while those portions outside these buildings are outside of the ABWR design scope. The portions of the system outside the design scope, as well as the design details of the in-scope portion of the system, will be provided by the COL applicant referencing the ABWR design. GE has provided a conceptual design and interface requirements for the out-of-scope portion of the system as required by 10 CFR Part 52.

The NRD system consists of sump pumps, valves, and associated piping and instrumentation to direct waste liquids, valve and pump leakoffs, and component drains and vents to the sanitary drainage portion of the PSW system. SSAR Table 3.2-1 states that the system serves all areas of the plant within the ABWR scope except for the primary containment and the MST. The areas of secondary containment housing the EDGs contain no drains. The system drains by gravity and contains no active components from the point of drainage to the sumps. Valves that are relied upon to prevent backflow can be inspected and tested and are designed to withstand the effects of a SSE. Flooding is prevented by designing the system with sufficient capacity to accommodate expected flooding as well as placing safety-related equipment on raised pads or gratings (see Section 3.4.1 of this report for the staff's review of flood protection). Based on this information, the staff concludes that the system can withstand the effects of natural phenomena and, therefore, meets the requirements of GDC 2.

The NRD system is arranged with separate piping in each quadrant. Flooding or backflow in one quadrant cannot affect other quadrants. This system has no connections to the radioactive drains transfer system, and drains are designed to withstand the adverse effects (including high pressure) associated with pipe and equipment failures in building compartments. Open drainage lines that are needed to maintain an air pressure differential are provided with a water seal. Based on this information, the staff concludes that the NRD system meets GDC 4 as it relates to protection of the system from environmental and dynamic effects associated with piping and components failures.

The NRD system design ensures that radioactive material cannot be discharged. However, effluent is sampled for radiation prior to discharge. In addition, level switches are provided in each sump to monitor leakage (see Section 5.2.5 of this report for the evaluation of leakage detection methods). GE included a COL action item in Amendment 29 of the SSAR requiring COL applicants referencing the ABWR design to develop a sampling and analysis program to ensure that radioactive liquids are not inadvertently being discharged from the nonradioactive drain system. This was identified as DFSER COL Action Item 9.3.3-1. Based on these design and operational provisions, the staff concludes that the system contains adequate means to detect and control the release of radioactive materials in the system effluent and meets the requirements of GDC 60 and is acceptable.

SSAR Amendment 29 includes a conceptual design and interface requirements for the part of the NRD system outside of the ABWR design scope. The COL applicant referencing the ABWR design shall provide the details of the system design from the standard plant buildings to the site discharge structure. The system collects wastewater from plant buildings, precipitation, and other surface runoff and directs this water to dual settling basins where suspended solids are settled and oil is collected. Tests and analyses are performed to meet the plants discharge permit. Failure of this portion of the system will not adversely affect any safety-related equipment. Based on this information, the staff concludes that the out-of-scope portion of the nonradioactive drain system will also meet the requirements of GDC 2, 4, and 60.

The staff concludes that the design criteria, conceptual design, and interface requirements provided in the SSAR will allow the applicant to design a nonradioactive drainage system which will comply with the GDC.

GE submitted the design description, ITAAC, and the relevant interface requirements relating to the NRD system. The adequacy and acceptability of the design description, the ITAAC, and the interface requirements are evaluated in Section 14.3 of this report.

The basis for the staff review has been conformance of the COL applicant's system design and design criteria to the Commission's regulations as set forth in the GDC. The staff concludes that the nonradioactive equipment and floor drain system provides adequate design criteria to ensure compliance with GDC 2, 4, and 60 with respect to protection from natural phenomena, seismic design, environmental conditions, and control of potentially radioactive material, respectively. Therefore, the staff concludes that the design will comply with the guidelines of SRP 9.3.3 and is acceptable.

9.3.4 Chemical and Volume Control System

The ABWR does not include this system.

9.3.5 Standby Liquid Control System

The SLCS was reviewed in accordance with SRP Section 9.3.5 (NUREG-0800). An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed according to the guidelines provided in the "Review Procedures" portion of the SRP. Conformance with the acceptance criteria formed the basis for our evaluation of the SLCS with respect to the applicable regulations of 10 CFR Part 50.

The SLCS is a reactivity control system, which will inject sodium pentaborate solution into the reactor coolant to provide an independent means for shutting down the reactor. The SLCS can bring the reactor from rated power to cold shutdown any time during core life if the normal reactivity control system becomes inoperable. Thus, together with the control rod system, it satisfies the requirements of GDC 26 for reactivity control systems redundancy and capability. (Refer to Section 4.6 of this report for a discussion of reactivity control.)

The system consists of a storage tank, a test tank, and two positive displacement pumps with a motor-operated injection valve at each pump discharge, a motor-operated valve at each pump suction, piping, and controls.

All of the SLCS is located within the secondary containment. The maximum temperature at which the solid material would precipitate from solution is 15 °C (59 °F); the room in which the equipment containing the borated solution is located is kept at a temperature of about 38 °C (100 °F). An electrical resistance heating system maintains the solution between 24 °C and 29 °C (75 °F and 85 °F) to prevent precipitation of the sodium pentaborate solution during storage. Both the high and low liquid level and temperature in the tank are alarmed in the control room.

The two pumps in parallel trains take suction on the storage tank through separate suction lines and discharge it into the reactor vessel through a common injection line.

The liquid is piped into the reactor through the HPCF line downstream of the HPCF inboard check valve. The discharge from each pump is provided with a check valve (to prevent backflow) and a crossover line. Similarly, the piping at the pump suction is also connected by a crossover line. To meet the 10 CFR §50.62 requirements for poison injection, the system is designed for both pumps to start simultaneously and inject 379 L/m (100 gpm) pentaborate solution.

The ABWR SLCS design includes motor-operated ac valves instead of the squib- activated (explosive) injection valves used in current BWRs. On earlier BWR plants, the SLCS piping was not completely isolated from the SLCS storage tank and it was possible for boron to be present in the SLCS piping. Consequently, GE decided to provide leakproof explosive valves so that boron would not leak into the reactor during SLCS testing. In the ABWR SLCS design, the boron storage tank is provided with normally closed isolation valves and a suction pipe keep fill system to prevent the boron solution from entering the SLCS piping. Because of this design change, GE concluded that the leak-tight explosive valves are not required in the ABWR pump discharge piping.

Each pump and its associated valves are powered from a redundant emergency power supply. They are arranged so that failure of a single pump or valve will not prevent adequate amounts of sodium pentaborate solution from entering the reactor vessel to effect shutdown.

The SLCS is automatically initiated after receiving an anticipated transient without scram (ATWS) signal or can be manually actuated by either of two keylocked spring-return switches in the control room. The SLCS system meets the ATWS rule 10 CFR 50.62 because the SLCS pumps are started automatically (see Section 15.5 of this report). Originally, in the DSER SLCS was started manually, which was identified as an open item (DSER Open Item 15). With the SLCS now modified to be automatically initiated, this open item is resolved.

The ATWS initiation signals for SLCS automatic start are high RPV pressure or low RPV water level 2, and startup range neutron monitor (SRNM) ATWS permissive for 3 minutes. The time delay of 3 minutes is provided to allow completion of electric scram which will take about 2 minutes. When the SLCS is automatically initiated to inject the boron into the reactor, the two injection valves and two storage tank discharge valves are opened, the two injection pumps are started, and both pumps run simultaneously. The CUW isolation valves are also closed automatically to prevent a loss of the sodium pentaborate from the vessel.

Turning either key-locked switch in the control panel switch to the "run" position starts an injection pump, opens one motor operated injection valve on the pump discharge, opens a pump suction valve (which is also the tank outlet valve), and closes the reactor cleanup system isolation valves to prevent loss or dilution of boron. If the instrumentation provided indicate that the solution is not entering the reactor vessel, the operator can turn the other key-operated switch to the "run" position to actuate the alternate train.

The SLCS is located in a compartment outside the drywell and below the refueling floor in the seismic Category I secondary containment, which protects the SLCS from floods and tornadoes. All portions of the SLCS necessary to inject sodium pentaborate solution into the reactor are seismic Category I, QG B (or QG A if they are part of the reactor coolant pressure boundary). Thus, the SLCS meets the requirements of GDC 2, and the guidelines of RG 1.29, "Seismic Design Classification," Position C.1.

The secondary containment protects the SLCS against externally or internally generated missiles. The SLCS is separated from non seismic system components and from the effects of breaks in other high and moderate-energy piping systems (see Sections 3.5.1.2 and 3.6.1 of this report). Thus, the SLCS meets the requirements of GDC 4.

To ensure the availability of the SLCS, the system includes two parallel sets of components required to actuate the system (pumps and injection valves). The injection portion of the system can be functionally tested by injecting demineralized water from a test tank into the reactor. To ensure the storage tank discharge valves are reliable, the staff requires COL to confirm that the valves will have adequate reliability requirements and that the valves be incorporated into the operational reliability assurance program (ORAP). This issue is a COL Action Item 9.3.5-1. This was previously identified as Confirmatory Item 9.3.5-1 and the Confirmatory item is resolved.

GE submitted the design description and the inspections, tests, analysis, and acceptance criteria (ITAAC) for the SLCS. This was Open Item 9.3.5-1. GE has provided a revised set of descriptions and ITAAC. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Section 14.3 of this SER. On the basis of this evaluation, this item is resolved.

The SLCS meets the acceptance criteria of SRP Section 9.3.5 and concludes that the SLCS meets the requirements of 10 CFR 50.62 and GDC 2, 4, 26, and 27 as they relate to protected against natural phenomena, system function and redundancy, and testability, and the guidelines of Position C.1. of RG 1.29, as related to seismic classification of the system, and is, therefore, acceptable.

9.3.6 Instrument Air System

Because the IA system is one of the four systems that perform functions addressed in SRP Section 9.3.1, the staff reviewed this system as part of an integrated review of the ABWR CA systems. The results of this review are presented in Section 9.3.1 of this report.

9.3.7 Service Air System

Because the SA system is one of the four systems that performs functions addressed in SRP Section 9.3.1, the staff reviewed this system as part of an integrated review of the ABWR CA systems. The results of this review are presented in Section 9.3.1 of this report.

9.3.8 Radioactive Drain Transfer System

The staff reviewed the DTS in accordance with SRP Section 9.3.3. Staff acceptance is based on meeting the requirements of GDC 2 as it relates to safety-related portions of the system being capable of withstanding the effects of natural phenomena; GDC 4 as it relates to the capability of the system to withstand the effects of the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents and to be compatible with these conditions; and GDC 60 as it relates to providing a means to control the release of radioactive materials in liquid effluent, including anticipated operational occurrences.

The DTS is designated as non-safety-related (except for containment penetrations and piping in the drywell) and designed to collect radioactive or potentially radioactive effluents in equipment or floor sumps and then transfer the effluents to the LWMS for processing. The DTS system is within the scope of the ABWR design certification. The system includes floor drains, drain lines from the equipment to the sumps, the sumps, the two sump pumps for each sump (each 100-percent capacity), sump instrumentation, and piping and valves from the sumps to the radwaste system.

The drains, piping, pumps, instrumentation, and valves of the DTS system are classified as non-safety-related, except for the containment (drywell) penetrations and containment isolation valves, which are safety Class 2, and designed in accordance with seismic Category I and QG B criteria, and the reactor building penetrations that meet ASME Code III, Section 3, requirements. In DSER (SECY-91-355) Open Item 76, the staff noted a discrepancy in the identification of these containment isolation valves between Figure 11-2 and Table 6.2-7. Figure 11-2 identified these valves as air-operated valves, while Table 6.2-7 showed these valves as motor-operated valves. In SSAR Amendment 20, Figure 11-2 was modified to indicate that these valves are motor-operated valves. This modification resolved DSER (SECY-91-355) Open Item 76.

All system piping is designed to remain intact following a seismic event. The drain system is not the only method of leak detection available for any of the areas served by the system and this method is not considered in the facility flood analysis. However, the staff indicated in DSER (SECY-91-355) Open Item 77 that the check valves that provide backflow protection for sumps in the ECCS equipment rooms should be classified as safety Class 3 and designed to seismic Category I and QG C criteria. GE committed to revise SSAR Table 3.2-1 and Figure 11-2 to include the designation of these valves as being non-safety designed to seismic Category I and QG C criteria. This information resolved DSER Open Item 77, however, incorporation of the information into the SSAR was identified as DFSER Confirmatory Item 9.3.8-1. The staff reviewed Amendment 26 of the SSAR and concludes that GE modified Section K11 of Table 3.2-1 to require that the check valves be non-safety class and meet seismic Category I and QG C requirements. GE modified Section 9.3.8.1.1 to require that the check valves be inspectable, testable, and able to withstand an SSE. The staff finds that these modifications ensure that the check valves will provide adequate protection from backflow into areas containing safety-related equipment. DFSER Confirmatory Item 9.3.8-1 is resolved.

All drywell sumps are automatically isolated on a LOCA signal to prevent the uncontrolled release of primary coolant outside primary containment.

In the DSER (SECY-91-355), the staff discussed two additional items in its review of the systems and components shown in SSAR Figures 11.2-1 and 11.2-2. First, Figure 11.2-1 showed the shower facility discharging into the high conductivity waste collector tank. In all other figures, and in the text, the shower facility discharges to the hot shower drain (HSD) receiver tank. Second, GE did not identify the points at which changes in component qualification requirements occur (e.g., for the containment isolation valves described in the preceding paragraphs). The staff designated resolution of the discrepancy in Figure 11.2-1 and the addition of component qualification requirements to the figures as Open Item 78.

SSAR Amendment 20 revised the figure to show the shower facility discharging into the HSD receiver tank. In addition, the staff reviewed Amendment 27 of the SSAR and concludes that GE clarified the description of the containment (drywell) sump isolation valves. The valves meet seismic Category I, safety Class 2, and QG B, requirements.

The staff noted in the DSER (SECY-91-355) that a GE interface requirement in the SSAR prevents connections between radioactive and nonradioactive systems. Upon further evaluation, the staff determined that this requirement can be accomplished by identifying a COL Action Item requiring this information. This was identified as DFSER COL Action Item 9.3.8-1. GE has included this information in the SSAR. This is acceptable.

IE Bulletin No. 80-10 identified an issue concerning the potential contamination of nonradioactive systems which could result in unmonitored, uncontrolled releases of radioactivity to the environment. The ABWR design employs several methods to prevent such an occurrence. The methods include ensuring that no cross-connections exist between nonradioactive and potentially radioactive systems (such as with the DTS system), by providing sampling and monitoring of both radioactive and nonradioactive effluents before discharge, and by separating nonradioactive systems from potentially radioactive systems by barriers, along with radiation monitors to detect leakage across the barrier (such as with the RCW system). These methods are discussed in detail in various SSAR sections (e.g., 9.2.11 and 11.5) and are reviewed in various sections of this report (e.g., 9.2.11 and 11.5). These methods ensure that all discharges to the environment are monitored and controlled. Based on this information, the staff concludes that the ABWR design has provided sufficient design features and design requirements to ensure that the issues discussed in IE Bulletin No. 80-10 are resolved.

In the DSER (SECY-91-355), the staff discussed a GE interface requirement for monitoring the effluent from nonradioactive systems before discharge to ensure that no unacceptable (radioactive) effluents are discharged from the nonradioactive drain systems. This requirement for monitoring nonradioactive effluents will allow the COL applicant to design a NRD system that will satisfy the requirements of GDC 60 (the nonradioactive drain system is evaluated in Section 9.3.3 of this report). Upon further evaluation, the staff has determined that this requirement can be accomplished by establishing a COL Action Item requiring this information. This was identified as DFSER COL Action Item 9.3.8-2. GE included this information in the SSAR, which is acceptable.

The staff stated in the DSER (SECY-91-355) that SSAR Section 9.3.8.2 inaccurately referred to SSAR Section 9.3.9.1, when the interface requirements discussed in SSAR Section 9.3.12 would be the appropriate reference. The staff stated that GE should revise the first design basis discussed in SSAR Section 9.3.8.1 to clearly indicate, consistent with the staff's dialogue with GE, that only portions of the drain system are considered safety-related. GE corrected the SSAR reference and revised SSAR Section 9.3.8.1, which resolved DSER Open Item 79. Incorporation of the revisions in the SSAR was identified as DFSER Confirmatory Item 9.3.8-2. The staff reviewed Amendment 29 of the SSAR and determined that the required corrections were made. The staff determined that the classifications in Section K1 of Table 3.2-1 clarify which parts of the system are safety-related and which are not. The staff has determined that these classifications will ensure that the check valves will provide adequate backflow protection. Therefore, the staff finds these modifications acceptable. Therefore, DFSER Confirmatory Item 9.3.8-2 is resolved.

GE submitted the design description and the ITAAC relating to the DTS system. This was identified in the DFSER as Open Item 9.3.8-1. GE provided a revised set of design description 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, DFSER Open Item 9.3.8-1 is resolved.

The staff concludes that the design of the RD system is acceptable and conforms to the requirements of GDC 2, 4, and 60 for protection against natural phenomena, environmental conditions, missiles, and the release of radioactivity to the environment. Therefore, the staff finds that the design complies with guidelines of SRP Section 9.3.3, and is acceptable.

9.3.9 Hydrogen Water Chemistry System

Hydrogen water chemistry (HWC) reduces intergranular stress corrosion cracking (IGSCC) by using feedwater additions of hydrogen to decrease the oxidizing power of water and reduce its aggressiveness toward plant material. To suppress IGSCC, reactor coolant conductivity must be maintained below 0.3 micro-Siemens per centimeter and sufficient hydrogen must be added to the feedwater to reduce the electro-chemical potential below -0.23 volts (Standard Hydrogen Electrode). These conditions are specified in EPRI Report NP-4947-SR, "BWR Hydrogen Water Chemistry Guidelines: 1987 Revision," October 1988. SSAR Section 9.3.9 references this report and commits to use its guidelines for design and operation of the HWC system.

Section 9.3.9 of the SSAR also addresses the means of storing and handling of hydrogen. These operations will be performed in accordance with the recommendations of EPRI Report NP-5283-SR-A, "Guidelines for Permanent BWR Hydrogen Water Chemistry Installations."

The staff finds that the EPRI guidelines presented in these reports describe a satisfactory means for storing and handling hydrogen for the ABWR, and the GE commitments in the SSAR to the EPRI guidelines are thus acceptable. The staff concludes that DSER (SECY-91-355) Open Item 100 is resolved.

Certified Design Material

GE has proposed ITAAC and included Tier 1 design description of the hydrogen water chemistry system in Section 2.11.17 of the CDM. This was identified as DFSER Open Item 9.3.9-1. The adequacy and acceptability of GE's design descriptions and ITAAC are evaluated in Section 14.3 of this report. On the basis of this evaluation, DFSER Open Item 9.3.9-1 is resolved.

9.3.10 Oxygen Injection System

The oxygen injection system is designed to add sufficient oxygen (20 to 50 ppb) to suppress erosion/corrosion, general corrosion, and the release of corrosion products in the condensate and feedwater systems. The requirements for design, operation, maintenance, surveillance, and testing of the oxygen storage facility are specified in EPRI Report NP-5283-SR-A. The staff finds that the EPRI guidelines describe a satisfactory means for storing and handling oxygen for the ABWR, and are thus acceptable. Section 9.3.10 of the SSAR references this report and commits to use these guidelines. The staff concludes that DSER (SECY-91-355) Open Item 101 is resolved.

Certified Design Material

GE had not proposed ITAAC nor a Tier 1 design description of the oxygen injection system in Section 2.11.7 of the CDM. This was DFSER Open Item 9.3.10-1.
The adequacy and acceptability of GE's design descriptions and ITAAC are evaluated in Section 14.3 of this report. On the basis of this evaluation, DFSER Open Item 9.3.10-1 is resolved.

9.3.11 Zinc Injection System

The control of build-up of radiation in reactor systems has been of concern in BWR plants. GE found that operating BWR plants having 5 to 15 ppb of soluble zinc in the reactor water had lower piping dose rates than plants that had only trace amounts of zinc.

Laboratory tests confirmed that Co-60 deposition is greatly reduced in both normal and hydrogen water chemistry with the presence of soluble zinc. Zinc injection into the feedwater system to provide reactor water concentrations of 10 to 15 ppb zinc during initial conditioning and 5 to 10 ppb over the fuel cycle will help keep radiation levels as low as possible; thereby, reducing personnel exposure especially during outages.

Based on the foregoing, the staff concludes that Section 9.3.11 of the ABWR SSAR is acceptable.

Certified Design Material

At the time of the DFSER, GE had not proposed ITAAC nor a Tier 1 design description of the zinc injection system in Section 2.11.18 of the CDM. This was DFSER Open Item 9.3.11-1. GE has provided a final version of the CDM and it has been evaluated in Chapter 14.3 of this report. Based on this evaluation, DFSER Open Item 9.3.11-1 is resolved.

9.4 Heating, Ventilation, and Air Conditioning Systems

Certified Design Material

GE submitted in Sections 2.15.5 of the CDM Tier 1 design description and the ITAAC for HVAC systems which were under staff review. The results of the staff's review were to be provided in the FSER. This was DFSER Open Item 9.4-1. GE provided a revised set of design description and ITAAC on August 31, 1993. 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 open item is resolved.

9.4.1 Control Building Heating, Ventilating, and Air Conditioning System

The staff reviewed the control building HVAC systems in accordance with SRP Section 9.4.1. The system consists of two separate HVAC systems: one for the MCR habitability area and another for the control building safety-related equipment area. The staff evaluated these systems as described in Sections 9.4.1.1 and 9.4.1.2, respectively, of this report, and are evaluated for compliance with the SRP acceptance criteria as discussed below.

9.4.1.1 Control Room Habitability Area Heating, Ventilating, and Air Conditioning System (CRHA HVACS)

The CRHA HVAC system serves the main control area envelope (MCAE) containing the control room proper (including the critical document file), computer room, control equipment room, upper and lower corridors, office and chart room, instrument repair room, and sleeping area, as described in SSAR Section 6.4.2.1. The MCAE is maintained at a minimum positive pressure of 3.2 mm

(1/8 in.) of water gauge relative to the surrounding spaces with outside makeup air of not more than 360 m^3/hr (212 cfm) @ 760 mm Hg (30 in.), 0 °C (32 °F). The kitchen and lunch rooms, men's lavatory, and women's lavatory and lounge facilities are in the Service Building.

The system consists of two fully redundant trains of equipment, including ductwork, air handling units (AHUs), control dampers, fire zone dampers, and supply and exhaust fans. Each AHU consists of a bag-type filter, a chilled-water cooling coil, an electric heating coil, and a humidifier as described in SSAR Section 9.4.1.1.3. Each train also incorporates an emergency filtration unit (EFU) which is an ESF. The EFU is comprised of an electric heating coil, pre-filter, pre-high efficiency particulate air (HEPA) filter, charcoal adsorber (50 mm (2 in.) deep, as a minimum), post-HEPA filter, and two circulating fans. Independent and separate discharge to and return from the MCAE is provided to each EFU. Chilled water to the cooling coils is supplied by the HVAC emergency cooling water (HECW) system. The CRHA HVAC system is designed to maintain a controlled temperature environment under normal and accident conditions, and provides for detection and removal of smoke and filtration of radioactive material. The range of design conditions for the MCAE is 21 °C through 26 °C (70 °F - 79 °F) and 10 percent to 60 percent relative humidity. SSAR Section 6.4.4.2 discusses the protection from exterior smoke, toxic chemicals, and chlorine releases.

Normally, one AHU, one supply CRHA HVAC system fan, and one exhaust fan are in operation. During normal operation, the CRHA HVAC system performs HVAC functions and pressurizes the MCAE using a combination of filtered outdoor air and recirculated indoor air. The combined air stream is passed through an AHU. Two parallel, 100-percent capacity supply fans (one standby) draw air from the instrument panel areas, corridors, MCR, computer room, office areas, and the switch and tag room, and return it to the AHU. The exhaust fan starts automatically when the supply fan starts. Two parallel, 100-percent exhaust fans (one standby), controlled by a pressure controller, draw air from the areas and exhaust it to the environment to maintain positive pressure of 3.2 mm (1/8 in.) of water gauge relative to the surrounding spaces. The pressure controller is located in the instrument panel area of the MCR. The supply and return ducts have manual balancing dampers that are locked in place after the system is balanced. The supply and return ducts also have modulating dampers to maintain the required positive pressure. Sufficient air is provided to pressurize the control room equipment HVAC envelope. As described in SSAR Section 6.4.2.1, the control room area envelope is maintained at a positive pressure of 3.2 mm through 6.4 mm (1/8 in. through 1/4 in.) water gauge with respect to the surrounding spaces at all times. The CRHA HVAC system flow diagram is shown in SSAR Figure 9.4-1, Sheets 1 and 2, flow rates are given in SSAR Table 9.4-3, and component descriptions are given in SSAR Tables 9.4-4 and 9.4-4a through 9.4-4c.

GE submitted a draft of a revised SSAR Section 9.4.1.1.4 which addresses IE Bulletin 80-03 compliance by stating that the charcoal tray and screen will be of all welded construction to preclude the potential loss of charcoal from adsorber cells per this bulletin. Therefore, the emergency air filtration system of the CRHA HVAC system precludes the potential loss of charcoal from adsorber cells as reflected in SSAR Amendment 34.

The CRHA HVAC system and components are located in a seismic Category I control building that is protected against tornado missiles and floods and are operable during LOOP. All essential control room HVAC equipment, including ductwork, is of seismic Category I design. SSAR Table 3.2-1 states that those nonessential portions of the system which, by their failure during a seismic event can affect safety-related portions, of the system are designed in accordance with Position C.2 of RG 1.29. Outside air intake valves are protected against freezing and other environmental conditions, and tornado missile barriers protect the intake vents. The CRHA HVAC system meets the requirements of GDC 2 by complying with the provisions of Position C.1 of RG 1.29 for the essential portions, and Position C.2 for the applicable nonessential portions.

SSAR Figures 1.2-15 and 1.2-21 show that the redundant CRHA HVAC system trains are located in separate rooms. In SSAR Amendment 7, GE stated that the walls, floor, and ceiling of the room that houses each safety-related system act as missile barriers or shields from missiles generated outside the room. In SSAR Amendment 17, GE stated that non-safety-related components are arranged in such a way that any missile-generating component is in a separate room away from safety-related components. For these reasons, the staff concludes that the redundant CRHA HVAC system trains are protected against internally-generated missiles. The CRHA HVAC system complies with GDC 4 with respect to protection of the system against environmental and dynamic effects, as discussed in Section 3.6.1 of this report regarding the effect of postulated piping failures outside the containment on the system. Therefore, the first part of DSER Open Item 60 which deals with staff's concern relating to compliance of this system with the requirements of GDC 4, is resolved.

Smoke detectors in the MCAE will actuate an alarm on detection of smoke when smoke is detected in a division of the CRHA HVAC system, MCR operators will manually switch the appropriate divisional HVAC system to a smoke removal mode, in which the exhaust fan is stopped, the recirculation damper is closed, and the exhaust bypass damper is opened. The MCR operators can exhaust 100 percent of the conditioned air to the atmosphere by manually activating a switch, which closes return dampers and opens exhaust dampers. In SSAR Amendment 32, GE stated that the CRHA HVAC system fire dampers are equipped with fusible links and are capable of closing under anticipated air flow conditions after the fusible link melts. Tests will be performed to verify these conditions at a test facility.

In the DSER (SECY-91-235), the staff stated the system design meets the control room habitability requirements of TMI Action Item III.D.3.4 of NUREG-0737 with regard to smoke removal, contingent upon GE submitting revised P&IDs showing smoke detection capability (second part of DSER Open Item 60). The staff stated the above contingency, since the earlier P&IDs for the control room and the control equipment room did not include smoke detectors at the air intakes. However, GE committed to place smoke detectors at the air intakes which will cause the control room HVAC system to shift to the recirculation mode, if smoke is detected in the air supply. Since this commitment resolved the second part of DSER Open Item 60, the staff classified the SSAR commitment as DFSER Confirmatory Item 9.4.1.1-1. SSAR amendments through Amendment 32 included revised P&IDs which show the smoke detectors at the air intakes. Therefore, this confirmatory item is resolved.

The evaluation of requirements for protection against hazardous chemical releases will be site specific as stated in Section 6.4 of this report. Therefore, the staff stated in the DFSER that COL applicants should provide information regarding compliance with RG 1.78, "Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release," and RG 1.95, "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release," if the potential for hazardous chemical, including chlorine, release exists at their sites. The staff further stated in the DFSER that COL applicants should identify plant design features that will automatically isolate the control room from the outside when necessary, and thus protect the control room operators from toxic substances in the plant vicinity. The need for providing the above information was identified as DFSER COL Action Item 9.4.1.1-1. By SSAR amendments up to and including Amendment 32, GE identified COL license information (Section 6.4.7.3) which states that the design acceptance is based upon meeting the guidelines of the subject guides.

A four-channel radiation monitoring system is provided in the outside air intake ducts. A radiation monitor is provided in the MCR to monitor MCR area radiation levels. These monitors alarm in the MCR upon detection of high radiation levels. On receiving a high radiation signal, the redundant normal outside air intake dampers and exhaust dampers close, the emergency outside makeup air intake dampers open, and only the corresponding EFU of the operating division starts automatically. Also, the emergency recirculation mode can be started manually. In the emergency recirculation mode, the makeup air and part of the return air pass through the ESF-grade filter train, which filters the radioactive particles and iodine to keep the control room operator exposures below the GDC 19 limit of 5 rem whole body or its equivalent for the duration of a design-basis accident (DBA). The control room equipment HVAC system thus meets the requirements of GDC 19 regarding protection of the control room. For further information on system compliance with GDC 19 limits, see Section 6.4 of this report. Radioactivity monitors at the inlets allow operator selection of inlet vent manually. Connections are provided for testing the HEPA and charcoal filters pressure drops and radioiodine removal efficiency. A differential pressure switch across the emergency filtration system alarms on high pressure. A flow switch in the emergency filtration fan discharge duct initiates an alarm if the fan fails and automatically starts the standby system.

After the May 5, 1992 meeting between the NRC and GE, in a telephone conversation on May 12, 1992, GE committed to provide SSAR appendices identifying the compliance of (1) the ESF-grade filter train in the system with RG 1.52 guidelines and (2) control room filter train instrumentation with the minimum instrumentation requirements listed in SRP Table 6.5.1-1. These appendices were to be similar to those provided for the SGTS. GE provided the information in the SSAR, Appendices 9C and 9D. The staff reviewed GE's justifications for compliance with RG 1.52 and NUREG-0800, Standard Review Plan, Table 6.5.1-1, "Minimum Instrumentation, Readout, Recording and Alarm, Provisions for ESF Atmosphere Cleanup Systems," for the CRHA HVAC system and found that the proposed RG 1.52 conformance and SRP Table 6.5.1-1 instrumentation and controls in the MCR, as well as selected local panels, meet the intent of the above guidance. While the staff would prefer that GE provide all of the local instrumentation and controls described in the above guidance for operational efficiency and convenience, the staff finds that the CRHA HVAC system instrumentations and controls meet the intent of the above guidance based on the following:

1. The instrumentation and controls (i.e., readout, recordings, and alarms) provisions for the related

parameters (i.e., pressure drop, temperature, humidity, flow) are available and would be made accessible in the MCR.

2. The COL applicant will prepare plant-specific procedures to respond to the instrumentation and controls.

3. The filtration train fully complies with the regulatory position of RG 1.52.

4. The acceptance, pre-operational, and surveillance testing meet the intent of ASME Codes N509 and N510.

5. GE revised SSAR Sections 9.4.1.1.4 and 9.1.1.1.5, stating that the unfiltered in-leakage is controlled by the use of welded ducts, except galvanized steel is used for outdoor air intake and exhaust ducts, and unfiltered in-leakage testing will be performed periodically on all system ductwork outside MCAE, in accordance with ASME Code N510.

6. CRHA HVAC system dose analysis is bounded by the minimum outside air intake flow, and any higher flow will be filtered through the EFUs.

Based on the above, the staff has determined that the CRHA HVAC system's EFUs will have a creditable removal efficiency of 95 percent for all forms of radio-iodine.

In DFSER Open Item 9.4.1.1-1, the staff stated that the SSAR did not specify heaters to maintain proper humidity (less than 70 percent) to ensure proper operation of the charcoal adsorbers. GE revised SSAR Section 9.4.1.1 and Table 9.4-4 to include electric heaters in the ESF filter trains. Therefore, this item is resolved.

Based on the above, the staff concludes that the control room habitability area HVAC system complies with GDC 2, 4, 19, and 60 as they relate to protection against natural phenomena, maintaining proper environmental limits for equipment operation, and protecting those who access the control room under accidental radioactive releases. This system also complies with the guidelines of RGs 1.29 and 1.52 relating to the system seismic classification and system capability to clean the control room atmosphere during emergency operation, and thereby meets the applicable acceptance criteria of SRP Section 9.4.1. Therefore, the staff concludes that the system is acceptable.

9.4.1.2 Control Building Safety-Related Equipment Area (CBSREA) HVAC System

The ABWR design includes three similar, but independent, HVAC systems to service the safety-related electrical, HVAC, and mechanical equipment (including the refrigerator, pump, and heat exchanger equipment) of Divisions A, B, and C of the CBSREA HVAC system. Safety-related areas serviced by each system include a safety-related battery room(s), HECW refrigerator room, reactor building cooling water pump and heat exchanger room, HVAC equipment room, and safety-related electrical equipment room. The systems differ with regard to safety-related areas serviced, in that the system serving Division B includes two safety-related battery rooms, while the systems serving Divisions A and C include one safety-related battery room each. Each divisional system serves non-safety-related passages. In addition, Division A serves a non-essential battery room, and a non-essential electrical equipment room, and Division C serves the non-safety-related MG sets. Each divisional system draws outside air through a normal air supply unit, and system exhaust fans discharge to the atmosphere. Chilled water to the cooling coil is supplied by the emergency cooling water (HECW) system. Each supply unit consists of a bag filter and cooling coil. Each divisional system contains an air supply unit with two 100-percent capacity supply fans and exhaust fans. The supply and exhaust fans of each system (Divisions A, B, and C) are powered from the corresponding Class 1E Electrical Divisions I, II, or III. One of the two supply fans is started manually for normal operation. When the ACU supply fan is started, one of the two exhaust fans start automatically. The exhaust fans can be started manually to exhaust the air to the atmosphere as needed such as to remove hydrogen and smoke.

In normal operating mode, one ACU, one supply fan, and one exhaust fan in each division operate. The HVAC system includes fire dampers for the supply and exhaust duct penetrations of firewalls for Division IV safety-related electrical equipment room and battery room, served by Division B of CBSREA HVAC system. The smoke removal mode is manually initiated by closing the recirculation damper, closing the exhaust fan, and opening the exhaust fan bypass damper to allow outside air purging of the affected area. The smoke removal mode of the HVAC systems is discussed in CBSREA Section 9.5.1 of this report. SSAR Amendment 32 stated that the CBSREA HVAC system fire dampers separating safety-related electrical equipment and battery Divisions II and IV rooms use fusible links in HVAC ductwork which will close under anticipated air flow conditions after the fusible link melts, and that tests will be performed to verify these conditions at a test facility.

The HVAC subsystems and components are located in a seismic Category I structure that is tornado missile-protected and flood-protected. The essential components of each system are of seismic Category I design. The low-pressure safety-grade ESF ductwork is designed to withstand the maximum positive pressure to which it can be subjected under normal or abnormal conditions. The outdoor air intake and exhaust are conducted of galvanized steel, and all other ducts are of welded black steel construction. Tornado missile barriers protect the air intake and exhaust vents. Applicable nonessential portions of the system which, by their failure during a seismic event can effect safety-related portions, are designed in accordance with Position C.2 of RG 1.29, and as stated in the SSAR Table 3.2-1. The systems meet the requirements of GDC 2 regarding protection from natural phenomena by complying with the provisions of Positions C.1 of RG 1.29 for the essential portions and Position C.2 for the applicable nonessential portions.

Each system is designed to maintain hydrogen concentration below 2 percent by volume in the battery rooms, and temperature controllers regulate heating and cooling to maintain conditions within pre-set limits. SSAR Figures 1.2-15 and 1.2-21 show that the divisional CBSREA HVAC systems are located in separate rooms. SSAR Amendment 7 stated that the walls, floor, and ceiling of the room that houses each safety-related system act as missile barriers or shields from missiles generated outside the room. SSAR Amendment 17 stated that non-safety-related components are arranged in such a way that any missile-generating component is in a separate room away from safety-related components. For these reasons, the staff concludes that the redundant essential electrical equipment HVAC systems are protected against internally-generated missiles. The CBSREA HVAC systems comply with GDC 4 with respect to protection of the system against environmental and dynamic effects (see Section 3.6.1 of this report for an evaluation of the effect of postulated piping failures outside the containment on the system).

In DSER (SECY-91-235) Open Item 61, the staff stated that the RCW pump rooms are located in the secondary containment and that the design of the HVAC system as constituted did not account for the potential for radioactivity contamination following an accident. The staff re-examined this issue and finds that SSAR Figure 1.2-15 shows that the subject rooms serviced by the HVAC system are in the control building and are unlikely to be contaminated to the extent that the exhaust from the subject areas has to be filtered. The staff further discusses this in Section 11.5 of this report and concludes that no monitoring of this exhaust is required. Therefore, this resolved DSER Open Item 61.

The rooms served by this system are not occupied under accident conditions and the system is not reviewed against the requirements of GDC 19 regarding protection of the control room and the provisions of RG 1.78 and RG 1.95.

The staff concludes that the system complies with GDC 2 and 4 as they relate to protection against natural phenomena and maintaining proper environmental limits for equipment operation, and with the guidelines of RG 1.29 relating to the system seismic classification, and thereby meets the acceptance criteria of SRP Section 9.4.1. Therefore, the staff concludes that the CBSREA HVAC system is design acceptable.

9.4.2 Spent Fuel Pool Area Ventilation System

The spent fuel pool area ventilation system is part of the reactor building ventilation system and is evaluated in Section 9.4.5 of this report.

9.4.3 Auxiliary Area Ventilation System

The auxiliary area ventilation system is part of the reactor building ventilation system and is evaluated in Section 9.4.5 of this report.

9.4.4 Turbine Island HVAC System

The staff reviewed the turbine island ventilation system in accordance with SRP Section 9.4.4. The system consists of the turbine building (TB) HVAC system and the electrical building (EB) HVAC system. The TB HVAC and EB HVAC systems supply filtered and tempered air to all turbine island spaces during normal, plant startup, and shutdown operations. The TB HVAC and EB HVAC systems do not serve or support any safety function and have no safety design basis. The system description, equipment list, and figures are given in SSAR Section 9.4.4, Tables 9.4-5 and 9.4-5a through 9.4-5c, and Figures 9.4-2a through 2c. Failure of the system does not compromise the operation of essential systems, the ability to shut down the reactor, or result in unacceptable releases of radioactivity. Both the TB HVAC and EB HVAC systems comply with GDC 2 because the applicable nonessential portions of the systems are designed in accordance with Position C.2 of RG 1.29 as stated in SSAR Table 3.2-1. The requirements of GDC 60 for control of releases of radioactive materials to the environment do not apply because the ventilation exhausts of these systems do not include HEPA filters or charcoal adsorbers. The TBVS exhausts, including turbine building compartment exhaust and lube oil areas exhaust (SSAR Figures 9.4-2a through and 2c), will be monitored at the plant vent. The EB HVAC system will exhaust directly to the atmosphere (for additional information on effluent monitoring, see Section 11.5 of this report). The staff concludes that the turbine island HVAC system design complies with the applicable GDC referenced in SRP Section 9.4.4 and consequently with the subject SRP criteria and is, therefore, acceptable.

9.4.5 Reactor Building Ventilation System

The reactor building ventilation system (RBVS) consists of eight subsystems: (1) reactor building (RB) secondary containment HVAC system, (2) RB safety-related equipment HVAC system, (3) RB non-safety-related equipment HVAC system, (4) RB safety-related electrical equipment HVAC system, (5) RB safety-related diesel generator HVAC system, (6) RB primary containment supply exhaust system, (7) RB main steam tunnel (MST) HVAC system, and (8) RB reactor internal pump (RIP) power supply panel adjustable speed drive (ASD) HVAC system.

The RBVS cools all areas in the reactor building serviced by these subsystems except the area serviced by the RB primary containment supply/exhaust system (cooling for this area is evaluated in Section 9.4.9 of this report). Certain subsystems including the RB secondary containment HVAC system, RB safety-related electrical equipment HVAC system, RB safety-related diesel generator HVAC system, and the primary containment supply/exhaust system have outside air intakes and exhaust air from the areas serviced by these subsystems. The HVAC systems are reviewed in accordance with SRP Section 9.4.5, as applicable, for ESF ventilation systems. The staff reviewed the RB safety-related equipment HVAC system in accordance with SRP Section 9.4.2, the subject HVAC system also services the spent fuel pool area.

9.4.5.1 RB Secondary Containment HVAC System

In the DSER (SECY-91-235), the staff included only a partial evaluation of the system, stating that the evaluation would be completed on receipt of the system P&IDs and interface descriptions. The staff designated this lack of information as Open Item 62 in the DSER (SECY-91-235). SSAR amendments through Amendment 34 included the above information in revised SSAR Section 9.4.5.1, Figure 9.4-3, and Tables 9.4-3, 9.4-4f through 9.4-4h, 11.5-1, and 11.5-2. Therefore, the DSER Open Item 62 is resolved based on the staff's evaluation given below. SSAR Section 9.5.1.1.6 describes the smoke control mode of the secondary containment HVAC system. In Section 9.5.1.4.4 of this report, the staff evaluated the HVAC system design for smoke control in the secondary containment.

The RB secondary containment HVAC system provides the HVAC needs of the secondary containment area of the reactor building. The system provides the HVAC needs of the safety-related equipment areas in the secondary containment, during normal operation. The system also provides the HVAC needs of the non-safety-related equipment areas in the secondary containment in conjunction with the RB safety-related equipment HVAC system during normal operation. The system is a once-through type. Outdoor air is filtered (bag-type filter), tempered, and delivered to the secondary containment areas. The exhaust air from all the areas serviced by the system is filtered (bag-type) and discharged to the monitored plant vent. The supply system consists of a bag-type filter, a cooling/heating coil, and three 50-percent supply fans, two of which operate normally (one is standby). The exhaust system also has three fans (one standby), each of which is provided with a bag-type filter. The system includes radiation monitors for air exhaust through the refueling floor area ducting, and the common secondary containment area ducting which provide input signals to process radiation monitoring system. Isolation dampers automatically isolate the system and the SGTS is initiated on (1) secondary containment exhaust high radiation signal, (2) refueling floor high radiation signal, (3) LOCA signal, or (4) loss of secondary containment HVAC supply/exhaust fans signal (for SGTS evaluation, see Section 6.5.1 of this report). The secondary containment is maintained at a negative pressure with respect to the surrounding spaces during normal operation by the system and during accident conditions by the safety-related SGTS. The supply fans will operate only when the exhaust fans are operating. Fire dampers with fusible links in the HVAC duct work will close under anticipated air flow conditions after the fusible links melt. The system is started manually.

The secondary containment HVAC contains no equipment needed to maintain the integrity of the reactor coolant pressure boundary or to ensure the safe shutdown of the reactor. The interface of the secondary containment HVAC system with the SGTS, which provides radioactivity filtering capability for containment releases, does include components (e.g., isolation devices and monitors) whose function is required to maintain releases below acceptable levels.

The system is located in the seismic Category I reactor building, which protects against flood and tornado missiles. Although the system is non-safety-related, SSAR Table 3.2-1 states that the radiation monitors and the inboard and outboard dampers for isolating the system under situations identified above are designed to seismic Category I requirements and that the applicable nonessential portions are designed in accordance with

Position C.2 of RG 1.29. Thus, the system complies with GDC 2 by meeting Position C.1 of RG 1.29 for the essential portions and Position C.2 for the nonessential portions. The inboard and outboard isolation dampers are pneumatically operated and they fail to the closed position in the event of loss of pneumatic pressure or loss of power to the actuating solenoids. The safety-related components are protected against damage from internally-generated missiles by separation of redundant equipment and are also protected against piping failures. Thus, the essential components of the system comply with GDC 4 regarding protection from the environment and dynamic effects associated with equipment failures. GDC 17 is not applicable to the system because the system is not safety related. However, because the secondary containment HVAC system air intake is filtered, the system meets GDC 17 requirements as it relates to protection of the electrical components against accumulation of dust and particulates.

The system complies with GDC 60 regarding the control of releases of radioactive materials to the environment because under certain situations, the system is automatically isolated to prevent significant radioactivity release to the environment. When isolated, the system discharges to the SGTS, which will filter any release to the environment.

Based on the above, the staff concludes that the system design complies with applicable GDC referenced in SRP Section 9.4.5 and consequently with the subject SRP acceptance criteria, and, therefore, is acceptable.

9.4.5.2 RB Safety-Related Equipment HVAC System

The system cools the following safety-related equipment areas in the reactor building during accident conditions: the RHR pump rooms, HPCF pump rooms, RCIC pump room, FCS equipment rooms, SGTS equipment rooms, and the CAMS equipment rooms. As stated in Section 9.4.5.1 of this report, the RB secondary equipment HVAC system cools the above areas under normal conditions. Increased and reliable cooling of these rooms will be required during accident conditions when the equipment in the rooms operate. The system is designed to provide a controlled temperature environment in the rooms in accordance with equipment temperature qualification. Separate safety-related fan coil units (FCUs), one for each room serve the 12 rooms that house the redundant components of the five systems mentioned above and the RCIC pump room. The pump room FCUs are automatically initiated when RHR pumps, HPCF pumps, and RCIC turbines are started. The safety-related FCUs in other rooms are automatically initiated upon isolation of the RB secondary containment HVAC system. The FCS room FCUs are initiated upon receiving a manual FCS start signal.

All components of the RB safety-related equipment HVAC system are designed to ESF requirements and are safety-related. The units are open-ended and continuously recirculate cooling air within the space served. Divisional RCW is the cooling medium for the associated cooling coils. Space temperatures in the pump rooms are maintained less than 66 °C (150 °F) during pump operation. Space temperatures in all rooms, including the pump rooms are normally maintained less than 40 °C (104 °F). The units will receive power from the same divisional power source as that for the equipment being served. The design of the FCUs includes both remote and local manual override of the system controls. The system description, figure, and tables are listed in SSAR Section 9.4.5.2, Figure 9.4-3, and Table 9.4-4e, respectively.

All components of the RB safety-related equipment HVAC system (including associated cables) serving safety-related functions are located in seismic Category I structures that protect against flood and tornado missiles. The components of this safety-related system are designed to seismic Category I requirements as follows from SSAR Table 3.2-1. Thus, the system meets Position C.1 of RG 1.29, and consequently complies with GDC 2 regarding protection from natural phenomena.

In the DSER (SECY-91-235), the staff included only a partial evaluation of the system for compliance with GDC 4 as it relates to protection of the system against internally-generated missiles and piping failures. The staff stated that it would complete the evaluation upon receiving system P&IDs and clarifying information on system compliance with GDC 4. The DSER (SECY-91-235) designated the above lack of information as Open Item 63. SSAR amendments through Amendment 34 included the required information, which the staff reviewed. The staff finds that the design satisfies the requirements of GDC 4 for protection against the effects of piping failures and internally-generated missiles by compartmentalization and separation. (For additional information, see Section 3.6.1 of this report, SSAR Section 3.6.4, and SSAR Table 3.6-2.) This resolved DSER Open Item 63.

This internally recirculating cooling system has no supply air from the outside or exhaust air from the equipment areas to the outside when the system operates. The system operates only after the secondary containment HVAC system is isolated. The normal RB secondary containment HVAC system air intake is filtered. Therefore, the system meets GDC 17 as it relates to protection of the electrical components of the system against accumulation of dust and particulates. The system does not include any HEPA filters or charcoal adsorbers. Therefore, GDC 60, which pertains to the control of radioactive material released to the outside, does not apply to the system. Position C.4 of RG 1.13, (an acceptance criterion for SRP Section 9.4.2), which deals with the ventilation and filtration basis for the release of radioactive materials to the environment from the spent fuel storage facility, does not apply to that part of the essential equipment HVAC system that services the fuel pool cooling equipment rooms. The SGTS that serves the fuel pool area during a DBA, including the fuel-handling accident, meets Position C.4 of RG 1.13 and, therefore, complies with GDC 61. For further information, see Section 6.5.1 of this report.

The staff concludes that the system complies with applicable GDC referenced in SRP Section 9.4.5 and consequently with the subject SRP acceptance criteria. The part of the system that services the spent fuel pool area complies with the applicable GDC (GDC 2 and 4) referenced in SRP Section 9.4.2, and consequently with the subject SRP acceptance criteria. Therefore, the staff concludes that the RB safety-related equipment HVAC system design is acceptable.

9.4.5.3 RB Non-Safety-Related Equipment HVAC System, RB Mainstream Tunnel HVAC System, and RB RIP Adjustable Speed Drive (ASD) HVAC System

DSER (SECY-91-235) Sections 9.4.5.3, 9.4.5.7 and 9.4.5.8 are consolidated in this section as part of this report.

The RB non-safety-related equipment HVAC system consists of six FCUs and four AHUs, each consisting of a cooling coil and a fan. Each fan coil unit cools the associated room during normal conditions in conjunction with the RB secondary containment ventilation system. The 10 individual rooms are identified in SSAR Section 9.4.5.3.2. The units are open-ended and continuously recirculate cooling air within the space served. This system is manually started. The system description, figure, and component description are given in ABWR SSAR Section 9.4.5.3, Figure 9.4-3, and Tables 9.4-4i, respectively.

The RB main steam tunnel HVAC system consists of two closed-loop fan-coil recirculation units that remove heat from the tunnel area. Each fan-coil unit consists of a cooling coil and two fans, one of which operates while the other is in standby. A flow switch in the operating fan discharge ductwork automatically starts the standby fan on receiving indication of operating fan failure. The system description, figure, and component description are given in SSAR Section 9.4.5.7, Figure 9.4-3, and Tables 9.4-3, 9.4-4f, 9.4-4g, and 9.4-4i, respectively.

The Divisions I and II for the RB RIP ASD HVAC system are identical. Each division consists of two supply fans (one standby), one cooling coil for each fan, and a closed-loop system that cools the associated power panel. The divisions are started manually from the control room, and an air flow failure sensed by the flow switch automatically starts the standby fan and activates an alarm in the control room to indicate the fan failure. The system description, figure, and component description are given in SSAR Section 9.4.5.8, Figure 9.4-5, and Tables 9.4-3 and 9.4-4g, respectively.

The three separate HVAC systems are not safety-related and are not required to maintain reactor coolant pressure boundary or to achieve and maintain safe-shutdown conditions. These HVAC systems include features that provide reliability over the full range of normal plant operation such as redundant components for the RB main steam tunnel HVAC system and the RB RIP power supply panel room HVAC system. HVAC normal cooling water is the cooling medium for these systems. These three systems comply with GDC 2 regarding protection from natural phenomena because the applicable nonessential portions of the system are designed in accordance with Position C.2 of RG 1.29 as stated in SSAR Table 3.2-1. The requirements of GDC 60 regarding control of releases of radioactive materials to the environment are not applicable because these systems are recirculation type systems and do not contain any HEPA filters or charcoal adsorbers. Further, because the systems do not perform any safety function and are not used during an accident, GDC 4 and 17 are not applicable.

GE provided a set of design description and ITAAC relating to the RB non-safety-related equipment HVAC system, RB main steam tunnel HVAC system, and RB RIP ASD room HVAC system. 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, Open Item 9.4-1 as it relates to the above systems, is resolved.

The staff concludes that the RB non-safety-related equipment HVAC system, RB MST HVAC system, and RB RIP ASD HVAC system comply with applicable GDC referenced in SRP Section 9.4.5 and consequently with the subject SRP acceptance criteria and are, therefore, acceptable.

9.4.5.4 RB Safety-Related Electrical Equipment HVAC System

The system provides air and heat removal (cooling function) to maintain controlled temperature environments to ensure the continued operation of safety-related components (in the areas serviced by the system) under accident conditions. The system consists of three divisions, each serving five areas: the day tank, diesel generator engine, diesel generator MCC area, electrical equipment, and HVAC equipment rooms. Divisions A and B also serve remote shutdown panel rooms A and B, respectively, and Divisions B and C also serve RIP ASD control panel rooms. Division B of the HVAC system serves Division II electrical equipment room and the Division IV electrical equipment room. The staff evaluated these systems in accordance with SRP Section 9.4.5.

Each of the three RB safety-related electrical equipment HVAC system divisions consists of an ACU and redundant, 100-percent capacity supply and exhaust fans. Each ACU unit includes a medium-grade bag-type filter and a chilled water cooling coil for each supply fan and an electric heater. A safety-related divisional HECW system supplies chilled water. Each room is maintained at positive pressure with respect to the surrounding spaces under normal and accident conditions. Each divisional system is started manually from the control room. The system includes temperature control features. The system and the RB safety-related DG HVAC system supply fan maintain the DG room temperature below 45 °C (113 °F) when the DG is operating. The system maintains other areas served by it below 40 °C (104 °F). The system also has smoke removal capability. Upon receiving smoke alarm in a division of the system, operators will manually switch the system to smoke removal mode by closing the recirculation damper, stopping the exhaust fan, and opening the exhaust fan bypass damper to allow outside air to purge the affected area. No other division is affected by this action. Fire dampers separating electrical Divisions II and IV rooms that use fusible links in HVAC ductwork close under air flow conditions after the fusible links melt.

The system and components are located in the seismic Category I, reactor building, which protects against tornado missiles and flooding. All components are rated seismic Category I, and air intake and exhaust structures are protected against tornado missiles. Thus, the system meets the Position C.1 of RG 1.29 and, thereby, complies with the requirements of GDC 2 as it relates to protection from natural phenomena.

The power supply to the RB safety-related electrical equipment HVAC divisions during a LOOP. Medium-grade bag-type filters in each of the three divisions of the HVAC system will remove dust and particulate matter from air intakes (supply air). The intake louvers are located at 15.2 m (50 ft) above grade, thereby meeting the guidance of Item 2, Section A, of NUREG/CR-0660. Therefore, the essential electrical equipment HVAC system meets the pertinent requirements of GDC 17 relating to the protection of essential electrical components from failure as a result of accumulation of dust and particulate material.

In the DSER (SECY-91-235), the staff stated that it would evaluate the system for compliance with GDC 4 for protection from the environmental and dynamic effects associated with equipment failures upon receiving system P&IDs and clarifying information on compliance with the GDC. GE submitted this information in revised SSAR Section 9.4.5.4, SSAR Figure 9.4-4, and component description Tables 9.4-3, 9.4-4 9.4-4a and b. The staff reviewed this information, the reactor building arrangement plan drawings, Section 3.6.1 of this report, and SSAR Section 3.6.4 and Table 3.6-2, and finds that the system components are protected against the effects of piping failures and internally-generated missiles by compartmentalization and separation, thereby satisfying the requirements of GDC 4. This resolved the unnumbered open item in DSER Section 9.4.5.4 not listed in Section 1.8 of the DSER.

In DSER (SECY-91-235) Interface Information Item 32, the staff stated that it would evaluate the monitoring of the exhaust from the areas served by the HVAC system and the electrical controls (i.e., interfaces) upon receiving additional system description. SSAR amendments through Amendment 34, GE included additional system description. The staff reviewed the information and the revised SSAR Section 11.5.2.2.4, and finds that the areas serviced by the RB safety-related electrical HVAC system contain no radioactive systems. These areas are at positive pressure with respect to the surrounding spaces (supply fan flow rate exceeds exhaust fan flow rate) and potentially contaminated adjoining areas. Because radioactive releases through exhausts from these areas to the environment are only from what would have to be brought first into these areas by their own supply fans, the exhaust from these areas are not monitored. Therefore, the DSER Interface Information Item 32 is resolved. For the reasons stated above, no HEPA filters or charcoal adsorbers are provided to control releases from these areas to the environment. Therefore, GDC 60 is not applicable to the system.

The staff concludes that the RB safety-related electrical equipment HVAC system meets applicable GDC referenced in SRP Section 9.4.5 consequently with the subject SRP acceptance criteria and, therefore, is acceptable.

9.4.5.5 RB Safety-Related Diesel Generator HVAC System

The RB safety-related DG HVAC system consists of three independent and identical divisions, each of which will service one of the three DG rooms. The divisions supply fresh air to ensure the continued operation of the safety-related diesels under accident conditions. The system and the RB safety-related electrical equipment HVAC system maintain the DG room below 45 °C (113 °F) when the DG is operating. The staff reviewed the system in accordance with SRP Section 9.4.5. Accordingly, the following criteria were considered: GDC 2 as related to the system being capable of withstanding the effects of earthquakes (meeting the guidance of RG 1.29, Position C.1, GDC 4, with respect to maintaining environmental conditions in essential areas compatible with the design limits of the essential equipment located therein during normal, transient, and accident conditions, GDC 5, as related to shared systems and components important to safety, GDC 17, as related to assuring proper functioning of the essential electric power system (protection from the effects of dust and particulate materials), and GDC 60, as related to the systems' capability to suitably control the release of radioactive effluent to the environment.

Each division consists of two intake louvers, a medium-grade inlet air filter, two supply fans, ductwork, and an exhaust louver. Both fans of each system draw air from outside, filter it, and distribute it to the respective diesel generator room. The exhaust air is forced out through the exhaust louvers. Each system is interlocked with the associated diesel generator starting system, but remote manual override capability is provided for the system fans. The systems are designed to facilitate inspection. The air intake louvers are located at 11.5 m (37.7 ft) and exhaust louvers at 8.5 m (28 ft) above grade, thereby meeting the guidelines of Item 2, Subsection A, of NUREG-CR-0660. Therefore, the systems meet the pertinent requirements of GDC 17 for protection of the essential electrical components of the systems against dust accumulation and particulate material. The power supply to each of the systems allows uninterrupted operation during LOOP.

From its review of SSAR Table 3.2-1, the staff concludes that each essential diesel generator HVAC system is composed of seismic Category I components. The air intake and exhaust structures are protected against tornado missiles. Each system, by virtue of its location in the seismic Category I flood-, tornado-, and missile-protected reactor building, is protected from the effects of earthquakes, tornados and floods. Consequently, the safety-related systems meet Position C.1 of RG 1.29, and, thereby, comply with the requirements of GDC 2.

In the DSER (SECY-91-235), the staff included a partial evaluation of system compliance with GDC, stating that it would complete a full evaluation upon receiving adequate information to demonstrate the system compliance with GDC 4. Therefore, the staff identified this issue of system compliance with GDC 4 as Open Item 64 in the DSER (SECY-91-235). After the staff issued the DSER (SECY-91-235), GE submitted a revised SSAR Section 9.4.5.5, SSAR Figure 9.4-4, and component description Tables 9.4-3, 9.4-4a and 9.4-4b. The staff reviewed this information, the reactor building arrangement drawings, and SSAR Figure 9.4-4, and finds that the system components are protected against the effects of piping failures and internally-generated missiles by being located in adequately separated and dedicated diesel generator rooms, thereby satisfying GDC 4 regarding protection from environmental and dynamic effects associated with equipment failures. This resolved DSER Open Item 64.

In the DSER (SECY-91-235), the staff stated that it would evaluate the monitoring of the exhaust from the areas served by the system and associated interfaces upon receiving P&IDs and additional system information from GE. The staff designated this lack of information as Interface Information Item 33 in the DSER (SECY-91-235). SSAR amendments through Amendment 34 included additional system description. The staff reviewed this information and the revised SSAR Section 11.5.2.2.4 and finds that the areas serviced by the system contain no radioactive systems. Because radioactivity releases from these areas to the environment are only from what would have to be brought first into these areas by their own supply fans, the system exhaust is not monitored. Therefore, the DSER Interface Information Item 33 is resolved. The system includes no HEPA filters or charcoal adsorbers to control releases from the system to the environment. Therefore, GDC 60 does not apply to the system. The cooling function of the system and smoke removal aspects are separately addressed in Sections 9.4.5.4 and 9.5.1.4.4 of this report.

The staff concludes that the RB safety-related diesel generator HVAC system complies with applicable GDC 2, 4, and 17 of SRP Section 9.4.5 as discussed above and, therefore, is acceptable.

9.4.5.6 RB Primary Containment Supply and Exhaust HVAC System

The system ventilates the primary containment, using supply from, and exhaust to, the secondary containment HVAC system. The staff reviewed operation of the system in accordance with SRP Section 9.4.5. The staff reviewed other aspects of the system in accordance with SRP

Section 6.2.4, BTP CSB 6-4 as discussed in Section 6.2.4 of this report.

The system has no safety-related function and is not required to maintain the reactor coolant pressure boundary or to achieve and maintain safe-shutdown conditions. The system consists of a purge supply fan, a HEPA filter, a purge exhaust fan, duct work, and controls. The system has the capability to purge the drywell and the wetwell, if required. When the system is in use and the air is not highly radioactive, the system discharges to the secondary containment HVAC system for filtering by a bag-type filter and exhausting to the plant vent stack. However, if the air is highly radioactive, it is discharged through the SGTS system. A high-radiation signal actuates an alarm, closes the isolation valves in the exhaust duct, and initiates the SGTS (for evaluation of SGTS, see Section 6.5.1 of this report). The system description, P&ID, and component description are in SSAR Section 9.4.5.6, Figures 6.2-39 and 9.4-3, and Tables 6.2-7, 9.4-4g, and 9.4-4h, respectively.

SSAR Section 6.2.5 describes the use of the primary containment purge system during the long-term post-accident cleanup operation and the deinerting operation (shutdown) during which the system provides air to the drywell and wetwell through the purge supply. Refer to Section 6.2.4.1 of this report for the above interfaces and compliance with BTP CSB 6-4.

The purge supply is filtered twice, first by a bag-type filter (secondary containment HVAC system) and then by a system HEPA filter. The purge air supply has an isolation damper downstream of the system HEPA, and the exhaust has redundant isolation dampers in the flow path through the secondary containment HVAC system and the safety-related SGTS.

The system is located in the seismic Category I reactor building, which protects against flood and tornado missiles. The system is non-safety-related. However, SSAR Table 3.2-1, the staff concludes that the radiation monitors and dampers for isolating the system are designed to seismic Category I requirements, and the applicable nonessential positions are designed in accordance with Position C.2 of RG 1.29. Thus, the system complies with GDC 2 regarding protection from natural phenomena by meeting Position C.1 of RG 1.29 for the essential portions, and Position C.2 for the nonessential portions.

In the DSER (SECY 91-235), the staff stated that it would review the essential portions of the system for compliance with GDC 4 upon receiving detailed system description including P&IDs. The staff identified the lack of information pertaining to system compliance with GDC 4 as Open Item 65 in the DSER (SECY-91-235). SSAR amendments through Amendment 34 included the subject system information. The staff has reviewed the information and finds that the essential portions of the system are protected against damage from internally-generated missiles by separation of redundant equipment and are also protected against piping failures. Thus, the essential portions of the system comply with GDC 4 as they relate to protection from the environmental and dynamic effects of equipment failures. Therefore, DSER Open Item 65 is resolved.

GDC 17 does not apply to the system. The system complies with GDC 60 for control of releases of radioactive materials to the environment because under certain situations, the system is automatically isolated to prevent significant radioactivity release to the environment and facilitates the filtered release to the environment via the SGTS.

GE provided a set of design description and ITAAC relating to the primary containment (containment purge supply/exhaust) HVAC system. 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, DFSER Open Item 9.4-1 as it relates to the above systems, is resolved.

The staff concludes that the system design complies with applicable GDC referenced in SRP Section 9.4.5 and discussed above and consequently with the subject SRP acceptance criteria and, therefore, is acceptable.

9.4.5.7 RB Mainsteam Tunnel HVAC System

See Section 9.4.5.3 of this report.

9.4.5.8 RB Reactor Internal Pump Adjustable Speed Drive (ASD) HVAC System

See Section 9.4.5.3 of this report.

9.4.6 Radwaste Building HVAC System

The staff reviewed the radwaste building ventilation system in accordance with SRP Section 9.4.3. The system is located in the radwaste building. It is non-safety-related, and its failure does not compromise any safety-related system or component and does not prevent reactor safe shutdown. The system is designed to provide an environment with controlled temperature and air flow patterns to ensure both the comfort and safety of plant personnel and the integrity of equipment and components. The system services two zones: the radwaste building control room and the radwaste building process area.

Heating, cooling, and pressurization of the radwaste building control room are accomplished by an ACU which has two 100-percent supply fans (one standby). Outdoor air and recirculating air are mixed and drawn through a pre-filter, a high efficiency filter, a heating coil and a cooling coil by the associated fan. The supply sources for the heating and cooling coils are the hot water heating (HWH) system and the HNCW system, respectively. A differential pressure indicating controller modulates the inlet vanes in the supply fan air inlets to maintain the subject zone at positive pressure with respect to the surrounding spaces which includes the radwaste building process area. The zone has no exhaust fans; however, it has a 100-percent smoke removal fan to release smoke directly to the atmosphere, and this fan is operated manually. An area radiation monitor is provided in the radwaste control room and it will alarm on high radiation to alert personnel in the area.

The once-through HVAC system for the radwaste building process area consists of two 100-percent supply fans (one standby) each associated with a pre-filter, a high efficiency filter, a heating coil, and a cooling coil, and three 50-percent exhaust fans (one standby), each associated with a pre-filter and a high efficiency filter. Separate filters receive exhaust from the sorting table and compactor area. The exhaust air from the process area is monitored for radiation before it is released to the monitored plant vent. The overall air flow pattern in the zone is from the least potentially contaminated areas to the most contaminated areas. The design includes features to isolate manually affected areas on receipt of the area exhaust radiation alarm. The zone is at negative pressure with respect to surrounding spaces. The system description is given in SSAR Section 9.4.6.

The radwaste building ventilation system complies with GDC 2 regarding protection from natural phenomena because the applicable non-essential portions of the system are designed in accordance with Position C.2 of RG 1.29, as stated in SSAR Table 3.2-1. The design meets the requirements of GDC 60 (regarding control of releases of radioactive materials to the environment) for select areas in the radwaste building work areas by isolating the affected areas. Since the design of the system does not include HEPAs or charcoal adsorbers, the system is not required to meet RG 1.140 guidelines. In a letter dated August 22, 1990 (response to RAI 430.258), GE committed to submit the P&IDs, flow diagrams, and component description tables for the radwaste building HVAC system. However, the staff designated DFSER Open Item 9.4.6-1 because it did not receive this information until after issuing the DFSER. SSAR amendments through Amendment 32 specified COL License Information (Section 9.4.10, Item 9.4.10.2) which requires the COL applicant to submit flow rates and equipment lists. GE also submitted SSAR Figure 9.4-10, the P&ID for the system. The staff finds GE's approach to resolve the subject open item acceptable. Therefore, this item is resolved.

Based on the above, the staff concludes that the radwaste building ventilation system design complies with the applicable GDC referenced in SRP Section 9.4.3 and consequently, with the subject SRP acceptance criteria and is, therefore, acceptable.

9.4.7 Diesel Generator Area Ventilation System

The diesel generator building ventilation system is part of the reactor building ventilation system, which is reviewed in Section 9.4.5.5 of this report.

9.4.8 Service Building Ventilation System

The staff reviewed the service building ventilation system in accordance with SRP Section 9.4.3. The system description is given in SSAR Section 9.4.8. The system is non-safety-related and is not required for accident mitigation, maintenance of reactor coolant pressure boundary integrity, or achievement and maintenance of safe shutdown. It is located in the service building and operates during all normal conditions manually and continuously and provides an environment with controlled temperature and air flow patterns to ensure both the comfort and safety of the plant personnel, and the integrity of equipment and components in the building. The system consists of two non-safety-related HVAC systems: (1) the clean area HVAC system, which serves the technical support center (TSC), the operational support center (OSC), and other normally clean areas of the service building (instrument repair, locker, men and women's change and lunch rooms and laundry), and (2) the service building controlled area HVAC system, which serves the balance of the service building.

The clean area HVAC system supplies filtered, heated or cooled air to both the clean and controlled areas through a central fan system consisting of an outside air intake; an air conditioning unit consisting of filters, heating coils, cooling coils, two 50-percent capacity supply air fans; and a supply air duct. Additionally, the system has two 50-percent capacity exhaust fans and an emergency filtration unit (EFU). An automatic damper in the supply system ductwork regulates the flow of air to maintain the TSC and other clean areas at a positive pressure with respect to the surrounding spaces. The exhaust fans discharge the ventilation air from the clean areas to the outside from the service building roof top. The system also functions during a high radiation mode. For this purpose, radiation monitors are provided to detect radiation levels in the outside air intake. On receipt of a signal for high radiation in the normal outside air intake, the normal air intake damper closes, the minimum outside air intake opens, and the ventilation air for the clean areas is routed through the EFU. The EFU consists of a heater and demister, pre-filter, HEPA filter, 5.1-cm (2-in.) charcoal adsorber, a second HEPA filter, and two circulating fans. The EFU can remove at least 95 percent of all forms of iodine from the influent stream. If the reactor site is adjacent to a toxic gas source that could produce releases of significance to plant operating personnel in the TSC, the COL applicant will establish protection against the intrusion of toxic gas into the areas served by the system. GE has identified COL license information in SSAR Section 9.4.10 which calls for the COL applicant to locate required toxic gas monitors in the outside air intake of the clean area HVAC system, with capability to detect toxic gas concentrations at which personnel protective actions have to be initiated.

The controlled area HVAC exhaust system consists of two 50-percent exhaust fans. The controlled area exhaust is released to the environs through the monitored plant vent. The controlled area is maintained at a slightly negative pressure with respect to the surrounding spaces including the TSC and other clean areas.

The service building ventilation system complies with GDC 2 regarding protection from natural phenomena because the applicable non-safety-related portions of the system are designed in accordance with Position C.2 of RG 1.29, as stated in SSAR Table 3.2-1.

There is no need to monitor or control releases to the environment from the clean areas since these releases are from areas which do not contain radioactive sources and the only manner radioactive materials can get into these areas is due to its own HVAC system supply fans bringing outside air into the areas. The controlled area also does not contain radioactive sources; however, due to leakage from the secondary containment or turbine building, the radiation levels in the controlled area may become high. If this happens, the controlled area HVAC system can be manually isolated to prevent releases from the area to the environment. Thus, the service building HVAC system complies with GDC 60 with regard to control of releases of radioactive materials to the environment.

By letter dated August 22, 1990 (response to RAI 430.262), GE stated that the COL applicant will handle the details of the system, including P&IDs and component description tables, and compliance with applicable guidelines of RG 1.140 for the system filtration unit. The staff agreed with GE's approach. The staff designated this as DFSER COL Action Item 9.4.8-1. SSAR amendments through Amendment 34, discussed COL license information (Section 9.4.10) which requires the COL applicant to submit the above information. This is acceptable to the staff.

Based on the above, the staff concludes that the service building ventilation system design complies with GDC 2 and 60 as referenced in SRP Section 9.4.3 and consequently, with the subject SRP acceptance criteria and is, therefore, acceptable.

9.4.9 Drywell Cooling System

The staff reviewed the drywell cooling system in accordance with SRP Section 9.4.5. The system is not designed to ensure the reactor coolant pressure boundary or to achieve and maintain safe-shutdown conditions. However, the system provides features to include redundant components to provide reliability over the full range of normal plant operation. The system provides conditioned air and nitrogen to cool equipment and maintain temperature within limits, as specified in SSAR Section 3.11, during normal operation and in the drywell head area, upper and lower drywell, shield wall annulus, and the wet-well air space. The drywell cooling unit function is manually controlled from the control room.

Two of the three fan coil units operate under normal conditions. Each fan coil unit consists of two cooling coils arranged in series, a drain pan, and a centrifugal fan. The return air passes over the first coil, which is cooled by RCW. Part of the cooled air is subsequently cooled by the second coil, which is cooled by HNCW. The twice-cooled air is mixed with the air that bypasses the second cooling coil. During a LOOP (when no LOCA signal exists), the fan coil units start automatically if power is available from the diesel generators. During such a situation, only RCW coils will provide cooling. The system description, figure, and table are in ABWR SSAR Section 9.4.9, Figures 9.4-8 and 9.4-9, and Tables 9.4-1 and 9.4-2, respectively.

The drywell cooling system complies with GDC 2 for protection from natural phenomena because the applicable nonessential portions of the system are designed in accordance with Position C.2 of RG 1.29, as stated in SSAR Table 3.2-1. The requirements of GDC 60 regarding control of releases of radioactive materials to the environment do not apply because the system is only a recirculation system and does not contain any HEPA filters or charcoal adsorbers. GDC 14 and 17 do not apply because the system will not perform or support any safety function and is not used during an accident.

Based on the above, the staff concludes that the drywell cooling system meets the applicable acceptance criteria of SRP Section 9.4.5 and is, therefore, acceptable.

9.5 Other Auxiliary Systems

9.5.1 Fire Protection System

The Commission directed that special attention be given to measures for fire protection in addition to the staff's review of other aspects of the ABWR design in accordance with the requirements for current operating plants. For example, the Commission concluded that the ABWR design must incorporate the resolution of significant fire protection issues raised through operating experience and through the External Events Program.

The NRC established fire protection requirements for nuclear power plants in GDC 3, 10 CFR 50.48, and Appendix R to 10 CFR Part 50. The Commission considered Sections III.G., III.J, and III.O of Appendix R to be of particular importance. In July 1981, NRC revised BTP APCSB 9.5-1 (SRP Section 9.5.1) to include these provisions from Appendix R.

The staff has also issued supplemental guidance on fire protection in documents such as Generic Letter (GL) 81-12 (45 FR 76602, November 19, 1981), dated February 20, 1981, and GL 86-10, dated April 24, 1986. GL 81-12 presents information on safe-shutdown methodology, and GL 86-10 presents technical information on conformance with National Fire Protection Association codes and standards.

To minimize fire as a significant contributor to the likelihood of severe accidents for the ABWR, the staff concluded that current NRC guidance must be enhanced. As stated in SECY-90-016, "Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements," dated January 12, 1990 (and reiterated in SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs"). The proposed applicable regulation discussed below encompasses the enhanced, NRC Guidance.

The NRC expects any new reactor design to propose fire protection systems based on the best technology available, not on the methods allowed for plants already operating or in advanced stages of design and construction. Therefore, the staff evaluated the fire protection system of the ABWR against the criteria of SRP Section 9.5.1 (BTP CMEB 9.5-1 Rev. 2), which meets the requirements of GDC 3, and against the following proposed applicable regulation for fire protection:

> The standard design must comply with 10 CFR Part 50 Appendix R, Section III.G.1.a and ensure that safe shutdown can be achieved assuming that all equipment in any one fire area will be rendered inoperable by fire and that re-entry into the fire area for repairs and operator actions is not possible. The design must ensure that smoke, hot gases, or fire suppressant will not migrate into other fire areas to the extent that could adversely affect safe-shutdown capabilities including operator actions. The control room is excluded because an alternative shutdown capability is provided which is physically and electrically independent of the control room. In the reactor containment, redundant shutdown systems must be provided with fire protection to ensure, to the extent practical, that one shutdown division be free of fire damage. Because of unique design layout, other areas may be accepted on an individual basis.

9.5.1.1 General Evaluation Fire Protection Program

GE generally followed the NRC's concept of defense in depth for fire protection as described in the SSAR. The three steps of defense in depth and GE's implementation of these steps follow.

(1) To reduce the possibility of fire starting in the plant, GE used fire-resistant and fire-retardant materials in its design of the ABWR to minimize and isolate fire hazards. Low-voltage multiplexed circuits and fiber-optic circuits eliminate the need for cable-spreading rooms and thereby substantially reduce the amount of combustible cable insulation and higher voltage ignition sources in the control room.

(2) To promptly detect and suppress a fire, GE has provided automatic detection and a suitable combination of automatic and manual fire suppression capabilities in the ABWR design.

(3) GE designed the plant so that any fire that might occur will not prevent safe shutdown of the plant even if fire detection and suppression efforts fail.

SSAR Section 9.5.1 and Appendix 9A (Fire Hazards Analysis) describe a fire protection program intended to protect safe-shutdown capability, prevent release of radioactive materials, minimize property damage, and protect personnel from injury as a result of fire.

GE also considered such features of general plant arrangement as:

- access and egress routes
- equipment locations
- structural design features separating or isolating redundant safety-related systems
- floor drains
- ventilation
- construction materials

The SSAR reflects the use of applicable National Fire Protection Association codes and standards in design and layout of the ABWR. In the DSER (SECY-91-355), the staff identified as Open Item 102, the need for GE to describe any deviations from those codes and standards and to describe in the Fire Hazards Analysis the measures taken to ensure that equivalent protection is provided. Section 9.A.3 of SSAR Appendix 9A describes three deviations from the fire codes: (1) exceptions to penetration requirements (Section 9A.3.5), (2) wall deviations (Section 9A.3.6), and (3) door deviations (Section 9A.3.7). In each case, GE described each deviation, why it was required, and how the underlying fire protection requirements are still satisfied. The staff found the GE explanations acceptable. Therefore, this resolved DSER Open Item 102.

In the SSAR, Section 9.5.1, GE indicated that the ABWR design meets the commitments as specified in BTP CMEB 9.5-1 except in five cases:

1. Deviation from BTP CMEB 9.5-1, Section 7.j, "Diesel Fuel Storage Areas"

The BTP states that the diesel fuel oil tanks with a capacity greater than 4,164 L (1,100 gallons) should not be located inside buildings containing safety-related equipment. If located within 15 m (50 ft) of such buildings, the tanks should be housed in a separate building with a construction having a minimum, fire resistance rating of 3 hours.

GE states that the capacity of each tank provides 8 hours supply, 12,113 L (3,200 gallons), for the emergency diesel generators. The diesel fuel oil day tanks are located in the reactor building but outside of secondary containment. Walls, ceiling, floors, and doors are all 3-hour fire barriers. SSAR Section 9.5.1 states that the sunken volume of the room will accommodate (hold) the entire contents of the tank and discharge from the automatic foam sprinkler system for 30 minutes if an uncontrolled leak occurs.

The doors from each fuel tank room open into the respective DG equipment room which is in the same division, but in a different fire area. Therefore, should a fire occur in the fuel tank room, it will need to propagate and damage two 3-hour barriers before it can penetrate and threaten another division. Smoke removal in the fuel tank room is accomplished by the HVAC system.

The staff position is that the capacity of the fuel oil day tank exceeds the specified limits in the BTP; however, GE committed to add an automatic foam system with early detection and suppression capabilities. In addition, as required by the BTP, the SSAR indicates that the COL applicant will provide a fire brigade capable of extinguishing any oil-type fire that may occur onsite, including one in the diesel generator rooms.

Based on the above discussion, the staff finds GE's justification acceptable for having the diesel fuel oil day tanks inside the reactor building.

2. Deviation from BTP 9.5-1, Section C.7.b, "Control Room Complex"

The BTP 9.5-1 requires that automatic fire suppression should be provided for the control room complex subfloor if it contains cable runs unless cable is run in 4-inch or smaller steel conduit or the cables are in fully enclosed raceways internally protected by automatic fire suppression.

SSAR Section 9.5.1(2) states that the subfloor area will not contain a fire suppression system as recommended by BTP CMEB 9.5-1 Section C.7.b because the amount of cabling under the floor is substantially less than that used in current designs. The types of cables located in the raised floor area smolder for a long time and are self-extinguishing. Cables will be located within conduit. The control room is continuously staffed so that the presence of a fire will be quickly detected. There is also a fire detection system in the subflooring which will quickly detect a fire in that area. Finally, in the unlikely event that the control room were to require evacuation, the remote shutdown panels contain the necessary controls to bring the plant to cold shutdown.

The staff finds GE's justification for not installing a suppression system in the subfloor of the control room complex acceptable.

3. Deviation from BTP 9.5-1, Section C.7.b, "Control Room Complex"

The BTP 9.5-1 requires that the office space in the control room complex be provided with an automatic fire suppression system.

GE SSAR indicated in Section 9.5.1(3) that the office spaces contained in the control room complex do not have automatic fire suppression systems installed.

GE SSAR indicated in Section 9.5.1(3) that the control room complex is continuously manned so that any fire will be quickly detected and manual suppression commenced without delay. Papers are limited and stored in file cabinets, book cases or other storage locations except when in use. GE indicated that fire detection will be installed in the peripheral rooms including office spaces. (Refer to SSAR 9A.4.2.4.1.)

The staff finds GE's justification for not installing an automatic fire suppression system in the office spaces of the control room complex to be acceptable.

4. Deviation from BTP 9.5-1, Section C.7.b, "Control Room Complex"

The BTP 9.5-1 requires fire detectors for the consoles and cabinets in the control room complex.

SSAR Section 9.5.1(4) states that the control room complex is continuously staffed so that any fire will be quickly detected and manual suppression commenced without delay. The cabinets and consoles contain limited combustibles and are air cooled so that smoke from a cabinet fire will exhaust to the control room complex. A fire in any single cabinet or console will not disable the capability to safely shut down the plant.

The staff finds the above commitments acceptable since these consoles are not walk-in type and are small. Therefore, no detection need be installed within these consoles. If a fire was to occur, the control room detection system will alarm or a control room operator will see or smell the smoke. The automatic fire detectors and continuous presence of operators will likely result in the fire being discovered and suppressed early.

GE also indicated that fire detection is not provided in the cabinets or consoles because the control room ceiling and peripheral rooms are to be provided with early detection systems.

GE agreed to install a fire detection system in the sub-floor area of the control room.

The staff finds GE's justification for this deviation acceptable.

5. Deviation from BTP CMEB 9.5-1, Section C.7.i, "Diesel Generator Area"

The BTP indicates that an automatic fire suppression system should be installed to combat any diesel generator or lubricating oil fire; such systems should be designed to operate when a diesel generator is running without affecting the diesel.

GE states that the ABWR has an automatic foam sprinkler system for diesel generators and diesel day tanks. The generator is not a 100-percent sealed unit. Openings are provided for cooling purposes. When a diesel is operating and fire occurs, the automatic foam sprinkler suppression system will activate and there is a possibility that the foam could be siphoned into the generator openings, causing damage to the generator or a short circuit to occur.

The ABWR design assumes fire would result in the loss of function for one division. Thus, two more diesel generators are still available for duty. While the DG equipment is designed to continue operating, actuation of automatic foam system due to a fire or inadvertent actuation of the foam system should prompt an operator to shut down the affected diesel generator.

SSAR Section 9.5.1.3.7 states that the automatic foam sprinkler system is actuated by cross-zoned rate of temperature rise, rate-compensated heat detectors, and infrared detectors. The system will not discharge foam until a heat activated fusible link activates in the sprinkler head.

The applicant's automatic foam sprinkler system significantly reduces the probability of an inadvertent actuation of foam on the emergency diesel generators. The applicant's meets the design aspects of GDC 3 (BTP 9.5-1) and GI 57. Therefore, the staff finds GE's justification acceptable.

GE also indicated in the SSAR that excess water, foam, and diesel fuel oil spilled outside the reactor building will not have an adverse effect on any safety-related equipment.

9.5.1.2 Specific Features of Protection

9.5.1.2.1 Protection of Safe-shutdown Equipment

In general, the ABWR design relies on 3-hour-rated fire barriers to separate safe-shutdown equipment from the remainder of the plant and from redundant systems and components outside of primary containment. However, where safety related equipment is not divisionally separated by 3-hour fire-rated barriers, the safety-related equipment is electrically isolated from its respective division by fuses and/or breakers for breaker coordination and multiple high impedance faults. This is documented in the SSAR, Section 9A.5. SSAR Section 9.5.13.12 states that the COL applicant will demonstrate that multiple high impedance faults of those circuits described in Table 9A.5-2, resulting from a fire within any one fire area, will not negatively affect other equipment fed from the same power. The use of 3-hour-rated fire barriers exclusively for separation of safe-shutdown equipment in the nuclear power block (NPB) areas outside containment is in accordance with the review criteria, and is acceptable. The use of fuses and/or breakers to address breaker coordination and high impedance fault concerns where safety-related equipment is not divisional, separated by 3-hour fire-barrier equipment as discussed in SSAR Section A.5, is in accordance with the review criteria and is acceptable. This design will ensure the capability for safe shutdown assuming all equipment in any one fire area will be rendered inoperable. In addition, this design provides the operator the ability to achieve safe shutdown without re-entry into the fire area for repairs or operator actions and meets the fire protection guidance of SECY-90-016.

The staff recognizes the need for open communication between compartments inside containment to relieve and equalize pressure following a high-energy line break. GE stated in the SSAR that the safety divisions will be widely separated around containment so that a single fire will not fail any combination of active components that could prevent safe shutdown. The staff finds that this meets the fire protection requirements of SECY-90-016, SECY-93-087, SRP Section 9.5.1, and 10 CFR Part 50, Appendix R, and is acceptable. The ABWR containment will be inerted with nitrogen during power operation, which will prevent propagation of any potential fire inside containment. In accordance with BTP CMEB 9.5-1, Section 7.a., the inerted containment significantly reduces oxygen so that combustion and/or fire will not be supported. This is an acceptable means of preventing a fire inside containment. The SSAR commits to an inerted containment. Therefore, the staff finds that this fire-suppression feature meets the guidelines of BTP CMEB 9.5-1, Section 7.a., and is acceptable. Therefore, the use of structural walls inside containment as fire barriers to separate safety-related systems (cabling, components, and equipment), even though such walls may not fully enclose the equipment requiring separation, is acceptable.

GE proposes and justifies two areas outside containment and one inside containment that will not conform to the 3-hour-rated fire barrier separation criteria. The three exceptions discussed below were well justified by GE and are acceptable to the staff.

1. The MST is an exception to separation of redundant safety-related components outside containment.

   SSAR Section 9A.4.1.4.26(9) states that all valves in the MST are designed to fail with acceptable consequences. For example, power-operated valves are backed up with air-operated valves not subject to damage from the same fire, or redundant valves are located in another fire area.

2. The MCR is an exception to separation of redundant safety-related components outside containment.

   SSAR Section 9.5.1.1.2 describes alternative shutdown capability that is spatially remote from, and electrically isolated from, the MCR.

   In the event of fire in the MCR, control room operators will have full safe-shutdown capability available at this alternative shutdown station. In the control room, fiber optics are used to operate power, instrument, and control circuits for safe-shutdown equipment.

   In discussions with the staff, GE stated that all controls and instrumentation (except for scram function, HPCF, and MSIV) will utilize a touch screen and touch panel. The characteristics of the specific touch screen and touch panel used and the affects of a fire on this equipment will depend on the as-procured equipment. If a fire can cause inadvertent operation of the touch screen or panels, the COL applicant will need to demonstrate that under the worst-case scenario, the reactor can be safely shutdown in a controlled manner and that long-term core cooling can be established and maintained. In addition, at no time shall top of the active fuel be uncovered. This is a COL Action Item (GI 147, "Fire Induced Alternative Shutdown Control Room Interactions").

3. Inside containment is an exception to separation of redundant safety-related components.

   The entire containment is one fire area. SSAR Section 9.5.1.1.2 states that the shutdown trains will enter containment widely spaced around the perimeter. This spacing ensures that no single fire will be able to damage any combination of active components that would prevent safe shutdown. In addition, the ABWR containment will be inerted with nitrogen during power operation, which ensures that any potential ignition/fire hazards inside containment will not propagate. In accordance with BTP CMEB 9.5-1, Section 7.a., the inerted containment significantly reduces oxygen so that combustion will not be supported. This is an acceptable means of preventing fire within containment. GE committed to an inerted containment in the SSAR. Therefore, the staff finds that this fire suppression feature meets the guidelines of BTP CMEB 9.5-1, Section 7.a., and is acceptable.

9.5.1.2.2 Passive Fire Protection Features

Passive fire protection features for the ABWR design consist of building assemblies such as walls, partitions, floor-ceiling assemblies, columns, beams, fire dampers, fire penetrations seals, and fire doors. Penetrations through the building assemblies, such as doorways, hoistways, stairways, and cable trays and conduits are protected by appropriate fire-rated doors, dampers, plugs, and seals. SSAR Section 9.5.1.1.3 states that it intends to select passive fire-protection components of proven design, which have previously been tested and are listed by nationally recognized testing laboratories.

SECY-90-016 requires separation by 3-hour fire-barriers for redundant trains of all safety-related equipment, not just safe-shutdown equipment. Therefore, the staff reviewed the lack of 3-hour barriers between redundant trains of four separate systems: (1) SLCS, (2) SGTS, (3) FCS, and (4) spent fuel pool cooling system (FPC). The staff evaluations for these systems are provided below:

1. The SLCS is the alternative to the CRD system and is separated from it by 3-hour fire-rated barriers. This arrangement satisfies the SECY-90-016 requirements and is, therefore, acceptable.

2. During a meeting with the staff on May 5, 1992, GE committed to move one train of the SGTS to an available space on a mezzanine level directly above the location presently shown (SSAR Figure 9A.4-6). In a subsequent amendment to SSAR Section 6.5.1.3.1, GE states that the SGTS has independent, redundant active trains. The two SGTS trains are mechanically and electrically separated and are located in two side-by-side compartments separated by rated fire barriers and adjacent to the HVAC system exhaust. This arrangement satisfies the staff's concern because these two areas will be completely separated from each other by 3-hour fire-rated barriers.

3. GE changed SSAR Figure 9A.4-4 to show a 3-hour fire-rated barrier separating the redundant trains of the FCS. This arrangement satisfies the staff's concern and is, therefore, acceptable.

4. The FPC system is not a safety-related system. If the FPC is lost because of fire, the fuel pool will slowly heat up and eventually boil. GE calculated that the pool would begin boiling in about 16 hours under the most adverse conditions of decay heat load from a recently off-loaded full-core with no emergency makeup water supplied to the pool. However, emergency makeup water can be supplied to the fuel pool within 30 minutes from (1) the RHR system, (2) the SPCU system, (3) MUWC system, or (4) the fire protection water system by means of hoses. The staff finds that the FPC is not required to prevent off-site releases and is not a safety-related system. Therefore, this staff concern is resolved.

9.5.1.3 Fire Protection System Description

GE submitted the design description and the ITAAC for the fire protection system. The staff designated this as DFSER Open Item 9.5.1.3-1. GE submitted a revised set of design description and ITAAC, which are evaluated in Section 14.3 of this report. Open Item 9.5.1.3-1 is resolved.

9.5.1.3.1 Fire Detection Systems

A previous amendment to the SSAR stated that the ABWR automatic fire detection systems are designed and installed in accordance with National Fire Protection Association (NFPA) Standards 72D and 72E, and will protect all safe-shutdown components from all significant hazards. In the DFSER, the staff stated that NFPA 72A, 72B, 72D, and 72E have been incorporated into NFPA 72, and are no longer separate standards. Therefore, all references to NFPA 72D and 72E should be changed to NFPA 72. The staff designated this as DFSER Open Item 9.5.1.3.1-1. Amendment 33 to the SSAR states that fire detection systems are designed according to NFPA 72 Class A and NFPA 70 Class 1 requirements. Therefore, DFSER Open Item 9.5.1.3.1-1 is resolved.

The ABWR will include detection capability for major cable concentrations, safe-shutdown-related major pumps, switchgear, motor-control centers, battery and inverter areas, relay rooms, fuel areas, and all other areas that may contain appreciable in-situ or transient combustibles. Detector devices will be selected on the basis of the type of anticipated fire and located on the basis of ventilation, ceiling height, ambient conditions, and burning characteristics of the involved materials. Detection systems will alarm and be annunciated in the control room and will give a distinctive audible and, if necessary, visual local alarm to aid the fire brigade in finding the fire.

Therefore, the staff concludes that the automatic fire detection capability provided for the ABWR meets the guidelines of Section C.6.a of BTP CMEB 9.5-1, as discussed above, and is acceptable.

9.5.1.3.2 Fire Protection Water Supply System

The dedicated fire protection water supply and distribution system is designed and will be installed in accordance with NFPA Standards 11, 13, 14, 15, 16, 16A, 20, and 24 to meet the anticipated needs for fixed water suppression systems and manual hose stations.

The sprinkler systems in the reactor building and the wet standpipe systems in the reactor and control buildings are designed in compliance with ANSI B31.1 and analyzed to remain functional following an SSE (seismically analyzed). Portions of the water supply system, including a tank, a pump, and part of the yard supply main, are also designed to these requirements. The remainder of the water systems are designed to appropriate fire protection standards. During normal operation, the seismically analyzed systems will be separated from those not seismically analyzed by normally closed valves and a check valve so that a break in the non seismically analyzed portion of the system cannot impair the operation of the seismically designed portion of the system.

The water supply system is required to be a fresh water system, filtered if necessary to remove silt and debris. The system has two sources, each with a minimum capacity of 1,136,000 liters (300,000 gal.). If the primary source is a volume-limited supply such as a tank, a minimum of 455,000 liters (120,000 gal.) must be passively reserved for use by the seismically-designed portion of the fire suppression system. This reserve will supply two manual hose reels for 2 hours. A diesel-driven pump in the train is designed to remain functional following the SSE. A jockey pump will maintain pressure on the system.

The turbine building will have modified Class III standpipes, hose reels, and ABC portable extinguishers throughout the building. The following fire suppression systems will have primary fire suppression capability for the following areas:

- Automatic closed-head sprinkler systems in the open grating area of the three floors under the turbine.
- Deluge foam-water sprinkler systems in the lube oil conditioning area and the lube oil reservoir area.
- A deluge sprinkler system in the hydrogen seal oil unit area.
- A preaction sprinkler system in the auxiliary boiler area.

The turbine building fire suppression systems receive water from the portion of the supply system that is not required to be seismically analyzed for SSE.

The main power, unit auxiliary, and reserve transformers will have deluge water spray suppression systems. The systems are automatically actuated by flame or temperature detectors. A dike is provided to collect oil and water beneath each transformer. Drains are provided for each pit to divert oil and water away from buildings and transformers. Shadow-type fire-barrier walls separate adjacent transformers.

Alarm systems, both manual and automatic, will be in all areas of the plant as passive systems. They alarm without controlling an extinguishing function.

The two fire pumps are located in separate fire areas, separated from each other and the plant by 3-hour-rated fire barriers.

The fire-main loop in the yard will be designed and installed with sectional control valves that will deliver total fire flow and pressure design requirements to all automatic and manual fire suppression systems and manual hose stations with the shortest portion of the water distribution piping out of service.

On the basis that the fire water supply and distribution system conforms to the applicable NFPA standards mentioned above, the staff concludes that the system will meet the guidelines of Section C.6.b of BTP CMEB 9.5-1 and is acceptable

9.5.1.3.3 Water Fire Suppression Systems

The SSAR states that automatic water and foam-fire suppression systems will be installed over major fire hazards identified in the fire hazards analysis. Each system will be designed and installed in accordance with NFPA 11, 13, and 15.

Standpipe and hose stations will be installed throughout the plant as determined in the fire hazards analysis. The standpipe systems will be designed and installed in accordance with NFPA 14. Each hose station will be equipped with a maximum of 30 m (100 ft) of 3.8-cm (1.5-in.) hose and an adjustable on/off spray nozzle listed or approved by a nationally recognized testing laboratory.

Pressure-reducing orifices will be installed at each hose station as required, to ensure that excessive pressures are not delivered to the nozzle.

Exterior hydrants and hose houses will meet needs described in the fire hazards analysis and will be designed and equipped in accordance with NFPA 24.

Control and sectionalizing valves in the fire water system will be electrically supervised and will be indicated in the MCR.

The ABWR will not include floor penetrations that are susceptible to the potential of channeling water from fire extinguishing operations in one redundant fire area to an adjacent fire area. Floor penetrations will only be used for interconnections within one train of safe-shutdown equipment.

The fire protection water distribution and extinguishing systems will conform to the guidelines contained in Sections C.6.b and C.6.c of BTP CMEB 9.5-1 and is acceptable. This resolved DSER (SECY-91-355) Open Item 104.

9.5.1.3.4 Automatic Foam Fire Suppression Systems

GE specified an automatic foam sprinkler fire suppression system for protection of the EDGs. GDC 3 of Appendix A of 10 CFR Part 50 states that an inadvertent operation of the foam system should not adversely affect the diesel. Section C.7.i. of BTP 9.5-1 CMEB states that "Automatic fire suppression should be installed to combat any diesel generator or lubricating oil fire; such systems should be designed for operation when the diesel is running without affecting the diesel." GI 57 addresses the potential for safety-related equipment to become inoperable due to an inadvertent operation.

The applicant committed to meet the design aspects of GDC 3, BTP 9.5-1 and GI 57, therefore, the staff concludes that the automatic foam fire suppression systems are acceptable. This item was previously discussed in Section 9.5.1.1 of this report.

9.5.1.3.5 Fire Extinguishers

Portable fire extinguishers will be provided in areas with in-situ or potentially transient combustibles. They will be chosen on the basis of the anticipated type of fire in the area and the effect of the extinguishing agent on equipment in the area. The portable extinguishers will be selected, installed, and maintained in accordance with the requirements of NFPA 10.

The portable fire extinguishers will conform to the guidelines of Section C.6.f of CMEB 9.5-1 and will, therefore, be acceptable.

9.5.1.4 Fire Protection Support Systems

9.5.1.4.1 Emergency Communication and Lighting

The BTP 9.5-1 requires portable radios be provided for fire brigade and plant operations personnel to communicate during a fire incident. This communication system will have a distinct and separate frequency so that plant security force communications and actuation of protection relays will not be affected. The portable radio communication system will use fixed repeaters, as necessary, to ensure communications capability with any location in the station from the control room. The fixed repeaters will be arranged and protected so that exposure to fire damage will not disable the entire system. SSAR Section 9.5.2.6.2 states that "Design of fixed emergency communication and portable communication systems shall comply with BTP CMEB 9.5.1, Position C.5.g.(3) and (4)."

Sealed-beam emergency lights with individual 8-hour battery supplies will be provided in areas that must be occupied for safe shutdown and in routes used for access and egress to these locations. The lighted areas will include areas where operator actions occur if the control room is evacuated. In addition to the sealed-beam 8-hour emergency lights, portable sealed-beam battery-powered hand-held lights will be provided for use by fire brigade and plant operations personnel during a fire incident.

GE responded to staff questions about battery-powered emergency lights in harsh (extreme high or low temperature) environments by stating the following in SSAR Section 9.5.3.1.1(5):

(f) Non-essential battery pack lamps shall be self-contained units suitable for the environment in which they are located.

(g) The light fixtures for essential battery packs may be located remotely from the battery if the environment at the lamp is not within the qualified range of the battery. Alternatively, lamps powered from the station batteries may be provided.

Based on the applicant's commitments, the emergency communication and lighting design conforms to the guidelines of Section C.5.g of BTP CMEB 9.5-1 and is acceptable.

9.5.1.4.2 Emergency Breathing Air

The staff reviewed GE's commitments regarding the provision of emergency breathing air in accordance with BTP CMEB 9.5-1. Specifically, Position C.3(c) indicates that the fire protection program will include the provision of fire brigade equipment including self-contained breathing apparatis for brigade members.

In accordance with BTP CMEB 9.5-1, GE has made the following commitments concerning the provision of emergency breathing air for the fire brigade staff. Emergency breathing air will be provided for fire brigade and control room personnel. The breathing air will be delivered by a self-contained apparatus or a storage reservoir. Full-face positive-pressure masks approved by the National Institute for Occupational Safety and Health will be used by all personnel required to use emergency breathing air.

A minimum of 10 self-contained breathing units will be provided for fire brigade use. Each unit will be provided with two extra air bottles located on site. The rated service life for the self-contained units will be a minimum of 1/2 hour. In addition to the two extra bottles for each self-contained unit, compressors will be provided so that exhausted air bottles may be quickly replenished. The compressors will operate in areas free of dust and contaminants and will be powered from a vital power bus so that breathing air is available if off-site power is lost.

The specific provisions for emergency breathing air conform to the guidelines contained in Section C.3(c) of BTP CMEB 9.5-1 and are acceptable.

9.5.1.4.3 Curbs and Drains

Floor drains and curbs sized to remove expected fire-fighting water flow will drain areas protected by fixed-water fire-suppression systems or hand-held hose lines to prevent water accumulation from causing unacceptable damage to safety-related equipment. Water drained from areas that may contain radioactivity will be properly collected, analyzed, and treated before being discharged to the environment.

SSAR Section 9.5.1 states that the control room will not have floor drains. The SSAR describes the flow path that the water will travel if the subflooring was to overflow. The path traveled by the water will not jeopardize safety-related equipment because of floor drains and pedestals which elevate safety-related equipment. One division outside the control room will not be in the path of the discharged water.

Floor drains in areas containing combustible liquids will be designed so that these liquids cannot flow back into safety-related areas through the drainage system.

The provisions for curbs and drains conform to the guidelines in Section A.2 of BTP CMEB 9.5-1 and are acceptable.

9.5.1.4.4 Smoke Control

In SECY-90-016, "Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements," dated January 12, 1990, the staff stated that the ABWR design must ensure that smoke, hot gases, or the fire suppressant will not migrate into other fire areas to the extent that they could adversely affect safe-shutdown capabilities, including operator actions.

SSAR Section 9.5.1.1.6 states that the HVAC systems will control smoke and meet the requirements of ASHRAE's, "Design of Smoke Control Systems for Buildings," and NFPA's "Recommendations Practice for Smoke Control Systems." GE describes the major features that will be incorporated into their design which include the following:

- Venting of the area of the fire to prevent the fire from causing pressure to increase.
- Pressure control across the fire barrier to ensure that any leakage is into the area experiencing the fire.
- Pressure control and purge air supply to prevent back flow of smoke and hot gases when fire-barrier doors are maintained open for access for manual fire-fighting activities.

Smoke will be removed from the area of the fire by the normal ventilation systems (smoke removal mode). Release of smoke that may contain radioactive materials will be monitored to ensure compliance with applicable guidelines and regulations.

The general arrangement of the ABWR design of safe-shutdown trains features a high degree of separation with no piping and minimal cabling interconnections between divisions. With such a physical arrangement, the ventilation system can become the most likely pathway for fire propagation and smoke dispersal. Outside primary and secondary containment, the ABWR employs separate, dedicated HVAC systems for each fire area containing safe-shutdown equipment.

This arrangement of the ventilation systems serving the areas containing safe-shutdown equipment facilitates the venting of smoke from one area containing safe-shutdown equipment, and precludes the smoke from spreading to the redundant area containing safe-shutdown equipment. This arrangement also means that ventilation air supply, return, and exhaust for any fire area is independent of all other fire areas and that HVAC ducting does not penetrate the 3-hour-rated fire barriers separating fire areas. Therefore, the ABWR design will not include fire dampers outside the secondary containment. This simplifies not only design of the ABWR HVAC systems, but also installation and maintenance of the system throughout the life of the plant.

The HVAC system removes smoke in the event of fire. For all systems, except the secondary containment and diesel generator rooms, placing a HVAC system in the smoke-removal mode causes a damper in the return line to close and a much larger exhaust damper to atmosphere to open so that the system becomes a no-return, flow-through system with the large exhaust plenum vented directly to atmosphere. The pressure in the area experiencing the fire drops to atmospheric pressure plus the duct loss in the exhaust duct and the pressure in other divisional fire areas remains at their normal positive pressure of approximately 0.06 Kpa (1/4 of an inch of water). This assures that air leakage through any openings in the fire barriers surrounding the fires is towards the fire. GE has stated in SSAR Section 9.5.13.10, that during the detailed design phase the required differential pressure for each barrier will be calculated and the HVAC systems are to be designed to provide the required pressure. This is a COL Action Item and is not needed for design certification.

The smoke removal mode for the RB secondary containment HVAC system differs from that of the other systems, since common supply and exhaust systems are used for all three divisional areas within the secondary containment. The systems from each division are branched from the common system. Each supply and exhaust branch includes a dual purpose isolation/fire damper valve. Each exhaust branch also includes a two-position motor-operated volume damper. Upon detection of a fire, a normally non-operating exhaust fan is started to increase the negative pressure of the exhaust system. The motor-operated dampers in the exhaust ducts for the divisional/HVAC fire areas without the fire, reposition to their pre-determined fire settings to maintain normal negative pressure in their zones. The pressure in the HVAC and fire area will become more negative with the change in exhaust pressure. This establishes a differential pressure across the fire barriers surrounding the fire. As discussed above, the required differential pressure for each barrier will be calculated during the detailed design phase, and the HVAC systems will be designed to provide the required pressure.

GE also committed to install smoke detectors in the fresh air intakes for the ABWR secondary containment and the control building. If smoke enters the fresh air supply for these HVAC systems, they will isolate from the outside supply and start operating in the 100-percent recirculation mode.

Since the primary containment is inerted with nitrogen, the separation of HVAC systems inside primary containment is not an issue during plant operation.

To remove smoke from the diesel generator room, the once-through supply fans for the room start, and purge the room through the large exhaust opening [4 m^2 (about 43 ft^2]). The applicant has committed to design the smoke removal systems in accordance with ASHRAE's "Design of Smoke Control Systems for Buildings" and NFPA's "Recommendations Practice for Smoke Control Systems." The applicant states that a minimum of 0.06 pa (.25 in. H_20) will be maintained across the fire barriers. The applicant states that sufficient flow and pressure will be available to prevent smoke migration into the non-fire area when doors are open to support any fire fighting operations.

Based on the above applicants commitments, the staff concludes that the design of the ABWR HVAC systems in the smoke-removal mode of operation is in accordance with the review criteria and is acceptable. This resolved DSER (SECY-91-355) Open Item 103. (See also Section 9.5.1.2.2 of this report.)

9.5.1.4.5 Access/Egress Routes

Fire exit routes will be clearly marked provided for each fire area. These routes will be designed to comply with applicable life safety codes and standards. The provisions for access and egress routes in all areas needed for operation of safe shutdown equipment will conform to the guidelines contained in Section III.J of Appendix R to 10 CFR Part 50, and are acceptable.

9.5.1.4.6 Construction Materials and Combustible Contents

GE has defined non-combustible materials used for interior wall and structural components, thermal insulation, radiation shielding, soundproofing, interior finishes, and suspended ceilings in Section 9A.2.3(4) of SSAR Appendix 9A. GE also states acceptance criteria for surface finishes in Section 9A.2.4(6) of SSAR Appendix 9A. Both the definition and the acceptance criteria are based on the technical requirements of American Society for Testing and Materials (ASTM)-E84/NFPA 255, and are acceptable to the staff.

BTP CMEB 9.5.1 Section 5.A.12 indicates that transformers installed inside fire areas containing safety-related systems should be of the dry type or insulated and cooled with noncombustible liquid. Transformers filled with combustible fluid located indoors should be enclosed in a transformer vault.

With regard to non-combustible liquid-insulated transformers, the staff identified in Section 9.5.1.4.6 of the DSER (SECY-91-355), an interface requirement involving consideration of the potentially unacceptable health hazards in the event of the release of material from these transformers.

As a result of staff review of interface requirements, GE clarified in a letter dated March 11, 1992, that transformers located within fire areas containing safety-related equipment will be of the dry-type only. For those areas utilizing liquid-insulated transformers, the COL applicant will include features to prevent the insulating liquid from becoming an unacceptable health hazard to employees if the material is released to the building environment. The staff will review these features on a plant-specific basis. This was DFSER COL Action Item 9.5.1.4.6-1. GE later submitted SSAR Section 9.5.13.15, requiring the COL applicant to design protective features for liquid-insulated transformers. This is acceptable to the staff.

The use of non-combustible material and the use of dry-type transformers located within fire areas containing safety-related equipment conforms to BTP CMEB 9.5-1 and is acceptable.

9.5.1.4.7 Interaction with Other Systems

The aspects of the fire protection design to address interaction with other systems was reviewed in accordance with Section C.3.a of BTP 9.5.1 relative to the separation of safety-related systems from any potential fires in non-safety-related areas, and the existence of separate redundant divisions of safety-related systems not subject to damage from a single fire.

The vulnerability of safe-shutdown equipment to fire protection water is not an issue because the ABWR design includes three trains of safe-shutdown capability and assumes separate fire areas for each shutdown train, which can survive total loss of all equipment in any fire area. Safe-shutdown equipment in the ABWR design requires no special protection from the effects of failures of the fire-protection water-suppression systems. However, GE includes such protection and assumes no credit for it.

Pipe rupture criteria will ensure that the flood inventory in fire-protection piping will not cause damage to safety-related equipment. Drains and sumps in the NPB will be sized to control maximum flood inventory of fire-protection piping.

These provisions comply with the guidelines of BTP CMEB 9.5-1 Section C.5.a for building design with fire barriers and divisional separation of BTP CMEB 9.5-1 and are acceptable.

9.5.1.4.8 Preoperational Testing

All of the active components of the entire plant fire-protection systems are required to pass a preoperational acceptance test in accordance with the appropriate NFPA standard governing design and installation of the system. Components and systems that must pass the preoperational testing before being placed in service include the following:

- fire pumps - controls, flow volume and pressure
- water distribution - flush and hydrostatic
- control valves
- fire-detection and alarm systems, including electronic supervision for other fire-detection and fire-suppression systems
- water fire-suppression systems
- emergency radio communication systems
- emergency lights
- emergency breathing air systems and components

9.5.1.5 Administrative Controls

The staff will perform a detailed review of the administrative controls for various plant operations during the plant-specific licensing process of a COL application referencing the GE ABWR design. Items of interest under the administrative controls review will include:

- control of combustible materials such as combustible or flammable liquids and gases, fire-retardant-treated wood, plastic materials, and dry ion exchange resins
- transient combustible materials and general housekeeping, including health physics materials

- open-flame and hot-work permits, and cutting and welding operations

- quality assurance for fire-protection systems components, installation, maintenance, and operation

- qualification of fire-protection engineering personnel, fire brigade members, and maintenance and testing personnel for fire-protection systems

- instruction, training, and drills for fire brigade members

The staff designated this as COL Action Item 9.5.1.5-1 in the DFSER. GE added SSAR Section 9.5.13.18 to state that the COL applicant will submit the description of administrative controls outlined in SSAR Section 9.5.1.6.4. This is acceptable to the staff.

9.5.1.6 Summary

The staff finds the fire-protection design for the ABWR acceptable. As discussed above, the fire-protection features described in SSAR Section 9.5.1 and SSAR Appendix 9A (Fire Hazards Analysis) of the ABWR application conform to the applicable sections of CMEB BTP 9.5-1 and meet the applicable regulation discussed in Section 9.5.1 above.

9.5.2 Communications Systems

This topic is discussed in Section 7.7.15 of this report.

9.5.3 Lighting Systems

This topic is discussed in Section 8.3.5 of this report.

9.5.4 Diesel Generator Fuel Oil Storage and Transfer System

Section 9.5.4.1 of this report addresses compliance of all the diesel generator auxiliary systems with the requirements of GDC 2, 4, and 5. Section 9.5.4.2 of this report addresses issues specific to the diesel generator fuel oil storage and transfer system.

9.5.4.1 Diesel Generator Auxiliary Support Systems (General)

There are three standby (emergency) diesel generators (DGs) in the ABWR design. Each DG has the following auxiliary systems, which are addressed in detail in the sections of this report indicated: fuel oil storage and transfer (FOST) (Section 9.5.4.2), cooling water (DGCW) (Section 9.5.5), starting air (DGSA) (Section 9.5.6), lubrication (DGLS) (Section 9.5.7), and combustion air intake and exhaust (DGCA) (Section 9.5.8).

Adequacy of the systems is based on compliance with the requirements of GDC 2 (protection against natural phenomena), GDC 4 (protection from environmental and dynamic effects of equipment failure), and GDC 5 (sharing of SSCs between units) as well as the recommendations of NUREG/CR-0660, "Enhancement of Onsite Emergency Diesel Generator Reliability." Compliance with the requirements of other GDC will be reviewed on a system-specific basis in later sections of this report.

The diesel engine vendor has not been selected, therefore, the interaction between the diesel engine and the auxiliary systems cannot be fully defined. The staff will evaluate the COL applicant's design of specific engine and support systems on a plant-specific basis.

Most components of the diesel generators and their auxiliary support systems are located in the seismic Category I reactor building structure, which will protect against the effects of tornados, missiles, and floods. Each diesel generator and its associated auxiliary systems is physically and electrically separated and is located in separate divisional areas of the reactor building. Thus, a failure of equipment in one division cannot adversely affect the safety function of more than one division of the diesel generator and its associated auxiliaries. The diesel generator exhaust silencer is located on top of the reactor building, well above the probable maximum flood level and is designed to function during design-basis events such as seismic vibrations, wind, hail, tornados, rain, and snow storms. Fuel oil storage tanks, pump motors, valves, and piping are located underground and are of seismic Category I construction. The only portions of the FOST system located above ground are the fill, sample, and vent lines. In the DSER (SECY-91-355), the staff concluded that GE should submit additional information on provisions to minimize the effect of tornado missiles for these exposed components. SSAR Section 9.5.4 states that the FOST system is protected from damage from flying debris carried by tornados. Subject to confirmation that the fill, vent, and sample connections were adequately protected from tornado missiles, the staff concluded in the DFSER that the system designs met the requirements of GDC 2 and 4, and RGs 1.115 and 1.117. This was DSER Open Item 80 and was also identified as DFSER Open Item 9.5.4.1-1. GE submitted Amendment 24 of the SSAR, which resolved this issue, by stating that the fill, vent, and sample connections are located outside of the seismic Category I buildings at or slightly above plant grade. It is highly unlikely that one tornado missile could damage more than one redundant division because (1) these connections are not part of the fuel path from the storage tank to the diesel, (2) the fuel path is located entirely within the reactor building, and (3) the connections for each train are widely spaced around the outside of the reactor building. The staff finds the design features acceptable because the design complies with GDC 2 and 4 of the diesel generator auxiliary systems. Therefore, DFSER Open Item 9.5.4.1-1 is resolved.

The requirements of GDC 5 regarding the sharing of SSCs do not apply because the ABWR is designed as a single-unit facility. An application for a multi-unit facility will require review of the design for compliance with GDC 5.

The staff evaluated the design of the diesel generator support (auxiliary) systems for the effects of postulated pipe failures as discussed in Section 3.6 of this report. The adequacy of the fire protection for diesel generators and the associated auxiliary support systems is addressed in Section 9.5.1 of this report.

The applicant referencing the ABWR design will provide information to ensure compliance with the recommendations of NUREG/CR-0660, "Enhancement of Onsite Emergency Diesel Generator Reliability." The staff will review this information on a plant-specific basis. This was identified as DFSER COL Action Item 9.5.4.1-1.

The TS in SSAR Chapter 16 specify test intervals for the diesel generator auxiliary systems. Most auxiliary systems are tested during diesel generator start/run tests as required by RG 1.108. The TS prescribes additional tests for the fuel oil transfer system. The incorporation of these tests in the TS will resolve the open item in DSER Section 9.5.4.1 regarding the test frequency. This was TS Compliance Item 9.5.4.1-1. GE provided its set of TS in accordance with the standardized technical specifications for BWRs. Chapter 16 of this report discusses the evaluation of the TS. Based on the approval of the ABWR TS, this TS item is resolved.

Instrumentation is to be located in dust-tight steel cabinets with gasketed doors/openings and filtered louvers where ventilation is required. Ventilation is to be taken from a location high in the reactor building, approximately 11.5 m (38 ft) above grade. Construction-related activity will be required to use appropriate dust control techniques. Concrete flooring is to be painted with concrete or masonry paint.

GE submitted the design description and the ITAAC relating to the emergency diesel generator and its auxiliaries. This was identified as DFSER Open Item 9.5.4.1-2. GE provided a revised set of design description 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.

This section applies to the emergency diesel generator auxiliary systems, which include the diesel generator fuel oil storage and transfer system (FOST), diesel generator cooling water system (DGCW), diesel generator starting air (DGSA), diesel generator lubrication (DGL), and diesel generator combustion air intake and exhaust (DGCA). The staff reviewed these auxiliaries for conformance of the design criteria and 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 plant design is acceptable and meets the requirements of GDC 2 as it relates to the ability of the auxiliaries and the structures housing them to withstand the effects of natural phenomena such as earthquakes, tornados, hurricanes, and floods; GDC 4 as it relates to structures housing the auxiliaries and the auxiliaries themselves being capable of withstanding the effects of externally and internally-generated 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. In addition, GE has provided adequate guidance for an applicant referencing the ABWR design to ensure compliance with the recommendations of NUREG/CR-0660.

The staff concludes that the design of the auxiliaries conforms to the above-mentioned GDC and the guidelines of SRP 9.5.4 and is, therefore, acceptable.

9.5.4.2 Diesel Generator Fuel Oil Storage and Transfer System

The staff reviewed the FOST system in accordance with SRP Section 9.5.4. The system should be designed to meet the requirements of GDC 2 (protection against natural phenomena), GDC 4 (protection from environmental and dynamic effects of equipment failure), GDC 5 (sharing of SSCs between units), and GDC 17 (availability of electric power systems). The ability of the FOST system to meet GDC 2, 4, and 5 is discussed in Section 9.5.4.1 of this report.

The FOST system provides a separate and independent fuel oil supply division for each diesel generator. The FOST system provides minimum storage capability for full-load operation of each diesel generator for 7 days without replenishment of fuel. GE submitted the acceptance criteria of the diesel generator support systems.

The ABWR design includes three standby diesel generators. Each diesel engine fuel oil and transfer division consists of a day tank with sufficient capacity to supply fuel oil to power the diesel for 8 hours; a fuel oil storage tank with a capacity sufficient to power the diesel for 7 days; two fuel oil transfer pumps, an engine-driven and a redundant dc motor-driven fuel oil pump (both gravity fed) supplying fuel from the day tank to the engine fuel manifold; and associated piping, valves, instrumentation, and controls. Each FOST system division is independent and physically separated from the other two divisions, and each division is located in a separate quadrant of the reactor building. Thus, a single failure within any one of the divisions will affect only the associated diesel generator.

The SSAR states that selection of the fuel oil transfer pump is an interface requirement. Upon further evaluation, the staff determined that this requirement can be accomplished by identifying a COL action item in the SSAR for the applicant referencing the ABWR design to submit this information. Therefore, this interface requirement was reclassified as DFSER COL Action Item 9.5.4.2-1. Subsequently, GE added design information for one of the fuel oil transfer pumps. The system will consist of an engine-driven fuel oil transfer pump and a second electric motor-driven transfer pump. This is acceptable.

The staff noted as DSER (SECY-91-355) Open Item 81, that the type of motive power (required to be available during a LOOP) should be provided in the SSAR. GE amended the SSAR to identify the motive source for the pump as Class 1E bus power from its respective diesel generator. This resolved DSER Open Item 81.

All FOST system piping and components up to the diesel engine interface are designed to seismic Category I requirements. All piping and components (including engine-mounted) meet RG 1.29, and will be designed, fabricated, and installed in accordance with ASME Code Section III, Class 3, requirements.

Instrumentation provided for the FOST system includes level indication for the day tank, temperature sensors at the intake and discharge of the day tank, and pressure indication for the suction of the engine-mounted and dc motor-driven fuel oil pumps. In the DSER (SECY-91-355), the staff indicated as part of DSER Open Item 83 that the sensor on the tank discharge did not appear on the fuel oil system P&ID. GE submitted a revised figure showing this temperature sensor, which resolved this part of DSER Open Item 83. However, incorporation of the revised figure in the SSAR was identified as DFSER Confirmatory Item 9.5.4.2-1. Subsequently, GE modified Figure 9.5-6 in Amendment 25 of the SSAR which included this temperature sensor. Therefore, Confirmatory Item 9.5.4.2-1 is resolved.

Level sensors provide signals to start the fuel oil transfer pumps, one starting on low level, a second on low-low level. At the low level, a 60-minute supply (at full diesel generator load) of fuel oil is available for diesel generator operation. GE did not state whether storage tank level instrumentation was available. In describing its commitment for a stick gauge provision, GE stated that level switches are provided to monitor tank level. In the DSER (SECY-91-355), the staff identified inconsistencies in the listing of these level switches between parts of SSAR Sections 9.5.4 and 8.3. GE committed to include the level switches and the stick gauge in the appropriate parts of SSAR Section 9.5.4, Section 8.3.1.1.8.5, and the referenced figures. This commitment resolved DSER Open Item 82; however, incorporation of this information into the SSAR was identified as DFSER Confirmatory Item 9.5.4.2-2. GE submitted the necessary tank level information in SSAR Amendments 21, 22, and 25. Table 8.3-5 includes an alarm on low storage tank level, Section 9.5.4.5 discusses level alarms on the storage and day tanks, and Figure 9.5-6 includes the level sensor and stick gauge. The staff finds these modifications acceptable. Therefore, Confirmatory Item 9.5.4.2-2 is resolved.

The fuel oil storage tanks are located underground in three separate areas adjacent to the reactor building. The interior and exterior of the tanks and associated buried piping will have a protective waterproof coating. Also, the design will use an impressed current-type cathodic protection to control corrosion of underground piping.

SSAR Figure 9.5-6 depicts the standby diesel generator fuel oil system. From the review of this figure, the staff concluded in the DSER (SECY-91-355), that the fuel oil storage tanks and associated instrumentation should be added to the figure. In addition, the staff found discrepancies between the text and Figure 9.5-6, regarding the optional characterization of the electric fuel oil pump. Responding to RAI 430.273, GE stated, in part, that "two local fuel oil temperature indicators are provided (one in the suction line and one in the discharge line) from the day tank." Figure 9.5-6, however, shows only one temperature sensor. GE committed to include a revised Figure 9.5-6 in the SSAR that will include the fuel storage tanks and their associated instrumentation. This commitment resolved the other part of DSER Open Item 83. However, incorporation of these revisions into the SSAR was identified as DFSER Confirmatory Item 9.5.4.2-3. GE submitted the revised figure in Amendment 25 of the SSAR. The figure incorporates the tanks, their associated instrumentation, and temperature sensors at the suction and discharge of the day tanks, and contains a footnote which allows the motor-driven pump to be added as an option. The staff finds the revisions acceptable. Therefore, Confirmatory Item 9.5.4.2-3 is resolved.

Section III.5 of SRP Section 9.5.4 addresses the need to minimize the creation of sediment turbulence at the bottom of the fuel oil storage tank during refueling. To ensure continuous operation of the diesel generator while refueling, the ABWR design relies on duplex filters and strainers between the storage tank and the day tank and at the fuel oil pump suction to remove any sediment. The SSAR suggests that refueling would probably occur while the day tank is full, which would allow time for sediment to settle before fuel is transferred from the storage tank to the day tank.

In the DSER (SECY-91-355), the staff stated that refueling procedures should be established as an interface requirement to verify that the day tank is full before refilling the storage tank, thereby minimizing the likelihood of sediment obstruction of fuel lines and any deleterious effects on diesel generator operation. This was identified as DSER Open Item 84. Upon further evaluation, the staff determined that this requirement can be accomplished by adding a COL action item requiring these procedures. As a result, this interface was reclassified as DFSER COL Action Item 9.5.4.2-2. GE included this information in the SSAR, which is acceptable.

GE described a program to ensure that the diesel fuel oil is tested and maintained according to the appropriate ASTM and ANSI requirements. Fuel oil is to be sampled and tested monthly for quality and contaminants. New fuel will be visually inspected before being added to the storage tank, and will be analyzed within 2 weeks for other required properties. Fuel oil not meeting all requirements will be replaced within a week. The system will be tested as part of the required diesel generator tests and hydrostatically tested before startup. Each fuel storage tank will be tested to ASME requirements every 10 years. The system design fuel oil quality and tests meet the requirements of RG 1.137. Based on the above information, the staff concludes that the FOST meets the requirements of GDC 17.

The design of the FOST system will meet the requirements of GDC 17, as related to the capability of the fuel oil system to meet independence and redundancy requirements, RG 1.9 and RG 1.137. As discussed in Section 9.5.4.1 of this report, this auxiliary system also meets the requirements of GDC 2, 4, and 5 and will incorporate the recommendations of NUREG/CR-0660. The system will also incorporate appropriate industry standards, i.e., ANSI N195-1976 and IEEE Standard 387. Therefore, the staff concludes that the diesel generator fuel oil storage and transfer system meets the guidelines of SRP 9.5.4 and is acceptable.

9.5.5 Diesel Generator Cooling Water System

The staff reviewed the DGCW system in accordance with the GDC in SRP Section 9.5.5.

The function of the diesel generator cooling water system is to maintain the temperature of the diesel engine within a safe operation range under all load conditions and to maintain the engine coolant preheated during standby conditions. The system should be designed to meet the requirements of GDC 2 (protection against natural phenomena), GDC 4 (protection from environmental and dynamic effects of equipment failure), GDC 5 (sharing of structures, systems, and components between units), GDC 17 (availability of electric power systems), GDC 45 (inspection of cooling systems), and GDC 46 (testing of cooling systems). The ability of the ABWR diesel generator cooling water system to meet GDC 2, 4, and 5 is discussed in Section 9.5.4.1 of this report.

The DGCW system is a closed-loop system that cools the engine-jacket water, lube oil, and combustion air. The major components of this system include a jacket water heat exchanger, lube oil heat exchanger, combustion air heat exchangers (air intercooler and exhaust manifold), an expansion tank, two jacket water circulating pumps, an electric immersion heater, a jacket water keep-warm system, a three-way temperature control valve, and the required controls, alarms, instrumentation, piping, and valves. Heat generated during diesel generator operation is rejected to the RCW system through the jacket-water heat exchanger. All system piping and components are designated ASME Code Section III, Class 3, designed to seismic Category I requirements, and will be procured according to the requirements of 10 CFR Part 50, Appendix B.

The ABWR design includes three standby diesel generators, each having a physically separate and independent engine cooling water division, as described in the preceding paragraph. Each cooling water division is powered from the respective diesel generator's safety-related Class 1E motor-control center. Therefore, the system meets the redundancy and single-failure criteria requirements of GDC 17.

During operation of the diesel generators, the temperature of the diesel engine cooling water is regulated by three-way temperature control valves. When the standby diesel generators are idle, the cooling water is heated by an electric heater and maintained at 49 °C (120 °F), assuming ambient temperature of 16 °C (60 °F).

In the DSER (SECY-91-355), the staff stated that GE established interface requirements for specific information on the design and capability of the cooling water system and the keep-warm system including the following:

(1) jacket-water circulating pump characteristics (NPSH and motive-power source, i.e., shaft, engine, etc.)

(2) the keep-warm system description (design may or may not include a keep-warm pump)

(3) temperature sensor selection (Amot-type or equivalent)

(4) heat removal capability of system (to be based on maximum permissible diesel engine overload output)

(5) expansion tank size

(6) expected water loss over 7-day period and system volume capacity needed to ensure adequate volume is available to maintain system water level and pump NPSH without refill.

These requirements were identified as DFSER Open Item 9.5.5-1. Upon further evaluation, the staff determined that these requirements do not meet the definition of interface requirements implied in 10 CFR Part 52 and can be accomplished by adding a COL action item in the SSAR requiring the COL applicant referencing the ABWR design to submit this information. GE included this information in the SSAR, which is acceptable. This resolved DFSER Open Item 9.5.5-1.

In the DSER (SECY-91-355), the staff discussed a discrepancy between the text of SSAR Section 9.5.5 and Figure 9.5-7 regarding the circulating water pump. Section 9.5.5 described the jacket-circulating water pumps as engine- and motor-driven while Figure 9.5-7 described both as being motor-driven. (This discrepancy was considered part of DSER Open Item 85.) Additionally, the interface criteria of Section 9.5.13.6 stated that the selection of the motive power for these pumps was an interface requirement. This disagreed with the text of Section 9.5.5 and Figure 9.5-7, which clearly specified motive power for the pumps, although in an inconsistent manner. (This was the other part of DSER Open Item 85.) GE committed to submit consistent information regarding the power source for the jacket-cooling water pumps. The references to specific power sources were to be removed, and the selection of the power supply was to be incorporated as DFSER COL Action Item 9.5.5-1. These changes resolve DSER Open Item 85; however, incorporation of these changes into the SSAR was identified as DFSER Confirmatory Item 9.5.5-1. The staff reviewed Amendment 23 of the SSAR and concludes that the modification to Section 9.5.13.6, requiring the COL applicant to submit information on the motive power for the cooling water pumps, is acceptable. Therefore, Confirmatory Item 9.5.5-1 is resolved.

The interface criteria for selecting this valve did not specify an Amot-type or equivalent valve. The staff concluded in DSER (SECY-91-355) Open Item 86 that GE should include this selection information as DFSER COL Action Item 9.5.5-1 for temperature sensor selection. GE committed to incorporate the reference to an Amot-type temperature sensor, or equivalent into the SSAR. Therefore, DSER Open Item 86 was resolved. However, incorporation of this reference into the SSAR was identified as Confirmatory Item 9.5.5-2 in the DFSER. The staff reviewed Amendment 23 of the SSAR and concludes that the modification to Section 9.5.13.6, requiring the COL applicant to submit an Amot-type temperature sensor or its equivalent, is acceptable. Therefore, Confirmatory Item 9.5.5-2 is resolved.

The SSAR states that the diesel engine has the capability to operate at full load for 2 minutes without secondary cooling. This will ensure that the diesel engine can operate at full load in excess of the time required to restore cooling water (RSW and RCW), which are sequenced onto the emergency power supply within 1 minute after a LOPP.

The DGCW system conforms with RG 1.9 for engine cooling water protective interlocks. All trips are bypassed during LOCA conditions except low cooling water pressure and low differential pressure of secondary cooling water. Both of these trips are 2-out-of-2 logic trips (the diesel generator system protective interlocks are discussed in Section 8.3.3.7 of this report). The cooling water system is provided with an expansion tank and expansion tank vent line, both of which are to be located above the system piping and pump location. A static head will ensure that the pumps and piping are filled with water. On the basis of the discussion for resolution of DSER (SECY-91-355) Open Items 85 and 86, the system meets the requirements of GDC 44 regarding the provision for cooling systems.

The SSAR includes a commitment that the operating procedures for the diesel generator will require the loading of the engine up to a minimum of 40-percent of full load (or lower as specified in the manufacturer's recommendation) for 1 hour after up to 8 hours of continuous no-load or light-load operation. Such no-load or light-load conditions would exist for a LOCA with offsite power available. Procedures including this criterion will specifically meet the guidance listed in Item III.7 of SRP Section 9.5.5.

The components of the diesel engine cooling water system can be periodically inspected through surveillance testing and monitoring instrumentation for pressure, temperature, and level. The system cooling water would be analyzed periodically to ensure that adequate quality is maintained. The diesel generator would be tested in accordance with the requirements of RG 1.108. These commitments meet the inspection and functional testing requirements in GDC 45 and 46.

The design criteria and bases for the DGCW system will conform to GDC 17 and 44 for redundancy, physical independence, and cooling capability, and GDC 45 and 46 for inspection and testability of the system. As stated in Section 9.5.4.1 of this report, the design meets the requirements of GDC 2, 4, and 5, for the protection of equipment from environmental and dynamic effects and sharing of system components. In addition, GE has provided adequate guidance for an applicant referencing the ABWR design to ensure compliance with the recommendations of NUREG/CR-0660. The system meets the requirements of GDC 45 and 46 for inspection and testability. The design of the DGCW system will meet the requirements of GDC 44 for system operability. Therefore, the staff concludes that the system design meets the guidelines as specified above of SRP Section 9.5.5 and is acceptable.

9.5.6 Diesel Generator Starting Air System

The staff reviewed the DGCA system in accordance with SRP Section 9.5.6.

The design function of the DGSA system is to provide a supply of compressed air for starting the emergency diesel generator engines without external power. The air storage system is to perform its function in a manner that ensures that the time interval between a diesel engine start signal and a "ready-to-load" status is less than 20 seconds. The system is to be designed to meet the requirements of GDC 2 (protection from natural phenomena), GDC 4 (protection from the environmental and dynamic effects of equipment failure), GDC 5 (sharing of SSCs between units), and GDC 17 (availability of electric power systems). Compliance with the requirements of GDC 2, 4, and 5 is discussed in Section 9.5.4.1 of this report.

The ABWR design includes three emergency diesel generators, each of which has its own starting air division, separate and independent of the starting air divisions for the other two diesel generators. Each starting air division consists of two 100-percent capacity sections, each of which is capable of supplying sufficient air for five automatic or manual start attempts without recharging the air-receiver tanks. Each starting air division consists of two air-compressors, two air-receivers, four air-admission valves (two redundant air-admission valves on each of two engine starting air manifolds), and associated piping and valves to connect system components.

One division of the DGSA system consists of two compressors and two air receivers. Controls are provided to automatically start and stop each air compressor to maintain the required pressure in each air receiver. Each compressor can be manually started to restore pressure in the air-receivers if needed. Each division is equipped with an air-receiver low-pressure alarm, which is indicated locally and displayed in the control room as part of a diesel generator trouble alarm. Each receiver is also equipped with a blowdown connection, which would be used periodically to manually blow down the receiver to remove any accumulated water from the tank.

In Amendment 16 of the SSAR, GE responded to Q430.285, stating that each air-dryer system includes an air dryer equipped with pre-filters and after- filters. SSAR Figure 9.5-8 did not specifically identify pre-filters and after-filters for the air dryers. Addition of these filters to the P&ID (or a statement specifically identifying the filters as an integral part of the air-dryer component) was identified as DSER (SECY-91-355) Open Item 88. GE committed to revise SSAR Figure 9.5-8 to incorporate the pre- and after-filters into the design. This modification resolved DSER Open Item 88. However, incorporation of the modification into the SSAR was identified as DFSER Confirmatory Item 9.5.6-1. Subsequently, the staff reviewed Amendment 25 of the SSAR and concluded that the addition of note 4 on Figure 9.5-8 clarified the provisions for pre- and after-filters on the air dryers. Therefore, Confirmatory Item 9.5.6-1 is resolved.

Each diesel generator is completely separate and independent of the others so that a malfunction or failure in one starting air division does not impair the starting capability of the other diesel generators. Therefore, the design meets the independence and redundancy requirements of GDC 17.

GE identified several design parameters as future interface requirements to be determined once a diesel generator vendor has been selected. Interface requirements to be specified included the devices to crank the engine, the duration of the cranking cycle, and the number of engine revolutions per start attempt. These interface requirements would dictate design parameters such as the volume and design pressure of the air receivers (sufficient for five start cycles for each receiver) and compressor size (sufficient discharge flow to recharge the system in under 30 minutes). Once established, these interface criteria would provide an adequate basis for selecting component capacities. This was identified as DSER (SECY-91-355) Open Item 89 and incorporation of this information into the SSAR was identified as DFSER Confirmatory Item 9.5.6-2. Upon further evaluation, the staff has determined that this requirement can be accomplished by including a COL action item in the SSAR requiring the COL applicant to submit this information. GE included this information in the SSAR. Therefore, DSER Open Item 89 and DFSER Confirmatory Item 9.5.6-2 are resolved.

The air compressor's air-storage tanks, valves, and piping (up to the first connection on the engine skid) are designed in accordance with the requirements of ASME Code Section III, Class 3, requirements and are seismic Category I.

The DGSA system description did not include a reference to coolers at the discharge of the air compressors, although SSAR Figure 9.5-8 includes after-coolers located downstream of the starting air compressors. This discrepancy was identified as DSER (SECY-91-355) Open Item 90. GE committed to incorporate the coolers into the description of the starting air system. This modification resolved DSER Open Item 90; however, incorporation of the modifications into the SSAR was identified as DFSER Confirmatory Item 9.5.6-3. The staff reviewed Amendment 23 of the SSAR and concluded that Section 9.5.6.2 contained the required information regarding the coolers. The staff reviewed Figure 9.5-8 (Amendment 25 of the SSAR) and concluded that the after-coolers were adequately identified. The staff finds the SSAR modifications acceptable. Therefore, Confirmatory Item 9.5.6-3 is resolved.

The SSAR states that the starting air quality complies with the diesel engine manufacturer's recommendation regarding dew point, as opposed to the requirements stated in SRP Section 9.5.6 II.4.j. The staff will determine whether the system's air quality complies with SRP Section 9.5.6.II.4.j on a plant-specific basis.

Based on the above review, the staff concludes that the DGSA system meets the requirements of GDC 17 as it relates to the availability of electric power. As discussed in Section 9.5.4.1 of this report, this auxiliary system also meets the requirements of GDC 2, 4, and 5 and will incorporate the recommendations of NUREG/CR-0660. The system, therefore, meets the guidelines of SRP 9.5.6 and is acceptable.

9.5.7 Diesel Generator Lubrication System

The design safety function of the DGLS system is to provide a supply of filtered lubrication oil to the various moving parts of the diesel engine (including pistons and bearings) during engine operation and during periods of standby to enhance first-try-start reliability. The basis for acceptance in the review was conformance of the design to GDC 17, regarding redundancy and physical independence, and the guidance and additional acceptance criteria of SRP 9.5.7. The staff discusses the ability of the system design to meet the requirements of GDC 2 (protection from natural phenomena), GDC 4 (protection from environmental and dynamic effects of equipment failure), and GDC 5 (sharing of SSCs between units) and the recommendations of NUREG/CR-0660 in Section 9.5.4.1 of this report.

The major components of the lubrication system include the engine lube oil pump (within the engine frame), an engine-driven pump, an oil cooler, a generator shaft lube oil cooler, an electric lube oil heater, a keep-warm circulating pump, oil filter, and strainer. Local alarms signal low oil pressure, high oil temperature, and low oil level. These signals are part of a general diesel generator trouble alarm located in the control room. The low lubrication oil level alarm is identified in SSAR Table 8.3-5, and refill is described in SSAR Section 9.5.7 as being performed on indication of low level (a lube oil supply pump actuates on a low level indication in the engine sump). In DSER (SECY-91-355) Open Item 91, the staff stated that SSAR Figure 9.5-9 did not show any level indication for the lube oil system. GE committed to correct the figure to include the level instrumentation, which had been mistakenly identified as a flow transmitter. This correction resolved DSER Open Item 91; however, incorporation of the modifications into the SSAR was identified as DFSER Confirmatory Item 9.5.7-1. GE supplied the revised figure in SSAR Amendment 25. The staff finds the revised figure acceptable. Therefore, Confirmatory Item 9.5.7-1 is resolved.

Each of the DGLS divisions is completely independent of the other divisions and is dedicated to the support of a single diesel generator. A malfunction in a DGLS division will not impair the operational capability of the remaining lubrication divisions or diesel generators. This meets the requirements of GDC 17 for system independence and the single-failure criteria.

The system is designed to provide sufficient lubricating oil to support full-load diesel operation for 7 days.

In DSER (SECY-91-355) Open Item 92, the staff requested that GE submit the following specific design criteria: pump flows, operating pressure, temperature differentials, cooling system heat removal capabilities and electric heater characteristics for the DGLS. GE stated that the "lubrication system design criteria will be furnished as an interface criteria after selection of the diesel vendor is finalized." Upon further evaluation, the staff determined that this requirement can be accomplished by identifying a COL action item in the SSAR requiring the COL applicant to submit this information. As a result, the interface was reclassified as DFSER COL Action Item 9.5.7-1. The staff reviewed Amendment 23 of the SSAR (Section 9.5.13.5) and concludes that the COL action item would ensure that the COL applicant submits the required information. GE included this COL action item in the SSAR. This is acceptable.

GE stated that the protective features to prevent crankcase explosions and features to mitigate the consequences of such an event (such as relief ports) are vendor specific and would also be included as interface criteria after a diesel engine vendor is selected. The diesel generator would be protected from crankcase explosions by activating vacuum blowers to maintain the crankcase at negative pressure and shutting down the diesel on high-pressure conditions (unless a LOCA signal is present). Inclusion of these design criteria was identified as Open Item 92 in the DSER (SECY-91-355). GE committed to incorporate the above system design characteristics into the SSAR for use when the diesel vendor is selected. This information was to be incorporated into Section 9.5.8 of the SSAR and resolved DSER Open Item 92. Incorporation of the modifications into the SSAR was identified as DFSER Confirmatory Item 9.5.7-2. Upon further evaluation, the staff determined that these requirements should be included in the system design. GE modified SSAR Section 9.5.8 (Section 9.5.8.2) and Table 8.3-5 to include these design features. Therefore, Confirmatory Item 9.5.7-2 is resolved.

The DGLS is designed to maintain lubrication oil temperature and circulate heated lubrication oil under pressure to the moving parts of the diesel engine while the engine is in the standby mode. A lube oil priming pump will operate intermittently to keep the lube oil piping pressurized. This same pump is used in conjunction with the lube oil heater to maintain system temperature. On low lube oil temperature, both the heater and priming pump will automatically start thereby circulating heated oil throughout the system. The priming pump discharge pressure switch is Class 1E. Based on the above information, the staff concludes that the DGLS conforms to the recommendations of NUREG/CR-0660, for enhancing the starting reliability of the diesel engine.

All DGLS piping and components will be designed in accordance with ASME Code Section III, Class 3, requirements or ANSI B31.1 guidance and will be seismic Category I. GE did not designate the diesel engine interface for the lubrication system because the diesel engine is vendor-specific. Therefore, the components to be designated to meet ASME Code requirements, have not been separated from those required to meet the ANSI standard. To meet NRC requirements, all components up to the diesel engine interface must meet the ASME Code requirements. The NRC staff has accepted the ANSI classification for engine-mounted components if they are pressure tested to 1.5 times design pressure and the test is documented. Recognizing that the keep-warm heater and the priming pump do not have to be nuclear safety grade, the staff identified classification of components in the lubrication system as Open Item 93 in the DSER (SECY-91-355). The staff established Open Item 94 to have GE clarify which components meet the ASME Code requirements and which meet the ANSI requirements with the pressure testing provision. GE committed to revise the response to RAI 430.271 to clearly state which components are to meet the ASME Code requirements and which are to meet the ANSI requirements. Engine-mounted components are to be designed to the ANSI standards, all others, except the keep warm-heater and the priming pump, are to meet the ASME standard. The proposed modification resolved DSER Open Items 93 and 94; however, incorporation of the changes into the SSAR was identified as DFSER Confirmatory Item 9.5.7-3. The staff reviewed the revised response to RAI 430.271 in Amendment 22 of the SSAR and the revisions to Figure 9.5-9 in Amendment 25 of the SSAR. The staff found that the modifications to the RAI and the figure clearly state which components are to meet the ASME Code requirements and which are to meet the ANSI requirements. Therefore, Confirmatory Item 9.5.7-3 is resolved.

The DGLS conforms to RG 1.9 for the protective interlocks for the diesel engine lubrication system. All trips associated with the lubrication system are bypassed during LOCA conditions. The diesel generator system protective interlocks are discussed in Section 8.3 of this report.

The quality of the lubrication oil is maintained through periodic sampling and analysis of the lubrication oil. Access to the lubrication system is controlled. The system is located in locked diesel generator rooms, thereby limiting the possibility of contamination.

The staff concludes that the design of the DGLS meets GDC 2 (protection from natural phenomena), GDC 4 (protection from environmental and dynamic effects of equipment failure), and GDC 17 (availability of electric power), the guidelines of SRP Section 9.5.7, and the recommendations of NUREG/CR-0660 (GDC 5 does not apply to the ABWR design because it is a single-unit design). The DGLS design, therefore, is acceptable.

9.5.8 Diesel Generator Combustion Air Intake and Exhaust System

The design function of the DGCA system is to supply filtered air for combustion to the engine and to dispose of the engine exhaust to the atmosphere. Acceptance is based on conformance of the design to GDC 17, regarding redundancy and physical independence, the guidelines of SRP Section 9.5.8, the recommendations of NUREG/CR-0660, and industry codes and standards. The staff also assessed the ability of the system to supply sufficient combustion air and release sufficient exhaust gases to enable the emergency diesel generator to perform on demand. The compliance of the system design with the requirements of GDC 2 (protection from natural phenomena), GDC 4 (protection from environmental and dynamic effects of equipment failure), and GDC 5 (sharing of SSCs between units) is discussed in Section 9.5.4.1 of this report.

Combustion air for each diesel generator is taken from the associated inlet air cubicle above the diesel generator room through floor grates into the combustion air inlet plenum, duct work, intake silencer, turbocharger and air intercooler. The exhaust gas passes through the turbocharger and the exhaust ducting to the exhaust silencer located on the roof of the reactor building. Each of the three diesel generators is provided with a separate and independent combustion air intake and exhaust division. There are no active components (such as louvers) that can fail and obstruct the inlet or outlet air flow paths. Thus, the system's independence, redundancy, and single- failure criteria meet the requirements of GDC 17. System design air flow capacity has not been specified in the SSAR. As with the other diesel generator auxiliary systems, this design characteristic will be dependent on selection of a diesel generator vendor.

The staff designated selection of a combustion air flow capacity sufficient to ensure complete combustion as an interface requirement and as Open Item 95 in the DSER (SECY-91-355). Upon further evaluation, the staff determined that this requirement can be accomplished by including a COL action item in the SSAR requiring the COL applicant to submit this information. The interface was reclassified as DFSER COL Action Item 9.5.8-1. GE included this information in the SSAR, which is acceptable.

The DGCA meets the guidelines of RG 1.9, Revision 3 (July 1993) as it relates to system protective interlocks. All DGCA piping and components are designed to seismic Category I and ASME Code Section III, Class 3, requirements. Engine-mounted piping and components beyond the engine interface are considered part of the engine assembly and are seismic Category I as part of the diesel engine package.

The combustion air intakes are located on the side of the reactor building and are protected (by vertical grills) from tornado missiles. The intakes are located 11.5 m (37.7 ft) above grade and are designed to minimize any effects from dust and debris through the use of vertical grills set into the reactor building wall with filters located behind the grills. The intakes are protected from flooding by their location.

The diesel generator exhausts are partly housed within the reactor building with the exhaust silencer located on the roof of the reactor building. The design basis for the silencer requires that it be seismically qualified and able to withstand the effects of tornados. In DSER (SECY-91-355) Open Item 96, the staff stated that the means of protection from tornado missiles had not been adequately discussed. GE committed to update the SSAR to state that the silencers are seismically mounted and bolted in a horizontal position. However, this design change did not adequately address protection of the silencers from tornado missiles. Therefore, this issue was identified as DFSER Open Item 9.5.8-1. In SSAR Section 9.5.8, GE committed to house the system in a seismic Category I structure to protect against tornado-missiles. This resolved DSER Open Item 96; however, confirmation that the SSAR had been updated was identified as DFSER Confirmatory Item 9.5.8-1. The staff concluded that the system complies with GDC 4, RG 1.115, Revision 1 (July 1977) and RG 1.117, Revision 1 (April 1978) and the recommendations of NUREG/CR-0660. The staff reviewed Amendment 26 of the SSAR, which clarified that all parts of the system, except for the silencers, are housed in a seismic Category I structure (the reactor building). The exhaust silencers are seismically mounted and bolted down horizontally. Therefore, the silencers are unlikely to become missiles. The silencers are widely spaced around the top of the building so that it is unlikely that a single missile could damage more than one division of the system. A damaged silencer will not prevent a diesel from performing its safety function unless debris clogs the exhaust pipe. In such a case, the other divisions will be relied on to perform the safety function. The staff finds the additional information sufficient to conclude that the design includes adequate protection for the intake and exhaust system. Therefore, DFSER Open Item 9.5.8-1 and DFSER Confirmatory Item 9.5.8-1 are resolved.

Combustion air is not taken from the diesel generator room. Instead, combustion air and ventilation air enter the reactor building through common filters into the air inlet cubicle. Before the air enters the diesel generator room, it enters separate inlet plenums for ventilation and combustion. This design complies with NUREG/CR-0660 for combustion air and dust and dirt control in the portion of the reactor building housing the diesel generators.

The SSAR did not describe the design of the portion of the DGCA which extends from the crankcase vacuum blowers to the outside environment. DSER Open Item 97 (SECY-91-355) requested additional information regarding this area, identified in SSAR Figure 9.5-6. GE committed to modify the response to RAI 430.294 to state that this part of the system consists of piping only. This information resolved DSER Open Item 97; however, incorporation of the modification into the SSAR was identified as DFSER Confirmatory Item 9.5.8-2. The staff reviewed Amendment 26 of the SSAR. Amendment 26 contained the modified response to RAI 430.294 stating that the gases are exhausted through a 150 mm (6-in.) pipe which passes through the reactor building wall. This additional information, along with Figure 9.5-6, adequately addresses the staff's concern. Therefore, this confirmatory item is resolved.

Based on the above review, the staff concludes that the DGCA system meets the requirements of GDC 17 as it relates to the availability of electric power. As discussed in Section 9.5.4.1 of this report, this auxiliary system also meets the requirements of GDC 2, 4, and 5 and will incorporate the recommendations of NUREG/CR-0660. The system, therefore, meets the guidelines of SRP 9.5.8 and is acceptable.