POLICY ISSUE
(Information)

August 2, 1991 SECY-91-235

For: The Commissioners

From: James M. Taylor
Executive Director for Operations

Subject: DRAFT SAFETY EVALUATION REPORT ON THE GENERAL ELECTRIC BOILING WATER REACTOR DESIGN COVERING CHAPTERS 1, 3, 9, 10, 11, AND 13 OF THE STANDARD SAFETY ANALYSIS REPORT

Purpose: To inform the Commission of the staff's intent to issue selected sections of the draft safety evaluation report (DSER) on the General Electric Company's (GE's) Advanced Boiling Water Reactor (ABWR) design. The staff DSER addresses open items needing closure identified by the staff's review of GE's Standard Safety Analysis Report (SSAR) Chapters 3, 9, 10, 11, and 13.

Background: In SECY-91-153, "Draft Safety Evaluation Report on the General Electric Company Advanced Boiling Water Reactor Design Covering Chapters 1, 2, 3, 4, 5, 6, and 17 of the Standard Safety Analysis Report," which provided the previous DSER sections to the Commission, the staff discussed the ABWR review process and the Commission guidance that is being followed.

Discussion: Consistent with SECY-91-153, the enclosed DSER includes an update of previously issued Chapter 1, portions of Chapters 3, 9, 10, and 13, and Chapter 11 in its entirety. Staff issuance of this report will assist in resolving a number of open items identified by its review. Copies will be provided to the Advisory Committee on Reactor Safeguards (ACRS).

Conclusion: The staff concludes that the enclosed DSER contains no new policy issues. The report will be forwarded to GE to inform

NOTE: TO BE MADE PUBLICLY AVAILABLE IN 10 WORKING DAYS FROM THE DATE OF THIS PAPER

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them of the staff's current findings. The resolution of open items will be reflected by the staff in the final safety evaluation report for the ABWR. The staff will issue and place in the NRC Public Document Room the enclosed DSER in 10 working days of the date of this paper.

Coordination: The Office of the General Counsel has reviewed this paper and has no legal objection.

James M. Taylor
Executive Director
for Operations

Enclosures:
DSERs Chapters 1, 3, 9,
10, and 13

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DRAFT SAFETY EVALUATION REPORT

ON

CHAPTERS 3, 9, 10, 11, AND 13 OF

THE GENERAL ELECTRIC COMPANY'S APPLICATION

FOR CERTIFICATION OF THEIR

ADVANCED BOILING WATER REACTOR DESIGN

prepared by the

Office of Nuclear Reactor Regulation

U.S. Nuclear Regulatory Commission

July 1991

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1 INTRODUCTION AND GENERAL DISCUSSION

1.1 Introduction

This is a safety evaluation of the application submitted by General Electric Co., Inc. (GE) for the final design approval (FDA) and the design certification of its advanced boiling water reactor (ABWR) design.

The U.S. Nuclear Regulatory Commission (NRC) staff (NRC staff or staff) prepared this safety evaluation report (SER). The SER summarizes the results, to date, of the staff's safety review of Chapters 3, 9, 10, 11 and 13 of the ABWR Standard Safety Analysis Report (SSAR). A safety evaluation was previously issued to the Commission on May 24, 1991 which addressed ABWR SSAR Chapters 1, 2, 3, 4, 5, 6, and 17. The NRC Licensing Project Managers for the ABWR are Mr. Victor McCree, Mr. Chester Poslusny, and Mr. Dino Scaletti. They may be reached by calling (301) 492-1118 or by writing to the Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, DC 20555.

This safety evaluation addresses those portions of the SSAR that were reviewed against the Standard Review Plan (SRP) (NUREG-0800) Sections 3, 9, 10, 11, and 13 and the additional Commission guidance provided in the staff requirements memorandum (SRM), dated June 26, 1990, pertaining to SECY-90-016, "Evolutionary Light Water Reactor Certification Issues and Their Relationship to Current Regulatory Requirements," (see Appendix B to the DSER dated May 24, 1991).

See the ABWR DSER dated May 24, 1991 for a discussion of the history of the review effort for the ABWR and a discussion of the scope of the design.

The regulations governing the submittal of standard plant design reviews are contained in 10 CFR 2.110, "Filing and administrative action on submittals for design review or early review of site suitability issues," in 10 CFR Part 52, Subpart B, "Standard Design Certifications," and in Appendix O to 10 CFR Part 52, "Standardization of Design: Staff Review of Standard Designs."

1.2 General Plant Description

To be provided in a supplement.

1.3 Comparison With Similar Facility Designs

See the ABWR DSER dated May 24, 1991.

1.4 Identification of Agents and Contractors

See the ABWR DSER dated May 24, 1991.

1.5 Summary of Principal Review Matters

The SRM related to SECY-90-377, "Requirements for Design Certification Under 10 CFR Part 52," reflects the Commission's position on what level of design information is required in a standard plant application for certification, and the staff has followed the guidance in preparing this document. Thus, for each section of the staff's DSER where the GE SSAR has been found to be acceptable relative to the review criteria and guidance, it indicates that the staff believes that GE has provided a sufficient level of design detail to make its safety finding except as provided below. However, the staff has not completed its review of all other sections of the SSAR such as the probabilistic risk assessment, and has not begun its review of outstanding submittals such as the inspection, tests, analyses, and acceptance criteria (ITAACs) and the severe accident mitigation design alternatives (SAMDA). The results of these and other reviews may require the staff to request additional design detail information for areas which have been found acceptable in this DSER.

1.6 Modifications to the ABWR During the Course of the NRC Review

To be provided in a supplement.

1.7 Unresolved Safety Issues

The staff continuously evaluates the safety requirements which are used in its review against new information as it becomes available. Part 52 of 10 CFR requires that all new power plant designs address all unresolved safety issues (USIs) and all medium-priority and high-priority generic safety issues (GSIs). NUREG-0933, "A Prioritization of Generic Safety Issues," identifies these issues and their status. GE is to identify and address any of these issues that are applicable to the ABWR design and any new generic issues raised until the FDA is issued. The staff intends that there will be no open items remaining for the resolution of USIs or GSIs or other plant features for the ABWR when the NRC makes the FDA decision. The resolution of these issues will be discussed in Appendix C to a later supplement.

1.8 Outstanding Issues

Certain outstanding regulatory review issues for the ABWR remain unresolved with GE as of the date NRC issues this report. Many of these issues are in the unresolved category because the staff has not completed its review of the information GE provided in the SSAR; other issues are unresolved because GE needs to submit additional information. The staff will discuss the resolution of each of these issues in a supplement to this report. These issues and the sections (as indicated by the numbers in parentheses) in a previous SER, this SER, or its supplements in which they are or will be discussed are as follows.

Issue	Status
(1) Classification of containment spray piping (3.2.2)	Awaiting information
(2) Quality Assurance and classification of new and spent fuel storage equipment (3.2.2)	Awaiting information
(3) Seismic classification of main steam line (3.2.2)	Under review
(4) Design basis tornado (3.3.2)	Under review
(5) Flooding of safety-related (SR) buildings (3.4.1)	Under review
(6) Flooding analysis for ultimate heat sink pumphouse (3.4.1)	Under review
(7) Use of induction motors (3.5.1.1)	Awaiting information
(8) Protection from non-safety related component missiles (3.5.1.1)	Under review
(9) Missile protection (3.5.1.2)	Under review
(10) Natural phenomena missiles (3.5.1.4)	Under review
(11) Tornado design features (3.5.2)	Under review

Issue	Status
(12) Protection against piping failure (3.6.1)...........................	Under review
(13) SSAR Table 3.9-8 (3.9.6).................	Under review
(14) Program for inspection of SR pumps (3.9.6.1)..........................	Awaiting information
(15) Program for inspection of SR check valves (3.9.6.2.1).................	Awaiting information
(16) Prototype testing of motor operated valves (3.9.6.2.2)..............	Awaiting information
(17) Program for inspection of SR motor operated valves (3.9.6.2.2)..............................	Awaiting information
(18) Listing of isolation valves (3.9.6.2.3)..............................	Awaiting information
(19) Compliance with ASME Code leak rate analysis requirements (3.9.6.2.3)..............................	Awaiting information
(20) SSAR Table 3.9-9 (3.9.6.2.3)	Under review
(21) Equipment qualification time margin (3.11.3.2.1).................	Under review
(22) Appendix 3I (3.11.3.3)...................	Awaiting information
(23) Control rod acceptance criteria (4.2)....	Awaiting information
(24) Standby liquid control system reliability analysis and ATWS Rule compliance (4.6).........................	Under review
(25) Final report on the in-plant test program for the fine motion control rod drive system (4.6)...........	Resolved
(26) Hydraulic control unit design information (4.6).......................	Resolved

Issue	Status
(27) ASME Code Cases N-433 and N-451 (5.2.1.2)	Resolved (deleted)
(28) TMI-2 Action items related to safety/relief valves (5.2.2)	Under review
(29) Inservice inspection (ISI) of reactor coolant pressure boundary (5.2.4)	Under review
(30) Cleaning of stainless steel components (5.3.1)	Under review
(31) Residual heat removal design compliance with BTP RSB 5-1 (5.4.7)	Awaiting information
(32) Preservice inspection (PSI) (5.2.4)	Under review
(33) TMI-2 Action items related to emergency core cooling systems (5.4.6)	Under review
(34) Engineered safety feature-materials (6.1.1)	Under review
(35) Containment systems (6.2)	Under review
(36) Containment leak testing (6.2.6)	Under review
(37) ADS issues (6.3.3)	Under review
(38) Control room habitability (6.4)	Under review
(39) Atmosphere cleanup systems (6.5)	Under review
(40) ISI for class 2 and 3 piping (6.6)	Under review
(41) Main steam isolation valve leakage control (6.7)	Under review
(42) Instrumentation and controls (7)	Under review
(43) Electrical power systems (8)	Under review
(44) Seismic classification of spent fuel pool liner (9.1.2)	Under review
(45) Spent fuel pool level and leakage monitoring (9.1.2)	Under review

Issue	Status
(46) Spent fuel pool cooling and clean-up system (FPC) design issues. (9.1.3)	Under review
(47) Seismic classification of the new fuel stand (9.1.4)	Awaiting information
(48) Overhead heavy load handling system (OHLHS) SSAR clarification items (9.1.5)	Under review
(49) OHLHS compliance with NUREG-0612 (9.1.5)	Under review
(50) Addition of RG 1.72 requirements to interface for ultimate heat sink (9.2.5)	Awaiting information
(51) Makeup water condensate system flooding analysis (9.2.8)	Under review
(52) Makeup water system (purified) design concern (9.2.10)	Under review
(53) Reactor building cooling water system heat removal capacity and performance (9.2.11)	Under review
(54) Heating ventilation and air conditioning (HVAC) normal cooling water system safety related boundary, number of chillers, pumps (9.2.12)	Under review
(55) HVAC emergency cooling water system pump capacity, chemical feed tank design, P&ID clarification (9.2.13)	Under review

Issue	Status
(56) SSAR clarification for turbine building cooling water system (9.2.14)	Under review
(57) Reactor service water performance characteristics, design and location of pump (9.2.15)	Under review
(58) Turbine service water system design details (9.2.16)	Under review
(59) Standby liquid control system compliance with 10 CFR 50.62 (9.3.5)	Awaiting information
(60) Control building HVAC design details (9.4.1.1)	Awaiting information
(61) Essential electrical and reactor building cooling water equipment HVAC design details, compliance with RG for component and filtration specifications (9.4.1.2)	Awaiting information
(62) Engineered safety feature ventilation system design detail (9.4.5)	Awaiting information
(63) Essential equipment design detail (9.4.5.2)	Awaiting information
(64) Essential diesel generator HVAC compliance with GDC 4, design details (9.4.5.5)	Awaiting information
(65) Containment purge supply/exhaust system design detail (9.4.5.6)	Awaiting information

Issue | Status

(66) Nitrogen accumulator and main steamline seismic classification (10.3)....................Under review

(67) Radiation monitoring in turbine gland sealing system (10.4.3)............................Under review

(68) Circulating water system flood protection (10.4.5)................Under review

(69) Condensate and feedwater system power source (10.4.7).............Under review

(70) Condensate storage tank local level alarm (11.2.2)....................Under review

(71) Monitoring service building ventilation exhaust (11.3.1).............Under review

(72) Lack of charcoal adsorbers and filters (11.3.1)....................Under review

(73) Containment secondary exhaust monitoring (11.3.1).....................Under review

(74) Wet waste solidification process details (11.4.1).................Under review

(75) High integrity container information (11.4.2)....................Under review

(76) Incinerator fire protection features, solid waste shipments (11.4.2)................................Under review.

(77) Release points (11.5.1)..................Under review

(78) Service building exhaust monitoring, turbine gland sealing system monitoring and procedures, SSAR Table 11.5-5, RG 1.97 requirements (11.5.2)............Under review

Issue	Status
(79) Radiation protection (12)................	Under review
(80) Conduct of operations (13)...............	Under review
(81) Initial test program (14)................	Under review
(82) Transient and accident analysis (15)...........................	Under review
(83) Technical specifications (16).....................................	Awaiting information
(84) Control room design review (18)...........................	Under review
(85) Severe accident design considerations (19)......................	Under review
(86) TMI Action Plan issues..................	Under review

1.9 Confirmatory Issues

At this point in the review, the following issues have essentially been resolved to the staff's satisfaction. For these issues, GE has already provided or has committed to provide the confirmatory information in the near future. If the staff's review of the information that is to be submitted does not confirm its preliminary conclusions about and issue, that issue will be treated as open, and the staff will report its resolution in a supplement to this report. Each of these issues will need to be resolved prior to the issuance of the FDA.

(1) Fuel licensing acceptance criteria (4.2)
(2) BWR stability (4.4)
(3) Pressure-temperature limits (5.3.2)
(4) Neutron fluence (5.3.2)
(5) Reactor water cleanup system temperature capability (5.4.8)

(6) HVAC normal cooling water safety-related portions (9.2.12)

1.10 Interface Information

The ABWR is an essentially complete plant design encompassing the nuclear island, turbine island, and radwaste facility. The design scope is presently discussed in SSAR Section 1.2, and will be reflected in DSER Section 1.2 in a future supplement. For those portions of the plant for which GE does not seek certification, Section 1.9 of the SSAR defines and lists the interface requirements imposed on an individual applicant who references the design so as to provide site and design compatible features that will ensure the functional performance, and safe operation of the ABWR systems. Such interfaces include details of the ultimate heat sink. Other interfaces are site/plant specific and require confirmation that interfaces designed by the utility applicant conform to the design basis developed by GE, such as the protection of the ultimate heat sink. The final category of interfaces include procedural requirements which must be addressed by the utility applicant to satisfy TMI related requirements such as inplant radiation monitoring.

A summary of the interface requirements for the sections included in this DSER are as follows. They include interfaces listed by GE in SSAR Section 1.9 and those identified by the staff in its review.

(1) Hydrologic features description (2.4.1)
(2) Potential dam failures (2.4.4)
(3) Ice flooding or blockage (2.4.7)
(4) Hydraulic design of canals and reservoirs (2.4.8)

(5) Cooling water supply (2.4.11)

(6) Surface water dispersion of accidental radiation effluent release (2.4.13)

(7) Technical specifications and emergency operation procedures addressing flood protection and shutdown water supply (2.4.14)

(8) Site-specific geology, seismology, geotechnical engineering interfaces (2.5)

(9) Site-specific design wind (3.3.1)

(10) Site-specific design basis tornado (3.3.2)

(11) Design for water level (flood) (3.4.2)

(12) Turbine missile maintenance/inspection (3.5.1.3)

(13) Leak-before-break plant-specific analysis (3.6.3)

(14) Site-specific design response spectra (3.7.1)

(15) Plant-specific seismic system and subsystem analysis (3.7.2)

(16) Seismic instrumentation (3.7.3)

(17) Structural integrity test program (3.8.1.2)

(18) Containment structural details (3.8.4)

(19) Soil parameters, foundation and structure settlement (3.8.5)

(20) Loose Parts Monitoring System (4.4)

(21) Plant Specific ISI, PSI (5.2.4.4)

(22) Drywell leak rate calculational procedures (5.2.5)

(23) Reactor vessel fracture toughness/surveillance data (5.3.1)

(24) Plant-specific pressure-temperature information (5.3.2)

(25) Steam isolation valve testing (5.4.6)

(26) Plant-specific protective coating data (6.1.2)

(27) Plant-specific control room habitability information (6.4)

(28) New fuel storage rack analysis and procedures (9.1.1)

(29) Design of new fuel storage racks (9.1.1)

(30) Spent fuel storage rack analyses (9.1.2)

(31) Engineered safety feature ventilation system design (9.4.5)

(32) Essential electrical equipment HVAC monitoring and electrical controls (9.4.5.4)

(33) Assurance of proper function of standby gas treatment system (9.4.5.5)

(34) Plant specific leak before break analysis (10.3)

(35) Turbine gland sealing system switchover to auxiliary steam upon detection of radiation (10.4.3)

(36) Access control (13.6.3.5)

(37) Control room evacuation analysis (13.6.3.5)

3 DESIGN CRITERIA FOR STRUCTURES, SYSTEMS, AND COMPONENTS

3.4 WATER LEVEL (FLOOD) DESIGN

3.4.1 FLOOD PROTECTION

Flood design was reviewed in accordance with Standard Review Plans (SRP) 3.4.1, and 3.4.2 (NUREG-800). To ensure conformance with the requirements of General Design Criteria (GDC) 2 of 10 CFR Part 50 with respect to protection against flooding, the review addressed the overall plant flood protection design, including safety-related structures, systems, and components (SSC) whose failure as a result of flooding could prevent safe shutdown of the plant or result in uncontrolled release of significant radioactivity.

In this section the applicant 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, control and radwaste buildings were identified as seismic Category I structures. The safety-related systems and components that must be protected against flooding are identified in Table 3.4-1 of the ABWR Standard Safety Analysis Report (SSAR). The flood levels and conditions are described in Table 2.0-1 of the ABWR SSAR. Maximum ground water and flood levels are assumed to be 2 feet and 1 foot below grade, respectively. Dynamic force due to flood is not considered, since the design flood elevation is assumed to be below the plant finish grade. Only the lateral hydrostatic pressure caused by the design flood water level, ground water, and soil pressures were considered in the analyses of the seismic Category I structures. Exterior or access openings and penetrations that are below the design flood level are identified in Table 6.2-9 of the SSAR for the ABWR.

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 which have a required wall thickness of not less than two feet for portions of the structures below the flood level; waterproof coating 8 cm (3 in) above the plant ground grade level of external surface; water stops in all construction joints up to 8 cm (3 in) above the plant ground grade level; and watertight doors and equipment hatches installed below design flood level. The ABWR Safety Analysis Report did not address the effects of standing water on the roofs of safety related buildings and the means to limit the amount of such water as required by the guidelines of RG 1.102. The ability of these structures to withstand the effects of standing water remains an open issue. Based on the review of the plant layout drawings in Section 1.2 of ABWR Safety Analysis Report, the staff concluded that upon resolution of the above identified open issue all the safety-related systems and components located inside seismic Category I structures (which have the harden protection approach described above) are protected from ground water seepage and external flood, and meets the requirements of SRP Sections 3.4.1 and 3.4.2.

Compartment flooding from postulated component or system failures are analyzed separately for reactor, control, radwaste, and service buildings. Single failure of an active component is considered for compartment flooding. Analysis of rupture of moderate-energy piping larger than one-inch diameter are performed 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 No. 410, #1973, "Flow of Fluids Through Valves, Fittings, and Pipe." Conservatively, the effect of drain sump pump operation was not considered. High energy line breaks in the main steam line tunnel are excluded from evaluation, since this area is instrumented for detection of leaks before a line break. However, even in the event of feedwater line break, water will be contained in the seismic Category I structure of the main steam line tunnel and be allowed to drain to the high conductivity water sumps. The applicant described the operator response period and type of flooding source. The staff as discussed in Section 3.6.3 of this report, believes that a leak-before-break analysis should use plant-specific data such as piping geometry, materials, fabrication procedures, and pipe support locations. Therefore, the staff will evaluate the acceptability of the above exclusion on a plant-specific basis.

Analysis of the worst flooding due to pipe and tank failures and their consequences are performed on a floor-by-floor basis within the reactor building to demonstrate the safe shutdown of the reactor. Prevention of flooding to the other divisions is prevented by watertight doors or sealed hatches. Flooding that may occur from fire fighting or from limited inventory sources is restrained by raised sills and elevation differences. Leakage of water from large circulating water lines can be controlled by system isolation.

The evaluation of the control building flooding include events which may result from failure of the service water or fire fighting systems. Propagation of the water to the control areas can be prevented by diverting water to non-critical areas, elevation differences, raised sills, system isolation; and may extend to the reduction of plant load or shutdown.

The radwaste building does not contain safety-related equipment and is isolated from the other plant structures except through a pipe tunnel. In case of a failure of pipe or tank, the building substructure serves as a large sump; therefore, safe shutdown of the plant cannot be jeopardized.

The Service Building is a non-seismic concrete structure, does not house any safety-related SSC, and provides access tunnels to the Control Building. The access tunnels are located below plant grade, and are water tight to prevent seepage into the tunnels. The service building has floor drains and two sumps for high conductivity water and hot shower drains, to collect and transfer the liquid waste or flooding from a pipe rupture.

The ABWR SSAR did not include flood analysis for any structures outside the scope of the generic design housing systems or components performing a safety function, such as the ultimate heat sink pump house. The identified interface requirements, essentially identifying only the normal groundwater and flood levels, are insufficient to insure adequate flood protection design for these structures. Interface requirements that will insure the ability of the plant specific application to meet the flood protection requirements as described in SRP Section 3.4.1 (that is the requirements of GDC 2 and the guidance of RG 1.102) need to be provided. This is an open issue.

The staff concluded that based on the criteria, assumptions, analyses presented by the applicant that pending resolution of the above identified open issues, adequate protection against floods and the effects of floods is provided, and the design meets the requirements of SRP Section 3.4.1 and GDC 2.

3.9 Mechanical Systems and Components

3.9.6 Inservice Testing of Pumps and Valves

The staff review under SRP Section 3.9.6 is concerned with the inservice testing of certain safety-related pumps and valves typically designated as American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) Class 1, 2, or 3. In Section 3.9.3 of this report, the staff discusses the design of safety-related pumps and valves in the ABWR nuclear island. 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, the applicant will periodically test the performance and measure performance parameters of all safety-related pumps and valves. The tests and measurements will be performed in accordance with Section XI of the ASME Code as required by 10 CFR 50.55(g). The tests will ensure the operational readiness of these pumps and valves. Periodic measurements of various parameters will be compared to baseline measurements to detect long-term degradation of the pump or valve performance. However, the staff has determined that the Section XI requirements must be supplemented to obtain the level of operability assurance desired for evolutionary light water reactor designs. SECY-90-016, "Evolutionary Light Water Reactor Certification Issues and Their Relationship to Current Regulatory Requirements," dated January 12, 1990, included a discussion of this issue. Therefore, the staff requested commitments to design and test the applicable components to meet these additional guidelines. The staff's evaluation of the applicant's responses to the Section XI issues, including these additional guidelines, follows.

In Section 3.9.6 of the SSAR, the applicant stated that inservice testing of safety-related pumps and valves will be performed in accordance with the requirements of Section XI, Subsection IWP and IWV of the ASME Code. The staff regards this commitment to be a satisfactory response to Question 210.47 regarding the staff's position that non-Code class safety-related pumps and valves be tested in accordance with Section XI. In Table 1.8-21 of the SSAR, the applicant also indicated that the applicable code of record for the ABWR is the ASME Code, 1989 edition. It should be noted that Part 6 of Operations and Maintenance (O&M), "Inservice Testing of Pumps" and Part 10, "Inservice Testing of Valves," were referenced in Section XI, ASME Code, 1989 Edition. The applicant also stated that code testing flexibility in the ASME/ANSI O&M Part 6 and Part 10 produced no need for relief requests. The staff finds this commitment acceptable.

In Table 3.9-8 of the SSAR, the applicant listed the inservice testing parameters and frequencies for safety-related pumps and valves. The staff has not completed its review of Table 3.9-8 to verify that system and component designs in the ABWR will accommodate inservice testing as described in the table. The results of its evaluation will be addressed in a future revision to this report.

The applicant further stated that details of the inservice testing program, including test schedules and frequencies, will be reported in the inservice inspection and testing plan which will be provided by the applicant referencing the ABWR design. The plan will include baseline preservice testing to support the periodic inservice testing of the components. The primary elements of this plan as described by the applicant are discussed below.

3.9.6.1 Inservice Testing of Safety-Related Pumps

In response to the staff's concern regarding the adequacy of mini-flow systems for safety-related pumps, the applicant committed that the ABWR 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. In addition, the sizing of each minimum recirculation flow path is evaluated to ensure that its use under all analyzed conditions will not result in degradation of the pump. The flow rate through minimum recirculation flow paths can also be periodically measured to verify that flow is in accordance with the design specification. In response to another staff request, the applicant also committed to 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 that these commitments as described have fully addressed the staff's concerns and conform to the applicable staff guidelines and are, therefore, acceptable.

In response to the staff's request to provide a commitment to periodically disassemble and inspect all safety-related pumps, the applicant stated that the safety-related pumps can be disassembled for evaluation when the ASME Code Section XI testing results in a deviation which falls within the required action range. The response did not fully address the staff's concern. In order to provide assurance of the reliability and operational readiness of safety-related pumps, it is important to establish a maintenance program that will include periodic disassembly and inspection in addition to the periodic inservice testing. The staff considers disassembly and inspection to be a maintenance procedure that is not equivalent to inservice testing by fluid flow. The staff also believes that periodic disassembly and visual inspection should be performed to assure the effectiveness of the periodic inservice testing, to detect degradation not revealed by fluid flow testing or through the use of advanced nonintrusive techniques, and to determine the need for component refurbishment. The frequency of inspection and the extent of disassembly may vary depending upon the service condition of the pump. The applicant's commitment to disassemble and inspect the safety-related pump when the ASME Code Section XI testing results in a deviation which falls within the "required action range" is considered as a corrective action rather than a maintenance procedure. Therefore, the applicant should revise the response to address the staff's concern. The staff requires, as a minimum, a commitment to develop a program that will establish the frequency and the extent of disassembly and inspection of all safety-related pumps, including the basis for the frequency and the extent of each disassembly.

3.9.6.2 Inservice Testing of Safety-Related Valves

3.9.6.2.1 Check Valves

In response to the staff's concern regarding the full-flow testing of check valves, the applicant stated that all ABWR safety-related piping systems incorporate provisions for testing to demonstrate the operability of the check valves under design conditions. In response to another staff request, the applicant also committed that inservice testing will incorporate the use of advanced nonintrusive techniques to periodically assess degradation and the performance characteristics of the check valves. These commitments are in accordance with the staff's position and, therefore, are acceptable.

The staff has requested that the applicant make a commitment to periodically disassemble and inspect safety-related check valves in addition to conducting the inservice testing. The periodic disassembly and inspection is to determine if there are any indications of unacceptable wear, corrosion, or other forms of degradation not revealed by fluid flow testing or through the use of advanced nonintrusive techniques. The frequency of inspection and the extent of disassembly may vary depending upon the service conditions. In response to this request, the applicant stated that the ASME Code Section XI tests will be performed, and check valves that fail to exhibit the required performance can be disassembled for evaluation. For similar reasons to those discussed in Section 3.9.6.1 of this report, the applicant's response did not fully address the staff's concerns. The applicant's proposal to disassemble and inspect a check valve when it fails to perform as required is in the nature of a corrective action rather than a maintenance procedure. The applicant should revise the response to address this concern. The staff requires, as a minimum, a commitment to develop a program that will establish the frequency and the extent of disassembly and inspection of all safety-related check valves including the basis for the frequency and the extent of each disassembly.

3.9.6.2.2 Motor Operated Valves

In response to the staff's concern regarding the design of motor operated valves (MOVs), the applicant stated that the MOV equipment specifications require the incorporation of the results of either in situ or prototype testing with full flow and pressure and/or full differential pressure to verify the proper sizing and correct switch settings of the valves. In Section 3.9.7 of the SSAR, an interface commitment is made that the applicant referencing the ABWR design will provide a study to determine the optimal frequency for valve stroking during inservice testing so that unnecessary testing and damage is not done to the valve as a result of the testing. Moreover, the concerns and issues identified in Generic Letter 89-10 for MOVs will be addressed before plant startup. The method of assessing the loads, the method of sizing the actuator, and the setting of torque and limit switches will be specifically addressed.

Furthermore, the inservice testing of MOVs will rely on diagnostic techniques that are consistent with the state of the art and that will permit an assessment of the performance of the valve under actual loading conditions. MOVs that fail the acceptance criteria and are "declared inoperable" for stroke tests and leakage rate can be disassembled for evaluation.

With the following three exceptions, the above commitments are acceptable. The first issue relates to the use of prototype testing of MOVs as an alternative to in situ testing. The staff has discussed its specific concerns and guidelines to justify prototype testing, as an alternative, in Generic Letter 89-10, Supplement 1, Questions 22 and 24 through 28. The applicant should revise Section 3.9.6.2.2 of the SSAR to reflect this guidance. Another issue relates to the MOV disassembly and inspection. For similar reasons to those discussed in the preceding sections, the staff determined that the applicant's response regarding MOV disassembly did not fully address the staff's concern. The applicant's proposed disassembly and inspection action is in the nature of a corrective action for an MOV that fails its acceptance criteria rather than a maintenance procedure. The applicant should revise its response to address the staff's concern. The staff requires, as a minimum, a commitment to develop a program that will establish the frequency and the extent of disassembly and inspection of all safety-related MOVs, including the basis for the frequency and the extent of each disassembly.

Finally, as a clarification of the above interface commitments, the applicant should ensure that the periodic testing is conducted under adequate differential pressure and flow conditions that allow a justifiable demonstration of continuing MOV capability for design-basis conditions, including recovery from inadvertent valve positioning.

3.9.6.2.3 Isolation Valve Leak Tests

In response to the staff's concern regarding the leak-tight integrity of isolation valves, the applicant committed to verify the leak-tight integrity for each valve relied upon to provide a leak-tight function. These valves include (1) pressure isolation valves - valves that provide isolation of pressure differential from one part of a system to another or between systems, (2) temperature isolation valves - 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 - 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 above scope of isolation valves as identified by GE may not be complete and needs to be clarified. GDC 54 requires that all piping systems penetrating containment be provided with isolation capabilities and designed with a capability to periodically test the operability of the isolation valves and to determine if valve leakage is within acceptable limits. This requirement applies to primary and secondary systems penetrating containment. The staff requires that the applicant address compliance with GDC 54 with respect to containment isolation valve design and commit to leak test these valves to appropriate limits.

In Section 3.9.6.2.3 of the SSAR, the applicant stated that leakage rate of valves other than containment isolation valves will be tested in accordance with the ASME Code Section XI. For the primary containment isolation valves, the applicant committed to perform required Type C leak-rate testing in accordance with 10 CFR Part 50 (Appendix J) and approved exemptions. (This commitment will be reflected in the ABWR Technical Specifications.) Appendix J, Type C, leak-rate testing adequately determines the containment leak-tightness provided by these valves. However, the 10 CFR Part 50 (Appendix J), leak-rate testing does not require that individual valve leakage limits be defined, nor is corrective action required based on individual valve leakage rates. Section XI is a component test code to monitor individual component condition and degradation to access operational readiness. Therefore, for these valves, the applicant is requested to commit to comply with the analysis of leakage rates and corrective action requirements of O&M Part 10, Paragraph 4.2.2.3, referenced in Subsection IWV of the ASME Code Section XI, 1989 edition.

Several safety systems connected to the reactor coolant pressure boundary have design pressures below the rated reactor coolant system (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 are placed in service to form the interface between the high pressure RCS and the low-pressure system. A list of RCS pressure isolation valves is included in Table 3.9-9 of the SSAR. The staff has not completed its review of this table.

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. The staff has requested a commitment from the applicant to perform periodic leak testing of all pressure isolation valves in accordance with the applicable sections of the Technical Specifications for recently licensed BWR/6 plants. In response to the staff's request, the applicant indicated that the testing requirement would be referenced in the ABWR Technical Specifications. The applicant also stated that the final proposed ABWR Technical Specifications to be considered and approved under the certification program will reflect the relevant findings of the BWR Standardized Technical Specifications (STS) Program. GE is currently reviewing the BWR STS issued by the NRC staff and will be submitting for approval a revised version to reflect the ABWR design. Until the ABWR Technical Specifications are approved, the staff considers the surveillance requirement of the ABWR pressure isolation valves to be an open item. The result of the staff's evaluation of Table 3.9-9 and surveillance requirement (leak testing) will be addressed in a supplement to this report.

Subsequent to resolution of the issues discussed above, the staff concludes that on the basis of its review of Section 3.9.6 of the ABWR SSAR that the applicant's commitment to a pump and valve inservice testing program is acceptable and meets the requirements of GDC 37, 40, 43, 46, and 54; and 10 CFR 50.55(g). This conclusion is based on the applicant's commitments to provide a test program to ensure that safety-related pumps and valves will be in the state of operational readiness to perform necessary safety functions throughout the life of the plant. This program will include baseline preservice testing and periodic inservice testing of the components in the operational state. The applicant has also committed to include all safety-related ASME Code, Class 1, 2, and 3 pumps and valves and to include those pumps and valves that are not Class 1, 2, and 3, but are considered to be safety related.

9 AUXILIARY SYSTEMS

Introduction

For some systems listed in the Standard Review Plan, the functions in the ABWR design are performed by one or more different systems. Such systems are cross referenced in this section. For example, the functions of the closed cooling water system are also performed by the reactor building cooling water system, the heating, ventilation, and air conditioning (HVAC) normal cooling water system, the HVAC emergency cooling water system, and the turbine building cooling water system.

9.1 Fuel Storage and Handling

9.1.1 New Fuel Storage

The new fuel storage capability was reviewed in accordance with the guidelines of SRP Section 9.1.1, "New Fuel Storage."

New fuel storage is provided in the new fuel storage vault located in the reactor building. The vault contains storage racks for up to 40 percent of one full core fuel load. New fuel is normally stored dry, however, the storage racks can be used in either a wet or dry mode. The storage racks, vault, and the reactor building which houses the facilities are designed to seismic Category I criteria. The reactor building is also designed to provide protection against flooding and tornado missiles. Therefore, the design satisfies the requirements of General Design Criteria (GDC) 2
and the guidelines of Regulatory Guide (RG) 1.29, Position C.1.

The new fuel storage vault is not located in the vicinity of any moderate- or high-energy lines, or rotating machinery. Separation from such potential missile sources protects the new fuel from internally generated missiles and the effects of pipe breaks.

The design does not include shared equipment, structures, or components important to safety, therefore, the GDC 5 requirements are met.

The facility is designed to store unirradiated, low emission fuel assemblies. Accidental damage to the fuel would release relatively minor amounts of radioactivity. They would be accommodated by the Standby Gas Treatment System (SGTS) which would prevent any possible release of radioactivity from a fuel handling accident. Thus, the requirements of GDC 61, "Fuel Storage and Handling and Radioactivity Control" are satisfied.

A fuel array of loaded new fuel racks is designed to assure that K_{eff} does not exceed 0.95 under all normal and abnormal conditions. The new fuel storage racks, according to GE, are purchased equipment.

Interface Requirement: The purchase specification for new fuel storage racks must require the vendor to provide the following analysis.

- Design details to prevent inadvertent placement of a fuel assembly in other than prescribed locations.

- Confirmatory dynamic and impact analyses. The input excitation for these analyses would utilize the horizontal and vertical response spectra provided in Figures 9.1-15 and 9.1-16 of the FSAR.

- The design of the new fuel storage racks will be such that the K_{eff} will not exceed 0.98 with fuel of the highest anticipated reactivity in place assuming optimum moderator conditions (foam, small droplets, spray, or fogging) as described in SRP Section 9.1.1.

Based on the review of the SSAR and the GE responses to the request for additional information, dated September 28, 1990, it is concluded that the new fuel storage facility meets the requirements of GDCs 2 and 61 as they relate to the new fuel protection against natural phenomena, missiles, radiation protection and the guidelines of RG 1.29, Position C.1 relating to the seismic classification. The verification of conformance to the requirements of GDC 62, regarding the prevention of criticality, requires the information identified above.

9.1.2 Spent Fuel Storage

The spent fuel storage capability of the ABWR was reviewed in accordance with the guidelines of SRP Section 9.1.2, "Spent Fuel Storage."

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 contains space for the storage of 270 percent of one full core load. This capacity is in accordance with guidance provided in ANS 57.2 for the capacity of a single unit facility.

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 pools. However, information on the seismic classification of the spent fuel pool liner is not contained in the SSAR. This is considered an open issue.

The reactor building is also designed to provide protection against flooding and tornado missiles. Therefore, with the exception of the open issue, the requirements of GDC 2 and the guidelines of RG 1.13, "Spent Fuel Storage Facility Design Basis," Position C.3, RG 1.29, "Seismic Design Classification, "Positions C.1 and C.2, and RG 1.117, "Tornado Design Classification," Positions C.1 through C.3, are satisfied for the design of the spent fuel storage facility.

The spent fuel pool is not located in the vicinity of any high-energy lines or rotating machinery and thus is protected from internally generated missiles and the effects of pipe breaks by physical separation. The reactor building provides protection against tornado missiles. Therefore, GDC 4 and RG 1.117, "Tornado Design Classification," Positions C.1 through C.3 are satisfied for the design of the spent fuel facility.

The design does not include shared equipment, structures, or components important to safety, therefore, the GDC 5 requirements are met.

The fuel storage racks are to be designed to meet design requirements specifying that the fuel assemblies will be stored in an array which 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 design requirements are specified, but the details of the design are not provided.

<u>Interface Requirement:</u> The detailed design of the spent fuel storage racks are not provided because, according to GE, they are purchased equipment. Therefore, the purchase specification must require the vendor to provide the following analyses:

- 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 from a height of 6 feet.

The design of the storage pool includes the provision of radiation monitoring systems described in SSAR Section 11.5, which were determined to satisfy in part the requirements of GDC 63, "Monitoring Fuel and Waste Systems." The SSAR and the GE response to the request for additional information (Response 430.191) do not include sufficient information to conclude that there are acceptable monitoring systems for pool water level and excessive pool liner leakage. This is considered an open issue.

Based on the review of the SSAR and GE responses to a request for additional information, dated September 28, 1990, it is concluded that the spent fuel storage design meets the requirements of GDC 4 and the applicable portions of RGs 1.13, 1.115, and 1.117 pertaining to environmental and missile protection. Upon successful resolution of the open issues identified above the ABWR spent fuel storage design will meet the requirements of GDCs 2, 61, 62, and

63 regarding seismic classification, fuel storage and criticality and radioactivity control, and monitoring.

9.1.3 Spent Fuel Pool Cooling and Cleanup System

The spent fuel pool cooling and cleanup (FPC) system was reviewed in accordance with SRP Section 9.1.3 (NUREG-0800). An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section as it relates to the cooling portion of the system was performed according to the guidelines provided in the "Review Procedures" portion of the SRP section. Conformance with the acceptance criteria for the cooling portion of the system formed the basis for the evaluation of the FPC system cooling portion with respect to the applicable regulations of 10 CFR Part 50. The staff's evaluation of the cleanup portion of the system is provided separately.

The FPC system is designed to remove decay heat generated by spent fuel assemblies in the pool, and to maintain water quality and clarity and remove corrosion products, fission products, and other impurities from the spent fuel 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 to the point of discharge to the radwaste system. Specifically, the system includes two 50 percent, flow-rated, circulating pumps, two 50 percent capacity heat exchangers, two filter/demineralizers, one post-demineralizer 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 and the overflow is collected in the surge tanks. Two circulating pumps draw water from these tanks and discharge through a common header to two filter/demineralizers located in parallel. System flow then passes through two heat exchangers in parallel, cooled by water from the reactor building cooling water system, and returns to the fuel storage pool. A bypass line is provided around the cleanup portion of the system (the filter/ demineralizers). Each circulating pump can be powered from redundant divisions of Class 1E power supplies.

The FPC system is housed in the reactor building (secondary containment area), a seismic Category I, flood-protected and tornado-missile protected structure. Thus, the system meets Position C.2 of RG 1.13, "Spent Fuel Storage Facility Design Basis," with regard to protection of this support system for the fuel storage facility from tornadic winds and missiles generated by these winds.

Normal makeup water is supplied to the storage pool by the non-safety-related condensate makeup system. Backup to the normal makeup is also available from the non-safety-related suppression pool cleanup (SPCU) system. Additionally, an emergency safety-related makeup (seismic Category I) to the spent fuel pool is 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 designed to seismic Category I standards. By letter dated July 23, 1990, GE stated that the FPC system piping from the pool to the system heat exchangers outlet valves including the outlet valves and other valves on the line as well as the SPCU system piping from the outer isolation valve (i.e., isolating the safety-related portion of the FPC system from the non-safety-related SPCU system) to the safety-related portion of the FPC system, including the FPC system isolation valves on the line, are all designed to seismic Category I standards. The rest of the FPC system is not designed to seismic Category I standards. However, as indicated in ABWR SSAR Table 3.2-1, the non-safety-related portions of the system which could affect any structures, systems, or components important to safety due to their failures during a seismic event are designed to assure 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 non-safety-related portions of the system, respectively. However, it should be noted that the entire cooling portion of the system does not meet Position C.1 of the RG 1.29. As discussed below, this is acceptable provided certain requirements are met. No connections are provided to the spent fuel pool that may 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 water are equipped with siphon breakers, check valves, or other suitable devices to prevent inadvertent pool draining. Interconnected drainage paths are provided behind the liner welds. There are no piping connections which penetrate the fuel pool liner to the fuel storage pool below the bottom of the fuel transfer canal. Leakage through the pool liner is collected in a drain system and transferred to an equipment drain tank. The FPC system is designed so that no single failure or 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 system which uses a safety-related piping segment of the FPC system return line. The FPC system includes back-flow protection in the form of check valves in the return lines that lead to the submerged FPC water return diffusers. Also, as stated above, the system includes redundant safety-related isolation devices to facilitate isolation of the safety-related makeup portion of the system from the non-safety-related SPCU system. In response to the staff's request for additional information relating to backup for the seismic Category I makeup water system, GE stated (June 2, 1989 submittal, response to Question 401.37) that fire hoses will be available to provide backup makeup water to the pool. Also, the SPCU system makeup uses the same FPC system safety-related piping segment which the RHR makeup uses, and so is equally vulnerable to the passive failure mentioned above. Furthermore, such a failure will also affect the decay heat removal capability of both the FPC and the RHR systems (discussed in one of the following paragraphs) since both of these use the same safety-related piping segment of the FPC system return line. On the basis of its review, the staff has determined that the FPC system meets the Position C.8 of RG 1.13.

SRP Section 9.1.3, Acceptance Criteria II.1.a and b, state that GDCs 2 and 4 need not apply to the FPC system design, if 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 are provided. 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 existence of a safety-related ventilation and filtration system, that is, the standby gas treatment system (SGTS), for the fuel pool area, and the provision of a safety-related makeup water system. As concluded in Section 6.5.1 of this report, the design of the SGTS remains as an open item at present, since GE has not demonstrated its compliance with all the applicable positions of RG 1.52 (e.g., redundancy of filters in the SGTS). Since the safety-related makeup for the pool is not totally independent but relies on the safety-related portion of the FPC system, GDCs 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, ABWR SSAR Sections 3.5.1 and 3.6.1 and SSAR Table 3.6-2 include the spent fuel pool cooling for protection against the effects of missiles (including those internally generated) and piping failures. GE's submittal dated June 2, 1989, further states that the major components of the FPC system are located in separately shielded rooms and that barriers, restraints, equipment compartments and the like protect fuel pool cooling components against failure of high energy piping systems. On the basis of this discussion, the staff concludes that the safety-related portion of the FPC system complies with GDCs 2 and 4 subject to GE's confirmation of protection of the portion against the effects of moderate-energy piping failures in the vicinity. The staff also concludes that the design of the major portion of the system as not safety related is acceptable, subject to an acceptable resolution of staff's concerns relating to the safety-related makeup capability for the pool and the SGTS design.

The evaluation of the capability of the FPC system to handle the decay heat load in the fuel pool is based on the guidelines provided in SRP Section 9.1.3, paragraphs II.1.d.(4) and III.1.h. Each of the two FPC system heat exchangers is rated at 6.55 X 10^6 BTU/hr at the design temperature of 125°F Based on an independent calculation of the spent fuel heat load using Branch Technical Position (BTP) ASB 9-2, "Residual Decay Energy for Light Water Reactors for Long Term Cooling," the maximum normal heat load at 150 hours after shutdown is approximately 1.7 x 10^7 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 since the fuel pool capacity (270 percent of a core) is 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 required 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. The RHR system uses a segment of the non-safety-related suction portion piping 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 is also used to supplement the FPC system capabilities to maintain pool water temperature below 150°F under the maximum abnormal heat load, that is, a full core unload.) GE has based the ABWR FPC heat exchanger design on the heat load at 21 days after shutdown, the time at which the fuel transfer canal can be closed. Independent calculations of the heat load at this time confirm the GE calculated value of 1.18 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.

The FPC system design does not appear to accommodate any single active failures. (For example, the failure of a single diesel generator during a loss of offsite power results in the failure of the heat removal capability of one FPC heat exchanger due to the failure of one train of the reactor building cooling water system.) A single heat exchanger has insufficient heat removal capability to maintain the fuel pool temperature at an acceptable level at all times. GE has indicated that for some active single failures, it may be necessary to use the RHR system to limit the temperature of the fuel pool to less than 140°F. However, GE has not explained how the RHR system can be used for the maximum normal heat load removal without violating the technical specification requirements for availability of the system for other purposes in Mode 5 (refueling mode). In addition to this concern, the staff has a more fundamental concern about the gross undersizing of the FPC heat exchangers for the ABWR. The staff considers it inappropriate for an advanced design to rely on the RHR system to supplement the FPC system's normal maximum spent fuel heat load removal capability during certain situations (e.g., single active failure coincident with loss of all offsite power). The staff recognizes that the FPC system for the ABWR is not safety related. The staff believes that this consideration should not inhibit providing two FPC system cooling loops, each independently capable of removing the normal maximum spent fuel heat load to maintain the fuel pool temperature within 140°F at 150 hours after shutdown. The staff requires GE to address the above concern. Until this concern is satisfactorily resolved and the information requested is provided for the staff to make an independent calculation, the staff cannot conclude that the FPC system design complies with GDC 44, "Cooling Water."

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 of components and the accessibility of the components. One of the FPC system pumps and both of the system heat exchangers are normally in operation. The second pump is 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, "Inspection of Cooling Water Systems," and GDC 46, "Testing of Cooling Water Systems" are satisfied by the components in the safety-related portion of the FPC system.

The system design characteristics needed to meet the requirements of GDC 61, "Fuel Storage and Handling and Radioactivity Control," have been described in the preceding paragraphs. 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. Additionally, the FPC system may not be capable of removing sufficient decay heat assuming both a loss of offsite power and a single active failure. Subject to satisfactory resolution of these two issues, the cooling portion of the FPC system meets the requirements of GDC 61.

The fuel pool level and temperature are monitored in the control room, as are area radiation monitors for the fuel storage and handling areas. High and low fuel pool level and high-radiation alarms are also in the control room. Interconnected drainage paths behind the liner welds provide liner leak detection and measurement. Several FPC system parameters are also displayed and recorded in the control room as well as 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 devices, including the alarm provisions, facilitate initiation of appropriate safety actions (e.g., erection of temporary shields to reduce radiation, supply of makeup water to the pool through a remotely operated valve) in a timely manner. Thus, the system instrumentation meets the requirements of GDC 63, "Monitoring Fuel and Waste Storage."

On the basis of this review, the staff concludes the following:

(1) Designing the FPC system for the ABWR as a non-safety-related system except for some portions identified in the preceding paragraphs meets the applicable acceptance criteria of SRP Section 9.1.3.

(2) The cooling portion of the FPC system complies with GDCs 44 and 61 with respect to 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 GDC 4 as it relates to 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 GDC 2 with regard to its protection from the effects of adverse natural phenomena.

(5) The cooling portion of the FPC system complies with GDCs 45, 46, and 63 with respect to inspection, testing and monitoring requirements.

It should be noted, however, that conclusions 1, 2 and 3 (given above) are subject to confirmation, submittal of information, or resolution of the issues listed below:

(1) GE's confirmation of protection of the system's safety-related portions from the effects of moderate-energy piping failures in the vicinity

(2) resolution of 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 design of the cooling portion of the FPC system for the ABWR meets the acceptance criteria of SRP Section 9.1.3, subject to resolution of the two items identified above.

9.1.4 Light Load Handling System (Related to Refueling)

The fuel handling system provides the means of transporting, handling and storing fuel (both new and spent fuel) in the reactor building. The fuel handling system was reviewed in accordance with the guidelines of SRP Section 9.1.4, "Light Load Handling System (Related to Refueling)."

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 5-ton auxiliary hook on the reactor building crane equipped with a suitable grapple. A 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 platform is a gantry crane which is 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 platform also has two 480 kg auxiliary hoists. The hoists can be normally used with appropriate grapples to handle control rods, guide tubes, fuel support pieces, sources and other internals of the core.

The refueling platform is designed to seismic Category I standards and the entire system is housed within the reactor building which is a seismic Category I, flood-protected and tornado-protected structure. However, the new fuel inspection stand is not classified as seismic Category I. It is not entirely clear from the SSAR description that in the event of an earthquake the fall of the new fuel stand would not be detrimental to criticality or radiological safety. This is considered an open issue. The design, therefore, meets in part the requirements of GDC 2, and the guidelines of RG 1.13, Positions C.1 and C.6, and RG 1.29, Positions

C.1 and C.2, relating to the protection of safety-related equipment and spent fuel from the effects of earthquakes.

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 to traverse over the spent fuel pool or the new fuel storage vault while carrying a heavy load.

The design does not include shared equipment and components important to safety, therefore, the GDC 5 requirements are not applicable.

Interface Requirement: As discussed in Section 9.1.2 of this report, the spent fuel storage racks are purchased equipment. The purchase specification must require the vendor to provide a confirmatory analysis which would consider the consequences of dropping of one fuel assembly and its associated handling tool from a height at which it is normally handled above the spent fuel storage racks.

Based on the review of the SSAR and the GE responses to the request for additional information dated September 28, 1990, it is concluded that pending resolution of the open issue identified above, the light load handling system design related to refueling, meets the requirements of GDCs 2, 5, 61, and 62 with respect to protection of safety-related equipment and spent fuel from the effects of earthquakes and the prevention of unacceptable radioactivity releases and criticality accidents.

9.1.5 Overhead Heavy Load Handling System

Inadvertent operations and/or equipment malfunctions due to 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, the 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 system was reviewed in accordance with SRP Section 9.1.5 (NUREG-0800). The acceptance criteria in SRP Section 9.1.5, except as noted below, provided the basis for the staff's evaluation of the system. The acceptance criteria for the OHLHS includes meeting the guidelines of American Nuclear Society (ANS) 57.1 and 57.2. In lieu of these guidelines, the staff used the guidelines contained in the SRP section titled "Review Procedures" and in NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants."

The OHLHS includes equipment necessary for 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 is used for handling heavy loads in the containment and fuel handling area of the reactor building. The single-failure-proof crane is used to handle 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, and some reactor internal pump (RIP) components during the pump servicing. Additionally, the OHLHS includes a single-failure-proof refueling bridge crane and associated equipment which are used to (1) perform fuel shuffling operations, (2) transport spent fuel from the core to the spent fuel pool (SFP) and place it in the SFP storage rack, (3) transport new fuel from its storage area to the reactor core and place it in the core, and (4) handle control rod components and neutron monitor sensors for servicing. Also, the OHLHS includes reactor servicing equipment, upper and lower drywell servicing equipment, and main steam tunnel 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 and the associated heavy load handling operations are performed in the reactor building. 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. The system 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 nonseismic Category I equipment that can adversely impact structures systems and components (SSC) important to safety should they fail during a seismic event have to meet Position C.2 of RG 1.29. ABWR SSAR Table 3.2-1 states that the reactor building and refueling bridge cranes are designed to operating-basis earthquake (OBE) requirements and to hold their positions during a safe-shutdown earth quake (SSE). The table further indicates that the nonseismic Category I system equipment that can adversely impact safety-related SSC during an SSE are designed to ensure their integrity under seismic loading resulting from an SSE. On this basis, the staff finds that the OHLHS for ABWR meets Positions C.1 an 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 GDC 2 with regard to protection of the system from the effects of adverse natural phenomena, subject to GE's clarification of the following issues:

(1) The applicable note (Note x for Table 3.2-1) for the reactor building and refueling bridge cranes should be clarified to state that these cranes will hold their loads under dynamic conditions of the SSE.

(2) The above criterion should be applied for <u>all</u> other applicable (i.e., affecting safe shutdown equipment) nonseismic Category I load handling equipment (e.g., possibly steam tunnel crane, load handling equipment in the control building).

(3) Different terms (such as "refueling bridge crane," "automatic refueling machine," and "spent fuel Handling crane") are used

in the SSAR to represent possibly the same load handling device. Also, it is not clear whether the ABWR has a jib crane and a fuel handling platform different from the refueling platform. Further, SSAR Tables 3.2-1 and 9.1-2 give different seismic classifications for the refueling bridge crane (automatic refueling machine) and the jib crane. All this should be clarified.

(4) 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 location of the subject equipment should be corrected as appropriate and if the location is "any other location," GE should clarify why housing the equipment in a nonseismic Category I structure is acceptable.

(5) SSAR Table 3.2-1 shows that some special service rooms load handling equipment is housed in nonseismic Category I radwaste and turbine buildings. GE should clarify why such locations are acceptable.

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 or 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 bridge crane are single-failure-proof cranes. The staff has the following concerns pertaining to compliance of the OHLHS with 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) ABWR SSAR Section 9.1.4.3 incorrectly states that the spent fuel handling crane's lifting height for the spent fuel cask is limited to 30 feet on the operating floor. This should be corrected to refer to the reactor building crane.

(2) SSAR Amendment No. 7 (response to Question 410.43) gives contradictory statements: "While carrying heavy loads, such as 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." GE should correct this as appropriate.

(3) GE has not identified how safety-related equipment will 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).

On this basis, the staff concludes that the OHLHS design meets the guidelines of RG 1.13 (Positions C.3 and C.5) and GDC 61 with respect to protection of the spent fuel storage facility from the effects of internally generated missiles and safe handling and storage of fuel subject to correction of SSAR Section 9.1.4.3 and correction of response to Question 410.43 as indicated in items 1 and 2 (above). The staff further concludes that safety-related equipment is protected from the effects of internally generated missiles during load handling operations, subject to resolution of concern 3 (above). Therefore, subject to resolution of these concerns, the OHLHS design for the ABWR complies with GDC 4 as it relates to protection of safety-related equipment from internally generated missiles.

As mentioned above, the reactor building and refueling bridge cranes are single-failure-proof cranes. SSAR Section 9.1.5.3 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, "Single-Failure-Proof Cranes for Nuclear Power Plants" or evaluations are made to demonstrate compliance with guidelines of NUREG-0612 Section 5.1, including Sections 5.1.4 and 5.1.5. Also, GE has indicated that the applicable components of the OHLHS will comply with the requirements of the following industry standards: American National Standards Institute (ANSI) Standard N14.6, "Standard for Special Lifting Devices for Shipping Containers Weighing (5 ton) or More for Nuclear Materials;" ANSI B30.9, "Slings;" ANSI B30.10, "Hooks;" ANSI B30.2, "Performance Standards for Overhead Electric Wire Rope Hoists;" ANSI B30.16, "Performance Standards for Air Wire Rope Hoists;" ANSI B30.11, "Overhead and Gantry Crane;" and Crane Manufacturers Association of America (CMAA) Specification 70, "Specification for Electric Overhead and Travelling Cranes." 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 NUREG-0612, Sections 5.1.1(4) and 5.1.1(5) requirements.

GE has provided a general description of the inspection, operation, maintenance, service, and test interface requirements for the OHLHS equipment. Heavy load handling equipment components (e.g., cranes, hoists, refueling platform) will be provided with operating maintenance and test procedures and instruction manuals that will comply with NUREG-0612, Sections 5.1.1(2) and 5.1.1(6). In accordance with the requirements of 10 CFR 50 Appendix B, the OHLHS equipment will be qualification load and performance tested, dimensionally inspected, and will be subjected to nondestructive examination (NDE) before being accepted by the Quality Assurance (QA) group. Lifting equipment components will be subject to appropriate inspection and testing 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 an interface control diagram (ICD) delineating the space around the equipment required for servicing will be developed. It will include pull space for internal parts, and access space for tools, handling equipment and alignment requirements. Further, it 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. Also, safe load paths and routing plans for each heavy load required to be handled will be developed 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. GE requires the referencing applicant to provide documents for these interfaces.

As evident from the preceding two paragraphs, the staff finds the information provided in ABWR SSAR Section 9.1.5 (also Section 9.1.4 which is cross-referenced in Section 9.1.5) lacking in details with respect to OHLHS compliance with applicable guidelines of NUREG-0612. The staff's concerns in this regard are listed below:

(1) GE has not identified all the hoists for the reactor building, refueling bridge, and steam tunnel cranes and any other crane, and the load handling capacity for all the hoists, including the monorail hoist, equipment hatchway hoist, equipment platform/lower drywell RIP hoist.

(2) GE has 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 specific alternative criterion applicable to the chosen heavy load handling device should be identified. If neither of the above criteria is applicable for some heavy load handling devices (because they do not affect required safety-related equipment), it should be so stated with justification for such a conclusion for the applicable devices.

(3) GE has not identified the specific limit and safety devices (e.g., interlocks, 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, GE has not provided 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 essential equipment or causing the release of radioactivity.

(4) GE has not identified 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

has not identified the heavy load handling equipment, their 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.

Subject to resolution of these concerns, the OHLHS for ABWR meets NUREG-0612 guidelines.

On the basis of its review, the staff concludes that the design of the OHLHS for the ABWR is in conformance with the requirements of GDCs 2, 4, and 61 as related to protection against natural phenomena, protection of safety-related equipment from the effects of internal missiles, and safe handling and storage of the fuel, and the guidelines of RG 1.13, Positions C.1, C.3, C.5, and C.6 and RG 1.29, Positions C.1 and C.2, subject to resolution of all the concerns identified in the preceding paragraphs. The staff further concludes that the OHLHS equipment conforms with the guidelines of NUREG-0612 with regard to general design criteria for such equipment subject to resolution of the concerns relating to details of compliance as identified above. However, since a major portion of NUREG-0612 does not apply to ABWR Standard design because it deals mainly with crane operation, operator training, operating and maintenance procedures and physical marking of safe load paths, the staff will review the specifics of compliance and final implementation of NUREG-0612 guidelines on a plant-specific basis. The staff's plant-specific review will include evaluation of (1) implementation of the interfaces identified above including maintenance procedures for detecting and correcting component degradation, (2) built-in safety margins for all heavy load handling equipment, (3) demonstration of the compatibility of the OHLHS design with load handling and movement requirements by comparing the size, shape, and dimensions of the potentially most damaging load, its weight and center of gravity, lifting points, stability and handling speeds with the performance specifications of the OHLHS components, and (4) heavy load handling operations in plant areas outside the ABWR scope and its effect on safety-related equipment.

The OHLHS design for the ABWR meets the acceptance criteria of SRP Sect on 9.1-5 subject to resolution of all the concerns identified above.

9.2 Water Systems

9.2.1 Station Service Water System

See Section 9.2.11.

9.2.2 Reactor Auxiliary Cooling Water System

See Sections 9.2.11 and 9.2.12.

9.2.3 Demineralized Water Makeup System

See Sections 9.2.8, 9.2.9, and 9.2.10.

9.2.4 Potable and Sanitary Water Systems

The design requirements for potable and sanitary water systems were reviewed in accordance with SRP Section 9.2.4, "Potable and Sanitary Water Systems."

The design of the potable and sanitary water systems are site dependent and are therefore not described in detail in the ABWR SSAR.

The ABWR SSAR identifies two interface criteria to be applied to these systems (Section 9.2.17.3). The first of these two criteria is that the potable and sanitary water systems shall be designed with no interconnections with systems having the potential for containing radioactive materials. Where necessary, protection shall be provided through the use of air gaps. Designs meeting these interface requirements will meet the intent of GDC 60.

Specific potable and sanitary water system designs will be reviewed as a part of site specific applications referencing the ABWR design. The ABWR SSAR has provided sufficient design interface requirements to assure that plant specific designs for these systems will meet the requirements of GDC 60.

9.2.5 Ultimate Heat Sink

The design interface requirements for the ultimate heat sink were reviewed in accordance with SRP Section 9.2.5, "Ultimate Heat Sink."

The Ultimate Heat Sink (UHS) has been designated as being outside the scope of the ABWR design. Interface criteria have been provided to allow a plant specific UHS design to be capable of dissipating reactor decay heat and essential cooling loads after a normal reactor shutdown or a shutdown following an accident, including a LOCA. The UHS will be designed to accept the heat loads of the Reactor Service Water System (Section 9.2.15 of this report) which in turn accepts the heat loads of the reactor building cooling water system (Section 9.2.11 of this report).

GE has provided a conceptual design for the UHS. The UHS is to consist of a seismic Category I spray pond from which the Reactor Service Water (RSW) system will receive cooling water. The spray pond is to be excavated below grade and contain adequate water volume to provide 30 days cooling under design basis conditions. Four spray networks, two functioning during normal operation, are to be provided to 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 are located in the spray pond pump structure, each pump being located in its own bay. The pond is also provided with a seismic Category I overflow weir to accommodate normal level fluctuations. Makeup to the spray pond is from the power cycle heat sink through a de'i-cated makeup line.

The structures and components of the UHS are to be designed to seismic Category I requirements and are to be designed to with-stand the effects of floods and tornadoes. The system design should insure that the UHS can perform its safety function given the occurrence of any of the following: 1) the most severe natu-ral 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 criteria will allow a utility applicant to design a UHS that meets the require-ments of GDC 2 and RGs 1.27 and 1.29.

The UHS design is to provide means to insure that safety related portions of the system are protected from spraying, steam impinge-ment, pipe whip, jet forces, missiles, fire and the effects of failure of non-seismic Category I equipment. Thus, the requirements of GDC 4 have been incorporated into the interface requirements.

The plant design is for a single unit site and therefore the requirements of GDC 5 are met.

The UHS shall have the capability to transfer heat loads from safety-related structures and systems during normal and emergency conditions and suitable redundancy so that it will function given a single failure coincident with a loss of offsite power. The UHS will provide cooling capability availability for 30 days. The conceptual design for the UHS is a spray pond. The requirements of RG 1.72 are applicable to the design of a spray pond as an UHS, but were not referenced as interface criteria. The identification of RG 1.72 design criteria as interface requirements remains an open issue. Pending resolution of this issue, these interface criteria insure that GDC 44 can be met.

Based on the above, the interface requirements for the UHS provide adequate guidelines to insure that the plant specific design can meet GDCs 2,4,5 and 44 with respect to protection against natural phenomena, missile and environmental effects. The system is to be designed to allow periodic inspections and tests and will therefore meet GDCs 45 and 46 with respect to inspection and testing requirements for cooling water systems. Pending resolution of the open issue associated with the selection a spray pond as the UHS (the proper spray pond design guidelines of RG 1.72) use of the interface criteria provided in the ABWR SSAR should allow a plant specific applicant to design an acceptable UHS.

9.2.5 Condensate Storage Facility

See Section 9.2.9.

9.2.7 Chilled Water Systems

See Sections 9.2.12 and 9.2.13.

9.2.8 Makeup Water System (Preparation)

The requirements for the Makeup Water System (Preparation) were reviewed in accordance with SRP Section 9.2.3 "Demineralized Water Makeup System." The makeup water system (preparation) supplies a source of water for the makeup water system (Purified) (MUWP) and is entirely beyond the scope of the GE ABWR design. System requirements established for the preparation system include a requirement for two system divisions each of which being capable of producing 200 gpm of demineralized water and each with a storage capacity of 200,000 gallons.

Interface Requirement: The makeup water preparation system is to be located in a building which does not contain any safety related components, systems, or structures. Any failure of the system (including failures that could cause flooding) shall not result in the failure of any safety-related structures, systems or components.

Based on the above, sufficient interface requirements have been provided to insure that a plant specific applicant can design a makeup water system (preparation) that will meet the requirements of RG 1.29 position C.2 and GDC 2 with regards to the ability of the non-safety portions of the system to withstand the effects of earthquakes without affecting safety related systems. The makeup water system (preparation) meets the applicable requirements of SRP Section 9.2.3.

9.2.9 Makeup Water Condensate System

The makeup water condensate (MUWC) system (the condensate storage and transfer system) was reviewed in accordance with SRP Section 9.2.6, "Condensate Storage Facilities." The entire MUWC system is within the scope of the ABWR.

The function of the MUWC system is to provide a source of condensate quality water and a piping distribution system from the source to the components that require this water during normal and emergency operations. A 557,000 gallon condensate storage tank (CST) located outdoors adjacent to the turbine building holds the water for this system. The CST reserves 150,000 gallons of this capacity for decay heat removal by the ac independent reactor core isolation cooling (RCIC) system for 8 hours following a station blackout. The CST is of concrete construction and has a stainless steel lining. The CST also serves as the surge volume for the condensate system. Level sensing instruments and transmitters automatically switch over the high pressure core flooder (HPCF) and RCIC pumps from the preferred CST to the safety-related suppression pool when the CST water level is low. The tank is also a suction source for the control rod drive supply pump (the preferred water source being the condensate treatment system) and the suppression pool cleanup (SPCU) pump which is used for filling the fuel pool makeup line when required. The MUWC system normally provides water via three system transfer pumps for charging, flushing, pump sealing, surveillance testing, room decontamination, and makeup as appropriate, for a number of 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 does not affect the capability of the reactor coolant system pressure boundary to achieve and maintain safe shutdown, or the capability to prevent or mitigate the consequences of accidents which could result in unacceptable offsite radiological exposures. Therefore, compliance with GDC 44, "Cooling Water;" GDC 45, "Inspection of Cooling Water System;" and GDC 46, "Testing of Cooling Water System" identified as acceptance criteria in SRP Section 9.2.6, is not applicable to the system. Also, compliance with GDC 5, "Sharing of Structures, Systems, and Components," identified as an acceptance criterion in SRP Section 9.2.6, is not applicable to the system, since the ABWR design is limited to a single unit. Although it is not safety related, this system is designed to provide adequate pump flow, CST overflow/drainage diversion to the radwaste system, material corrosion resistance, CST water level, control room instrumentation, outdoor piping freezing protection, and to allow testing of air-operated valves.

As stated in the ABWR SSAR Table 3.2-1, "Classification Summary," for the MUWC, RCIC and HPCF systems, certain parts of MUWC system piping, including supports and valves, are designed to seismic Category I and Quality Group B standards and are located in seismic Category I, flood-protected and tornado-missile-protected structures. The safety-related portions include those forming part of the containment boundary as well as system piping portions that interface with the safety-related RCIC and HPCF systems including the isolation/suction valves for the systems from the CST. The non-safety-related portions of the system which could affect any structures, systems, or components important to safety should they fail during a seismic event are designed to assure their integrity under seismic loading resulting from a safe shutdown earthquake. The level instruments 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 switchover of the suppression pool cleanup pumps is manual. On the basis of its review, the staff has determined that the system meets Positions C.1 and C.2 of RG 1.29 for its safety-related and non-safety-related portions.

GE has not provided the analysis for potential flooding resulting from possible failure of the non-safety-related MUWC system including the CST and for how safety-related structures, systems, and components are protected from such flooding. Subject to satisfactory resolution of this concern, the system meets the requirements of GDC 2.

Normal alignment for removal of decay heat is with the condensate storage tank. Water for RCIC operation is taken from either the CST or the suppression pool as described in the Emergency Procedure Guidelines of Appendix 18A of the SSAR. The volume of water in these two sources is sufficient to permit core cooling during station blackout 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."

As discussed above, the MUWC system complies with Positions C.1 and C.2 of RG 1.29, GDC 2 and applicable guidance of RG 1.155 and, therefore, with the applicable acceptance criteria of SRP Section

9.2.6, subject to satisfactory resolution of the concern identified above.

9.2.10 Makeup Water System (Purified)

The makeup water system (purified) (MUWP) was reviewed in accordance with SRP Section 9.2.3, "Demineralized Water Makeup System." That portion of the MUWP system which is inside the reactor building is within the scope of the ABWR. The portions of the system that involve demineralized makeup water preparation, storage, and transport to the reactor building are within the scope of the referencing applicants. The MUWP system, including the purified or demineralized water storage tank, is not safety related, except as noted below.

The function of the MUWP system is to provide a source of demineralized makeup quality water d a piping distribution system from the source to the components that require this water. A storage tank for purified water, which is not part of the GE scope of the ABWR, is the source of demineralized water for this system. The MUWP system normally provides demineralized water via two system transfer pumps at a maximum flow rate of approximately 600 gpm and within a temperature range of 50°F -100°F for flushing, sealing, surveillance testing, area decontamination, sampling and makeup as appropriate. A number of systems are supplied makeup water, which include the makeup water condensate (MUWC) system, reactor building cooling water (RCW) system, turbine building cooling water system, diesel generator cooling water (DGCW) system, liquid radwaste system, standby liquid control (SLC) system, and other plant auxiliary systems. Protection from flooding for safety-related structures, systems and components resulting from failure of the MUWP system is discussed in Section 3.4.1 of this report.

The MUWP system is not safety related except as noted below because it does not affect the capability of the reactor coolant system pressure boundary, capability to achieve and maintain safe shutdown, or the capability to prevent or mitigate the consequences of accidents which could result in unacceptable offsite radiological exposures. The MUWP system line 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 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. GE will provide additional system design features (identified below) which fall within its design scope. The referencing applicant will be required to provide the remaining design features. These features include testing capability for air-operated valves; adequate pump NPSH, purified water storage tank overflow/drainage diversion to the radwaste system; material corrosion resistance; adequate distribution piping, valves, instruments, and controls; control room instrumentation that indicates 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 which will be located outdoors. Table 9.2-2a of the ABWR SSAR presents chemistry requirements for the purified makeup water.

It is not clear whether the non-safety-related portions of the system, which upon their failure during a seismic event can adversely impact structures, systems, or components important to safety, are designed to ensure their integrity under seismic loading resulting from a safe shutdown earthquake

Subject to verification that the design of the MUWP system meets the above requirement, the 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.

Further, it is not clear whether the portions of the MUWP system in buildings other than the reactor building (e.g., turbine building) and transport of the demineralized water to these buildings are within GE's scope or the referencing applicant's scope. The staff requires GE to address all these issues to complete the information on the MUWP system provided in ABWR SSAR Section 9.2.10.

As discussed above, 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, subject to resolution of the concerns identified above relating to the design of safety-related and non-safety-related portions of the system.

9.2.11 Reactor Building Cooling Water System

The reactor building cooling water (RCW) system was reviewed in accordance with SRP Section 9.2.2, "Reactor Auxiliary Cooling Water Systems." The function of the RCW system is to remove heat from plant auxiliaries, some of which are required for safe shutdown and following a LOCA. The RCW system is required to operate at normal power, reactor shutdown, hot standby, both with and without preferred ac power available, 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: residual heat removal (RHR) and fuel pool cooling (FPC) heat exchangers; mechanical seals and motor bearings for RHR and high pressure core flooder (HPCF) pumps; air conditioning units for pump rooms (RHR, HPCF, FPC, and reactor core isolation cooling (RCIC)) and system rooms (standby gas treatment, containment atmospheric monitoring, and flammability control); jacket water coolers and filtered water and lubricating oil coolers for diesel generators; and HVAC emergency cooling water (HECW) system refrigerators. Additionally, the RCW system supplies cooling water to the nonessential reactor internal pump (RIP) motor coolers and motor generator sets, drywell coolers, reactor water cleanup (RWCU) pump coolers, instrument and service air (IA and SA) system coolers, RWCU non-regenerative heat exchangers, control rod drive (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 picks up heat from the plant auxiliaries it serves and rejects the heat through the RCW system heat exchangers to the reactor service water (RSW) system. The RSW system, in turn, rejects the heat to an ultimate heat sink that will be designed by the individual applicants referencing the ABWR. 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 has 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 are the interface requirements for the referencing applicant to design the plant-specific ultimate heat sink system which would be connected to the RSW system. Additionally, GE has specified the inlet and outlet temperatures during applicable modes of operation for components serviced by the RCW system.

The RCW system comprises three mechanically and electrically independent divisions. Each division consists of its own separate piping (including supply and return headers), two pumps, two heat exchangers, valves, and instrumentation. Each division of the RCW system is powered by a different division of the emergency 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 air conditioning unit, and RHR pump motor and seal coolers. Other safety loads and nonessential cooling loads are distributed among the three divisions; two divisions share the loads for systems with redundant components (e.g., HPCF, HECW, SGTS). Each division has one isolable train for nonessential loads.

Each division of the RCW system is equipped with a surge tank which GE 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, provide adequate pressure for pump suction, and allow for changes in system water volume without significant pressure variations. The system is initially filled with demineralized water from the makeup water (purified) 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 the use of high point vents in isolable portions of the system and operational procedures requiring filling and venting of any sections of the system before operation. Nonessential RCW system cooling loads are automatically isolated by applicable valve closure in the event of a LOCA with the exception of system cooling loads to IA and SA system coolers, CRD pump oil coolers, and RWCU pump coolers; these are isolated by the operator, if so desired. Level switches provided for the surge tank facilitate automatic isolation of nonessential cooling loads in the event of significant system leak resulting 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, assures isolation in the event of a single active component failure. However, this will not affect the supply of cooling water to essential cooling loads from other divisions. GE has described the detailed procedures for determining whether the system leakage occurs in the nonessential portion of the system. A falling surge tank level would indicate such leakage. Radiation monitors located downstream of the RCW system pumps and heat exchangers indicate that radiation has leaked into that division. The ABWR provides 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 comprises 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 which isolate the system from its non-safety-related portions) are designed to seismic Category I, Quality Group B or C and 10 CFR Part 50 (Appendix B) requirements. 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 Quality Group B requirements. Non-safety-related portions of the system that can adversely impact safety-related structures, systems or components should they fail during a seismic event are designed to ensure their integrity under seismic loading resulting from an SSE. The safety-related portions are located in seismic Category I, flood-protected and tornado-missile protected structures; GE shall provide confirmatory documentation that this includes also safety-related electric modules and safety-related cables as GE indicated in conversation with the staff in November 1989. On this basis, the staff finds that the design of the RCW system complies with GDC 2 with respect to its 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.

GE stated 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-Category I equipment failure; or a single active component failure in the system. GE has performed a failure analysis of the RCW system and presented the results in the ABWR SSAR 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 reactor service water system under all applicable modes of operation. Also, GE has provided design characteristics for RCW system components (e.g., pump design flow rate; heat exchanger heat removal capacity) to show that the system is capable of transferring the expected heat loads to the reactor service water under all operating conditions.

On the basis of this review, the staff has identified the following concerns:

(1) An RCW system heat exchanger heat removal design capacity may be inadequate based on heat load required to be removed during suppression pool cooling following a LOCA when the post LOCA pool temperature reaches 97°C. A reactor shutdown at 4 hours after a blowdown to the main condenser may be the bounding case. This may require a greater heat removal rate and consequently, a higher design capacity than that currently stated for the heat exchangers (see GE's response to Question 440.73).

(2) The projected heat loads and flow rates for hot standby conditions with a loss of ac power indicate that both heat exchangers and both RCW system pumps in a division are required. This will be the case for shutdown at 4 hours also. The current SSAR should incorporate this requirement.

(3) It is not clear whether the loss of an RCW system division during normal operation will result in plant shutdown or operation at reduced power. (For example, what will be the effect of the loss of Division A which supplies cooling water to 5 out of 10 RIP coolers and 2 out of 3 drywell coolers?)

Subject to the satisfactory resolution of these concerns, the safety-related portions of the RCW system comply with GDC 4 with respect to protection against dynamic effects resulting from postulated piping failures and internally and externally generated missiles and with GDC 44 with respect to the provisions of a system to transfer heat from structures, systems and components important to safety to an ultimate heat sink.

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 loss of offsite power 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. Upon the occurrence of a LOCA, most non-safety-related RCW system loads are automatically isolated as stated above, the second RCW system pump started, and the second heat exchanger in each division is placed in service.

As stated earlier, GE has provided interfacing requirements for an applicant to design the ultimate heat sink in terms of the total heat load, temperature drop, pressure drop, and flow through the RCW system heat exchangers. The RCW system water quality requirements are established the by makeup water system (purified), since this system provides the makeup water for the RCW system. This system is discussed separately 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. This testing capability consists of structural and leak-tightness visual inspection, entire system operability, and system component 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 also incorporates provisions for accessibility to permit inservice inspection as required. On this basis, the staff finds that the system complies with GDCs 45 and 46 with respect to inspection and testing requirements for cooling water systems.

The staff concludes that the design of the RCW system complies with GDCs 45, 46 and 2 with respect to inservice inspection and testing requirements and protection against natural phenomena for its safety-related portions. The staff also concludes that the system design meets the guidelines of Positions C.1 and C.2 of RG 1.29 with respect to seismic requirements for the safety-related and applicable non-safety-related portions of the system. Further, the staff concludes that the system design complies with GDCs 44 and 4 with respect to cooling water requirements and protection against internally and externally generated missiles and dynamic effects resulting from postulated piping failures, subject to satisfactory resolution of all the concerns identified above. The system design meets the applicable acceptance criteria of SRP Section 9.2.2 subject to satisfactory resolution of all the concerns identified above.

9.2.12 HVAC Normal Cooling Water System

The HVAC normal cooling water (HNCW) system was reviewed in accordance with SRP Section 9.2.2. The entire HNCW system is within the scope of the ABWR. With the exception of 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 HNCW system is not safety related.

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 reactor, control, radwaste, and service buildings. The HNCW system is not safety related because it is not required to ensure the RCS pressure boundary capability to achieve and maintain safe shutdown, and the ability to prevent or mitigate offsite radio logical exposures during accidents. Therefore, GDCs 44, 45, and 46, identified as acceptance criteria in SRP Section 9.2.2 for safety-related portions of cooling water systems are not applicable to 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, Quality Group 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.

The rest of the 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 assure their integrity under seismic loadings resulting from a safe-shutdown earthquake. On this basis, the staff finds that 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 finding is subject to GE's confirmation that the safety-related portions include the isolation valves for the secondary containment penetrations. 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, all safety-related systems are protected against flooding that may result in the event of system failure as concluded in Section 3.4.1 of this report. Therefore, the staff finds that the system 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 SRP Section 9.2.2 acceptance criterion.

Although it is not safety related, this system is designed to allow periodic testing and inspection of major components. Appropriate American Society of Heating, Refrigeration and Air Conditioning (ASHRAE), ASME, Tank Equipment Manufacturers Association (TEMA), and Hydraulic Institute standards are used for all tests. The major components of the HNCW system consist of five 25 percent-capacity chillers (one standby) each with an HNCW pump (one standby) and the associated piping, valves, and instrumentation. The system is also provided with a chemical feed tank and vents at high points, the latter to eliminate water-hammer. Cooling water to the chiller-condenser is supplied by the turbine building cooling water (TCW) system.

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. GE has provided the 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 be also operated manually by remote means. The piping and instrumentation diagram (ABWR SSAR Figure 9.2-2a) for the system shows only four chillers and four pumps, whereas ABWR SSAR Section 9.2.12 and Table 9.2-6 indicate five chillers and five pumps. GE should correct this discrepancy. On this basis, the staff concludes that the HNCW system meets the applicable acceptance criteria of SRP Section 9.2.2, subject to correction of the discrepancy mentioned above, and confirmation relating to the seismic design of the system isolation valves for the secondary containment penetrations.

9.2.13 HVAC Emergency Cooling Water System

The HVAC emergency cooling water (HECW) system is a closed cooling water system that was reviewed in accordance with SRP Section 9.2.2. The function of the HECW system is to provide cooling water to the main control room air conditioners, 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. The entire system is within the scope of the GE ABWR design. Specifically, 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. There is no interface requirement for referencing applicants for this system because it interfaces with systems that are also part of the GE scope of the ABWR design (e.g., the RCW and MUWP systems, which are discussed in Sections 9.2.11 and 9.2.10 of this report).

The HECW system comprises two mechanically and electrically independent and completely redundant divisions. Each division, with a surge tank, two 50 percent-capacity chillers (refrigeration units) and two 100 percent-capacity pumps, consists of its own separate piping (including supply and return headers), 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. Divisions A and B supply chilled water to cooling coils in the essential electric equipment rooms A and B and diesel generator Zones A and B, respectively. Both the divisions supply chilled water to their respective coolers in the main control room. The system also has a chemical feed tank to add chemicals to each division to protect the system components from fouling.

The HECW system, as well as the cooling water lines from the RCW system, is designed to seismic Category I, 10 CFR Part 50 (Appendix B), and Quality Group C requirements. Thus, the system meets Position C. 1 of RG 1.29 with regard to seismic classification for safety-related systems. By virtue of its location in seismic Category I, flood-protected and tornado-missile-protected structures, 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 which 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 provides normal makeup water. The tank includes level switches to detect system leakage and to facilitate supply of makeup water to the tank when required. These switches actuate the makeup water supply valves (open or close on low or high tank water level) and provide annunciation of control room alarms (high-high or low-low tank water levels). (See ABWR SSAR Section 7.3.1.1.9.)

The design of the HECW system includes sufficient separation for both mechanical and electrical components of the redundant trains and protection for the system to perform its function under all reactor conditions including 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 ABWR 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 provided for the chillers automatically start the redundant division whenever there is failure of the operating division (e.g., high temperature of the returned cooling water, 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 has also provided 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 RBCW cooling water under all reactor conditions.

On the basis of this review, the staff has identified the following concerns:

(1) ABWR SSAR Table 9.2-9 does not indicate that a single HECW system pump by itself can deliver the required total chilled

water flow rate. Therefore, the statement in ABWR SSAR Section 9.2.13.2 that each division of the HECW system contains two 100 percent capacity pumps has to be corrected.

(2) It is not clear whether the single chemical feed tank provided for the system is safety related. Also, GE has not indicated whether the associated isolation valve and the portion of the piping wherein it is located are safety related. Further, GE has not indicated how the non-safety-related portions of the HECW system (if there are any such portions) is isolated from the rest of the system whenever such isolation is warranted.

(3) It is not clear which division of the HECW system supplies chilled water to the cooling coils in the diesel generator Zone C. Also, ABWR SSAR Figure 9.4-4 indicates three divisions for the HECW system whereas ABWR SSAR Section 9.2.13 refers to only two divisions for the system. The staff considers that the Division C diesel generator support systems should be independent of Divisions A and B.

Subject to resolution of the above concerns, the system complies 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 the applicable requirements for this cooling water system.

As discussed above, the only interfacing requirements for this system are the HECW chiller condenser flow, supply temperature, and heat removal capacity which are directly input to the

GE-designed RCW system. No applicant interface requirements are necessary for the HECW system. The HECW system water quality requirements are established by the makeup water system (purified) since this system provides the water for the HECW system surge tanks.

Both divisions of the HECW system include 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 are provided to indicate vital parameters required in testing and inspections. For example, system chilled water flow rate and temperature can be checked out by readout of locally mounted pressure and temperature gauges at the main control panel. In conversation with the staff on February 26, 1990, GE stated that the system also includes provisions to permit periodic testing of system components as well as the system as a whole. Specifically, GE stated that this testing capability would include structural and leak-tightness visual inspection, entire system operability, and system component operability and performance. On this basis, the staff finds that the system complies with GDCs 45 and 46 with respect to the requirements for inspection and testing of safety-related cooling water systems.

The staff concludes that the design of the ABWR HECW system meets the requirements of GDCs 2, 45, and 46 with respect to its protection against natural phenomena, inservice inspection, and functional testing, contingent upon GE updating the ABWR SSAR reflecting what was stated in a February 26, 1990, conversation with the staff. The HECW system also meets the guidelines of Position C.1 of RG 1.29 with respect to its seismic classification. Further, the staff concludes that the system design complies with GDCs 4 and 44 with respect to protection against internally and externally generated missiles and dynamic effects resulting from postulated piping failures and cooling water requirements, subject to satisfactory resolution of all the concerns identified above. Also, as noted above, the staff requires the revision of the applicable entry in ABWR SSAR Table 3.2-1. The system design meets the applicable acceptance criteria of SRP Section 9.2.2 subject to satisfactory resolution of all the concerns identified above and appropriate revisions of Table 3.2-1 and Section 9.2-13 of the ABWR SSAR.

9.2.14 Turbine Building Cooling Water System

The non-safety related turbine building cooling water (TCW) system was reviewed in accordance with applicable portions of SRP Section 9.2.2, "Reactor Auxiliary Cooling Water Systems."

The TCW system is a non-safety related system designed to provide heat removal capability for various turbine island auxiliary equipment. The TCW is a closed loop system consisting of two 100 percent pumps, two 100 percent heat exchangers, a surge tank and associated piping, valves and instrumentation. The description of the TCW in this section of the ABWR SSAR contains several inconsistencies. The component and system descriptions of Section 9.2.14.2.3, "System Operation" and of Figure 9.2-6a do not agree with the descriptions of Sections 9.2.14.2.1, "General Description" and 9.2.14.2.2, "Component Description," and the responses to RAIs. The discrepancies (number of heat exchangers, pump capacities) should be corrected. Because the descriptions in Sections 9.2.14.2.1 and 2 are more recent versions of the system description, they were analyzed as the TCW system. These discrepancies remain an open item. GE has provided 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 one pump is in operation and the second is in standby. The second pump automatically starts on low pump discharge pressure. There are no connections between the TCW and safety related water systems and it is designed in accordance with Quality Group D standards.

The TCW is located in and near the turbine building away from safety related systems. In response to Questions 430.206 and 430.207, GE stated that failure of any TCW components, including the atmospheric surge tank, would not fail any safety-related equipment. From equipment layout diagrams this statement appears to be true for all equipment shown on the diagrams. The atmospheric surge tank does not appear on these diagrams. Verification that failure of this component will not affect safety-related systems is not possible. This remains an open issue.

Based on the above the TCW meets the requirements of Position C.2 of RG 1.29 pertaining to seismic requirements for non-safety related systems and components and therefore meets the requirements of GDC 2 subject to verification of the items identified above.

Because the TCW system is non-safety related and does not interface with a safety system the remaining requirements (GDCs 4, 44, 45, and 46) of SRP Section 9.2.2 do not apply.

Subject to verification of the issues identified above, the TCW system meets the requirements of SRP Section 9.2.2.

9.2.15 Reactor Service Water

The reactor service water (RSW) system was reviewed in accordance with SRP Section 9.2.1, "Station Service Water System."

The function of the RSW is to provide cooling water for the reactor building cooling water (RCW) system (reviewed in Section 9.2.11 of this report) which in turn supplies cooling water to several essential and non-essential loads. The RSW is required to operate at normal power, reactor shutdown, hot standby, and after a postulated LOCA has occurred. Under each of these conditions the RSW is required to function both with and without preferred ac power available.

The RSW is an open cycle system which provides cooling water to the RCW heat exchangers. No other heat loads are supported by the RSW system. The RSW system picks up heat from the RCW heat exchangers and rejects the heat to the ultimate heat sink which is to be designed by individual applicants referencing the ABWR. The GE scope of the RSW includes all the piping valves, pumps, heat exchangers, instrumentation, and controls from the system intake from the ultimate heat sink to the discharge back to the ultimate heat sink. Although the total heat rate, total flow rate, temperature drop and pressure drop at the RCW heat exchangers for all identified modes of operation were provided for the RCW system, similar parameters for the RSW system (including identification of sufficient net positive suction head at pump suction locations considering low water levels) were not provided. This is an open issue.

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.

GE stated that the RSW system will be able to perform its function during abnormally low or high water levels and that steps are taken to prevent organic fouling that may degrade system performance. Proposed steps include trash racks, biocide (or non-biocide where biocide treatment is not allowed) treatment, and thermal backwash capabilities. Selection of appropriate measures is site dependent and therefore the staff will review those measures on a plant-specific basis.

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 prior to operation.

All portions of the RSW system are designed to seismic Category I Quality Group C requirements. According to Table 3.2-1 of the SSAR, the RSW pumps are located in the structures associated with the ultimate heat sink and therefore all portions of the system are located in seismic Category I, flood and missile protected structures. Based on the above, the design of the RSW system complies with 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.

GE stated 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 perform reactor shutdown in the event of any of the following occurrences:

pipe rupture or equipment failure induced flooding, spraying steam release; pipe whip and jet forces from a postulated nearby high energy line break; missiles from equipment failure; fire; non-Category I equipment failure; or a single active component failure in the system.

However, insufficient detail has been provided to insure that this design criteria can be met. Specifically, location and design features for the RSW pump and associated equipment have not been specified. Subject to resolution of this open issue, the RSW complies with GDC 4 with respect to protection from dynamic effects resulting from postulated piping failures and internally and externally generated missiles.

All three divisions of the RSW system will have a least one RSW pump operating. This configuration ensures immediate availability of the RSW system for plant shutdown in the event of a LOCA. A loss of offsite power 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 second 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 ultimate heat sink.

Protection from adverse environmental conditions, specifically freezing and icing, and potential biofouling concerns are addressed as part of GEs response to Generic Issue 51, USI B-32 and USI B-29. The adequacy of the ABWR design provisions and interface requirements will be discussed in Appendix C to this report addressing these generic issues.

The ability of the RSW system to perform its function cannot be verified because pump design characteristics (flow, pressure, NPSH requirements) were not included in the SSAR (The required heat loads to be removed by the RSW system are identified.) While the design requirements of the RSW system meet the intent of GDC 44, the ability of the RSW system to meet the requirements of GDC 44 remain an open item.

All three divisions of the RSW system are designed to allow periodic inservice inspection of all the system components. This testing capability consists of structural and leak-tightness visual inspection, entire system operability, and system component operability and performance. Testing will be conducted to simulate as closely as possible the entire operational sequence of the RSW system for reactor shutdown and LOCA. The system design also incorporates provisions for accessibility to permit inservice inspection as required. Therefore, the system complies with GDC 45 and GDC 46 with respect to inspection and testing requirements for cooling water systems.

Based on the above, the design of the RSW system complies with GDCs 2, 45, and 46 with respect to inservice inspection and testing requirements and protection against natural phenomena for its safety-related portions. The system design meets the guidelines of Positions C.1 of RG 1.29 with respect to seismic requirements for the safety-related and applicable non-safety-related portions of the system. The system complies with GDC 5 with respect to the sharing of system components between units. Further, the system design complies with GDCs 4 and 44 with respect to cooling water requirements and protection against internally and externally generated missiles and dynamic effects resulting from postulated piping failures, subject to satisfactory resolution of all concerns identified above. The system design meets the applicable acceptance criteria of SRP Section 9.2.1 subject to satisfactory resolution of all the concerns identified above.

9.2.16 Turbine Service Water System

The non-safety related Turbine Service Water (TSW) system was reviewed in accordance with applicable portions of SRP Section 9.2.1 "Station Service Water System."

The TSW system is a non-safety related system designed to transfer heat from the TCW heat exchangers to the turbine plant heat sink. The TSW includes two 100 percent capacity pumps, two 100 percent capacity duplex strainers and associated piping, valves, and instrumentation. GE did not provide system parameters (pump flow requirements, system operating pressure) but did provide a requirement that water supplied to the TCW heat exchangers be at a temperature not to exceed 100°F. (This differs from the requirements of 95°F for service water inlet temperature provided in Section 9.2.14.1.2.) During normal operation one pump is operating and the second is in standby. The standby pump starts automatically if the operating pump trips or pump discharge pressure drops below a preselected limit.

The TSW is located in the intake structure (the power cycle heat sink pump house) and the turbine building. The system does not appear to have any connections with safety related systems, although insufficient detail is provided in the system description and diagrams to verify that no such connections exist. 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. (Due to the site specific nature of the location of some TSW components this requirement may need to be expressed as an interface requirement). Demonstration that GDC 2 can be met or the establishment of an interface requirement for the TSW to meet RG 1.29 Position C.2 remains an open issue.

Because the TCW system is non-safety-related and, pending verification of no connections to safety-related systems, the remaining requirements of SRP Section 9.2.1 (GDCs 4, 44, 45, and 46) are not applicable.

Subject to the resolution of the issues identified above the TSW system meets the requirements of SRP Section 9.2.1.

9.2.17 Interfaces

Section 9.2.17 provides a discussion of the interfaces established for the plant specific designs for the ultimate heat sink, the portable and sanitary water systems, and the makeup water systems (preparation).

These interface requirements have been addressed in the relevant system reviews of Sections 9.2.4, "Portable and Sanitary Water Systems;" 9.2.5, "Ultimate Heat Sink;" and 9.2.8, "Makeup Water System (Preparation)."

9.3 Process Auxiliaries

9.3.1 Compressed Air System

This section will be provided in a supplement to this SER.

9.3.2 Process and Post-Accident Sampling System

This section will be provided in a supplement to this SER.

9.3.3 Equipment and Floor Drainage System

This section will be provided in a supplement to this SER.

9.3.4 Chemical and Volume Control System

This system in not is not applicable to a BWR.

9.3.5 Standby Liquid Control System

The standby liquid control system 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 was performed according to the guidelines provided in the "Review Procedures" portion of the SRP. Conformance with the acceptance criteria formed the basis for the staff's evaluation of the standby liquid control system with respect to the applicable regulations of 10 CFR Part 50.

The standby liquid control (SLC) system is a reactivity control system. Its purpose is to inject sodium pentaborate solution into the reactor coolant to provide an independent means for shutting down the reactor. The SLC system can bring the reactor from rated power to cold shutdown any time during core life should the normal reactivity control system become inoperable. Thus, together with the control rod system, it satisfies the requirements of GDC 26, "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 and a motor-operated valve at each pump suction, piping and controls. All of the SLC system is located within the secondary containment. The maximum temperature at which the solid material would precipitate from solution is 59°F; the room in which the equipment containing the borated solution is kept at a temperature of 50°F to 100°F. An electrical resistance heating system maintains the solution between 75°F and 85°F to prevent precipitation of the sodium pentaborate from solution during storage. Both the high and low levels and temperature of liquid in the tank are alarmed in the control room. The two pumps in parallel trains take suction on the storage tank via separate suction lines and discharge it into the reactor vessel via a common injection line. The liquid is piped into the reactor through the high-pressure core flooder (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. If needed, the system is designed to operate both pumps simultaneously, capable of injecting in excess of 86 gpm (100 gpm) sodium pentaborate solution.

In the ABWR SLCS design, the Squib activated (explosive) injection valves are replaced by motor operated ac valves. On earlier BWR plants, the SLCS piping is not completely isolated from the SLCS storage tank and it is possible for boron to be present in the SLCS piping. Consequently, it was decided to provide leak proof explosive valves in early BWRs 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 fill system to keep boron solution in the storage tank and to prevent it 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. The staff finds this acceptable.

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.

System initiation is accomplished by manual actuation of either of two key-locked switches on the control room panel. Changing either switch status to "run" 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. Should 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 SLC system is located in a compartment within the seismic Category I, flood-protected and tornado-protected secondary containment building outside of the drywell and below the refueling floor. All portions of the SLC system necessary for injection of sodium pentaborate solution into the reactor are seismic Category I, Quality Group B (or Quality Group A if they are part of the reactor coolant pressure boundary). Thus, the SLC system meets the requirements of GDC 2, "Design Basis for Protection Against Natural Phenomena," and the guidelines of RG 1.29, "Seismic Design Classification," Position C.1.

The secondary containment in which the system is located provides protection against externally or internally generated missiles. The SLC system is separated from nonseismic system components and from the effects of breaks in other high- and moderate-energy piping systems (see Section 3.5.1.2 and 3.6.1 of this report). Thus, the SLC system meets the requirements of GDC 4, "Environmental and Missile Design Bases."

The pumps and valves are powered and controlled from separate buses and circuits so that a single active failure will not prevent system operation. The ATWS analyses have verified that one pump with 50 gpm capacity can safely shut down the reactor. The injection portion of the system can be functionally tested by injecting demineralized water from a test tank into the reactor.

Since the ABWR SLC system pump is started manually, rather than automatically as required by the anticipated transient without scram (ATWS) rule 10 CFR 50.62, technical justification in support of a Part 50.62 exemption request should be submitted for staff review. (See Section 4.6 of this report)

Subject to the open issue discussed above, the staff concludes that the SLC system meets the requirements of GDCs 2, 4, 26, and 27 as they relate to protected against natural phenomena, system function and redundancy, and testability, and the guidelines of RG 1.29, Position C.1., as related to seismic classification of the system, and is, therefore, acceptable. The SLC system meets the acceptance criteria of SRP Section 9.3.5. However, the SLC system does not satisfy the requirements of 10 CFR 50.62.

9.3.6 Instrument Air System

This section will be provided in a supplement to this SER.

9.3.7 Service Air System

This section will be provided in a supplement to this SER.

9.3.8 Radioactive Drain Transfer System

This section will be provided in a supplement to this SER.

9.3.9 Hydrogen Water Chemistry System

This section will be provided in a supplement to this SER.

9.3.10 Oxygen Injection System

This section will be provided in a supplement to this SER.

9.3.11 Zinc Injection System

This section will be provided in a supplement to this SER.

9.4 Heating, Ventilation, and Air Conditioning (HVAC) Systems

9.4.1 Control Building HVAC

The control building ventilation system was reviewed in accordance with SRP Section 9.4.1 (NUREG-0800). The system is composed of two separate ventilation systems; a system for the main control room, and control room equipment and a system for essential electrical and heat exchanger equipment. Each of the systems is described below and is evaluated for compliance with the SRP acceptance criteria.

9.4.1.1 Control Room Equipment HVAC

The control building HVAC system serves the electrical area, corridors, control room, computer room, office areas, and the HVAC equipment room. The system consists of two fully redundant trains of equipment including ductwork, filters, cooling and heating coils, control dampers, isolation dampers, recirculation loop and fans. Each train also incorporates an emergency recirculation loop comprised of filters, heating and cooling coils, charcoal adsorbers and fans. Chilled water to the cooling coils is supplied by the essential cooling water equipment described below. The HVAC system is designed to maintain a controlled temperature environment under normal and accident operating conditions and provides for detection and filtration of smoke and radioactive material.

During normal operation the control room HVAC provides ventilation using a combination of filtered outdoor air and recirculated indoor air. The combined air stream is passed through a recirculation unit comprised of a medium grade filter, a chilled water cooling coil, a hot water heating coil, and a humidifier. Two parallel, 50 percent capacity supply fans draw air through the recirculation unit and deliver to the six service areas. Two parallel, 50 percent exhaust fans draw air from the six service areas and exhaust to the environment. The supply and return ducts have manual balancing dampers which are locked in place after the system is balanced. The supply and return ducts also have modulating dampers which are used to maintain the required positive pressure. The controller is located in the electrical equipment area. Sufficient air is provided to pressurize both the control room and control building. Nominal design conditions for the control room are 75 degrees Fahrenheit and 50 percent relative humidity with outside temperatures ranging from -40 to 115 degrees Fahrenheit. Details of this system description will be finalized on submission of Piping and Instrumentation Diagrams (P&IDs) and clarifying information in a future SSAR amendment.

The control room HVAC system and components are located in a seismic Category I building that is protected against tornado missiles and floods and are operable during loss of off-site power. All control room HVAC equipment, including ductwork, is of seismic Category I design. Outside air intake valves are protected against freezing and other environmental conditions and the intake vents are provided with tornado missile barriers. In accordance with SRP review procedures, these statements constitute acceptable means of meeting the requirements of General Design Criteria 2 and the provisions of Regulatory Position C.1 and C.2 of RG 1.29.

Evaluation against the requirements of GDC 4 for maintenance of control room environmental conditions will be completed following submission of design descriptions in a future SSAR amendment.

Smoke detectors in the control room and the control equipment room activate an alarm on detection of smoke. Atmospheric exhaust of 100 percent of conditioned air is provided by manual activation of a switch which closes return dampers and opens exhaust dampers. Supply and exhaust fans operate at low speed in smoke removal mode. Contingent upon provision of revised P&IDs showing detection capability, this system design meets the control room habitability requirements of TMI Action Item III.D.3.4 of NUREG -0737 with regard to smoke removal. Since the evaluation of requirements for protection against hazardous chemical releases will be site specific, applicants referencing this SSAR will provide further review against 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."

Radiation monitors are provided for detection of high radiation levels at control building air intake vents and for monitoring radiation level in the control room. Monitors alarm in the control room and the system provides for automatic or manual isolation from the outside atmosphere. Isolation involves closing outside air inlet dampers, closing outside atmosphere exhaust dampers and adjusting recycle dampers to route air through the emergency recirculation unit. The emergency recirculation system is an engineered safety feature atmospheric control system and is used only in high radiation conditions. Redundant isolation dampers downstream of the exterior air inlet filters provide for isolation of the system for operation in emergency recirculation mode. Radioactivity monitors at the inlets allow operator selection of inlet vent. The emergency recirculation unit is comprised of an electrical heating coil, a prefilter, a HEPA filter, a charcoal adsorber, a second HEPA filter and a booster fan. Exhaust from the emergency recirculation unit enters the primary recirculation unit. Connections are provided for testing the HEPA and charcoal filters' pressure drop and radioiodine removal efficiency. A differential pressure switch across the emergency recirculation system alarms on high pressure and a limit switch will provide an alarm on high temperature of air leaving the recirculation unit. A flow switch in the exhaust fan discharge alarms on fan failure and automatically starts the back-up system.

Contingent upon the provision of revised P&IDs showing detection capability, and based on the above description, the system meets the design, redundancy, component, instrumentation and testing provisions of Regulatory Positions C.1, C.2, and C.3 of RG 1.52 for post-accident clean-up systems. Since the emergency recirculation unit is in series with the primary recirculation unit, the humidity control of the primary unit is considered sufficient to meet the demister provisions of Regulatory Position C.2(a) of RG 1.52 for protection of charcoal adsorbers. The safety-related portions of the system used during normal operations do not have a radionuclide removal function and therefore are not reviewed against the provisions of RG 1.52.

9.4.1.2 Essential Electrical and Reactor Building Cooling Water Equipment (HVAC)

Three similar but independent HVAC systems are provided to service the essential electrical and heat exchanger equipment of Divisions A, B and C of the Essential Electrical System. Safety-related areas serviced by each system include battery rooms, an essential chiller room, a reactor building cooling water pump and heat exchanger room, and a safety-related electrical equipment room. The systems differ in regard to safety-related areas serviced in that the system serving Division C includes an SGTS equipment room and 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 system serves non-safety-related passages and HVAC rooms and Subsystem 1, in addition, serves a non-essential battery room, and a non-essential electrical equipment room. Each system draws exterior air through a recirculation unit and the serviced areas using two 50 percent capacity supply and two 50 percent capacity exhaust fans. Each recirculation unit is comprised of a pre-filter, a HEPA filter, an electric heater and a cooling coil. During smoke removal mode the fans are placed on a low speed setting. This system description will be updated following submission of revised P&IDs.

All components of each system are of seismic Category I design, tornado missile barriers are provided for air intake and exhaust vents, and the surrounding building is of seismic Category I design. Therefore, the systems are protected from the effects of earthquake, tornado and flood and meet the requirements of GDC 2 and the provisions of Regulatory Positions C.1 and C.2 of RG 1.29.

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. By virtue of location within the control building the systems are protected from dynamic effects. Therefore, the systems meet the requirements of GDC 4.

Since the RBCW pump rooms are located in an area of the secondary containment which is potentially contaminated following an accident, the essential electrical and reactor building HVAC system is reviewed against the requirements of GDC 60 and the provisions of RGs 1.52 and 1.140. As presently constituted the design does not meet the component and filtration specifications of the RGs. Therefore, this remains an open item.

The rooms served by this system are not occupied under accident conditions and the system is not reviewed against the requirements of GDC 19 and the provisions of RGs 1.78 and 1.95.

9.4.2 Spent Fuel Pool Area Ventilation System

See Section 9.4.5.

9.4.3 Auxiliary and Radwaste Area Ventilation System

The auxiliary area ventilation system is part of the reactor building ventilation system which was reviewed in accordance with SRP Section 9.4.3. The review determined that the auxiliary area system is not safety related and failure of the system will not compromise the function of safety-related equipment. Failure of the system does not affect the integrity of the reactor coolant pressure boundary, the capability to shutdown the reactor, or result in unacceptable releases of radioactivity. Therefore, the system meets the requirements of GDC 2 and the provisions of RG 1.29 and the requirements of GDC 60 are not applied in this review.

9.4.4 Turbine Area Ventilation System

The turbine area ventilation system was reviewed in accordance with SRP Section 9.4.4. The review established that the turbine building ventilation system has no safety function and that failure of the system does not compromise the operation of safety-related equipment. 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. Therefore the requirements of GDC 2 and the provisions of RG 1.29

are satisfied and the requirements of GDC 60 are not applied in this review.

9.4.5 Engineered Safety Feature Ventilation Systems

The reactor building ventilation systems provide cooling and supply and exhaust air for engineered safety feature equipment located in the reactor building. These HVAC systems were reviewed in accordance with SRP Section 9.4.5 and serve: 1) the secondary containment, 2) the essential equipment rooms, 3) the non- essential HVAC rooms, 4) the essential electrical equipment rooms, 5) the essential diesel generator areas, 6) the primary containment, 7) The mainsteam/feedwater tunnel, and 8) the reactor internal pump control panel room. Certain past designs have included elements of these systems in the Auxiliary Area.

9.4.5.1 Secondary Containment HVAC

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 operation of these safety-related components will be reviewed with P&IDs and interface descriptions to be provided in a future SSAR amendment.

9.4.5.2 Essential Equipment HVAC

The essential equipment HVAC system provides cooling to areas containing safety-related core cooling water pumps, standby gas treatment equipment and radiation monitoring equipment. The system is designed to provide the controlled temperature environment required for operation of this equipment. Separate fan coil units serve 14 rooms which house the redundant elements of 7 systems and all elements of the cooling system are essential. Power is provided by the divisional power of the serviced system.

All components of the essential equipment HVAC system are located in the reactor building which is a seismic Category I structure and are thus protected from the effects of earthquake, tornado and flood. The fan-coil cooling units are designated as seismic Category I. The SSAR provides a statement that all components of the system are designed to ESF requirements, satisfying the SRP requirement for a statement that all components are protected from floods and tornadoes and operable in the loss of off-site power. Under accident conditions, the secondary containment ventilation system is isolated by redundant inlet and outlet dampers, protecting the system from accumulation of external dust. Therefore, the system meets the requirements of GDC 2 and GDC 17, and the provisions of RG 1.29.

Each cooling system is protected against damage from internally generated missiles by separation of redundant equipment. This meets one of the requirements of GDC 4. Complete evaluation against GDC 4 requirements are to be provided in a future SSAR amendment. Details of the essential equipment cooling systems description will be finalized on submission of P&IDs and clarifying information in a future SSAR amendment.

The system does not exhaust air from the equipment areas and consequently is not reviewed against the requirements of GDC 60 and the provisions of RG 1.52. The system is not used during normal operation and is therefore not reviewed against the provisions of RG 1.140.

9.4.5.3 Non-Essential HVAC System

The non-essential HVAC system was reviewed in accordance with SRP Section 9.4.5. The system is comprised of nine fan coil units; each unit cools one room of the secondary containment. The review established that this system has no safety function and that failure of the system does not compromise the operation of safety-related equipment. 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. Therefore the requirements of GDC 2 and the provisions of RG 1.29 are satisfied and the requirements of CDCs 4, 17 and 60 are not applied in this review.

9.4.5.4 Essential Electrical Equipment HVAC

The essential electrical equipment HVAC systems provide air inlet and exhaust, and heat removal to maintain environmental conditions suitable for operation under accident conditions. The essential electrical equipment HVAC is comprised of three divisions, each serving five areas. The five areas are the day tank, diesel generator, diesel generator control panel, electrical equipment, and HVAC equipment rooms. These systems are evaluated in accordance with SRP Section 9.4.5.

Each of the three essential electrical equipment HVAC systems is comprised of a recirculation unit and redundant, 100 percent capacity supply and exhaust fans. Each recirculation unit includes a medium grade filter, an essential chilled water cooling coil, and a hot water heating coil. Each room is maintained at positive pressure with respect to the environment under normal and accident conditions. The systems are located in the seismic Category I reactor building, all components are rated seismic Category I, and air intake and exhaust structures are protected against tornado missiles. Therefore, the systems are protected against the effects of earthquake, tornado, and flood, and meet the requirements of GDC 2 and the provisions of Regulatory Positions C.1 and C.2 of RG 1.29.

The power supply to the essential electrical equipment HVAC systems are uninterrupted during loss of off-site power and the air intake structures are located more than thirty feet above grade; thereby meeting the requirements of GDC 17 for protection against dust. Complete evaluation against GDC 4 requirements are to be provided in a future SSAR amendment. Details of the essential equipment cooling systems description will be finalized on submission of P&IDs and clarifying information in a future SSAR amendment.

The essential electrical equipment HVAC system serves areas which are physically separated from potentially contaminated areas and are maintained at positive pressure relative to these areas. Consequently, the system is not reviewed against the requirements of GDC 60 and the provisions of RG 1.52. The monitoring and electrical interfaces which assure the operation of the SGTS to maintain negative pressure will be reviewed following provision of the interface descriptions in a future SSAR amendment. The safety-related portions of the system used during normal operations do not have a radionuclide removal function and are, therefore, not reviewed against the provisions of RG 1.140.

9.4.5.5 Essential Diesel Generator HVAC System

The essential diesel generator HVAC system is comprised of three duplicate systems each of which services one of the three diesel generators. The system was reviewed in accordance with SRP Section 9.4.5.

Each essential diesel generator HVAC system is comprised of a medium grade inlet air filter, two supply fans, ductwork, and exhaust louvers. The fans are interlocked with the diesel generator starting system, but manual override is provided. The systems are designed to facilitate inspection, the air intake ducts are located more than thirty feet above grade, and the redundant systems function during loss of off-site power. Therefore, the systems meet the requirements of GDC 17 for protection against dust accumulation.

Each essential diesel generator HVAC system is comprised of seismic Category I components, the air intake and exhaust structures are protected against tornado missiles, and each system, by virtue of location in the seismic Category I reactor building, is protected from the effects of tornado and flood. Consequently, the systems meet the requirements of GDC 2 and the provisions of Regulatory Positions C.1 and C.2 of RG 1.29.

Due to location in redundant, dedicated rooms the essential diesel generator HVAC systems are protected against dynamic effects satisfying one of the requirements of GDC 4. Complete evaluation against GDC 4 will be provided in a future SSAR amendment.

The diesel generator HVAC system serves areas which are physically separated from potentially contaminated areas and are at positive pressure relative to surrounding areas due to the use of supply fans and operation of the SGTS. The monitors and interfaces which assure proper function of the SGTS will be reviewed following submission of interface descriptions and revised P&IDs in a future SSAR amendment. Given these conditions, the essential diesel turbine HVAC system is not reviewed against the requirements of GDC 60 and the provisions of RG 1.52.

9.4.5.6 Containment Purge Supply/Exhaust System

The containment purge supply/exhaust system ventilates the primary containment using supply from, and exhaust to, the secondary containment HVAC system. Operation of the system was reviewed in accordance with SRP Section 9.4.5.

The system is used during operation and following shutdown, but operation of the system is not required to ensure the integrity of the reactor coolant pressure boundary or to shutdown the reactor. The containment purge/supply system is comprised of a purge supply fan, a HEPA filter, a purge exhaust fan, duct work and controls. The system takes its supply from the secondary containment HVAC system supply. When radioactivity is not present the air is discharged to the secondary containment HVAC system for filtration and exhaust. When radioactivity is present the system exhausts to the SGTS.

All components of the system are seismic Category I and the system by virtue of location within the seismic Category I reactor building is protected from the effects of earthquake and flood. Therefore, the system meets the requirements of GDC 2 and the provisions of RG 1.29.

The safety-related components are those which transfer exhaust from the secondary containment system to the SGTS on detection of radioactivity. The function of these components will be reviewed following submission of revised P&IDs and descriptions of interface systems in a future SSAR amendment.

Review of the system's ability to meet the environmental compatibility and protection from dynamic effects requirements of GDC 4 and the single failure and isolation capability requirements of the SRP will be finalized on submission of a detailed system description, P&IDs, and clarifying information in a future SSAR amendment.

9.4.5.7 Mainstream/Feedwater Tunnel HVAC System

The mainsteam/feedwater tunnel HVAC system was reviewed in accordance with SRP Section 9.4.5. The system is comprised of two closed loop fan-coil recirculation units which remove heat from the tunnel area. The review established that the system has no safety function and that failure of the system does not compromise the operation of safety-related equipment. 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. Therefore, the requirements of GDC 2 and the provisions of RG 1.29 are satisfied and the requirements of GDC's 4, 17 and 60 are not applied in this review.

9.4.5.8 Reactor Internal Pump Control Panel Room HVAC System

The reactor internal pump control panel room HVAC system was reviewed in accordance with SRP Section 9.4.5. The system is comprised of a medium grade filter, heating/cooling coils, two supply fans and two exhaust fans. Identical systems ventilate reactor internal pump control panel rooms serving Divisions 1 and 2. The review established that the system has no safety function and that failure of the system does not compromise the operation of safety-related equipment. The 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. Therefore, the requirements of GDC 2 and the provisions of RG 1.29 are satisfied and the requirements of GDCs 4, 17 and 60 are not applied in this review.

9.4.6 Radwaste Building HVAC System

The radwaste building ventilation system was reviewed in accordance with SRP Section 9.4.3. The review determined that the system is not safety related and that failure of the system will not compromise the function of safety-related equipment. Failure of the system does not affect the integrity of the reactor coolant pressure boundary, the capability to shutdown the reactor or result in unacceptable releases of radioactivity. Therefore, the system meets the requirements of GDC 2 and the provisions of Regulatory Position C.2 of RG 1.29 and the requirements of GDC 60 are not applied in this review.

9.4.7 Diesel Generator Area Ventilation System

The diesel generator building ventilation system is part of the reactor building ventilation system reviewed in Section 9.4.5.

9.4.8 Service Building Ventilation System

The service building ventilation system was reviewed in accordance with SRP Section 9.4.3. The review determined that the system is not safety related and that failure of the system will not compromise the function of safety-related equipment. Failure of the system does not affect the integrity of the reactor coolant pressure boundary, the capability to shutdown the reactor, or result in unacceptable releases of radioactivity. Therefore, the system meets the requirements of GDC 2 and the provisions of Regulatory Position C.2 of RG 1.29 and the requirements of GDC 60 are not applied in this review.

9.4.9 Drywell Cooling System

The drywell cooling system was reviewed in accordance with SRP Section 9.4.5. The system is comprised of three fan coil recirculation units which control drywell temperature during normal operation. The review established that the system has no safety function and that failure of the system does not compromise the operation of safety-related equipment. 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. Therefore, the requirements of GDC 2 and the provisions of RG 1.29 are satisfied and the requirements of GDCs 4, 17 and 60 are not applied in this review.

9.5 Other Auxiliary Systems

9.5.1 Fire Protection Program

This section will be provided in a supplement to this SER.

9.5.2 Communications Systems

This section will be provided in a supplement to this SER.

9.5.3 Lighting Systems

This section will be provided in a supplement to this SER.

9.5.4 Emergency Diesel Engine Fuel Oil Storage and Transfer System

This section will be provided in a supplement to this SER.

9.5.5 Emergency Diesel Engine Cooling Water System

This section will be provided in a supplement to this SER.

9.5.6 Emergency Diesel Engine Starting System

This section will be provided in a supplement to this SER.

9.5.7 Emergency Diesel Engine Lubricating Oil System

This section will be provided in a supplement to this SER.

9.5.8 Emergency Diesel Engine Combustion Air Intake and Exhaust System

This section will be provided in a supplement to this SER.

10 STEAM AND POWER CONVERSION SYSTEM

10.1 Summary Description

The steam and power conversion system is designed to remove heat energy from the reactor and to generate electric power in the turbine generator. After the steam passes through the high- and low-pressure turbines, the main condensers deaerate the condensate and transfer the rejected heat to the closed-cycle circulating water system, which, in turn, rejects the heat to the power cycle heat sink (cooling tower basin, where applicable). The condensate is reheated and returned as feedwater to the steam generator. The entire system is designed for the maximum expected energy from the nuclear steam supply system.

A turbine steam bypass system is provided to discharge directly to the condenser up to 33-percent of the reactor's design steam flow around the turbine during transient conditions.

10.2 Turbine Generator

The turbine generator is an 1800-rpm unit. It is a tandem compound type with one double-flow high-pressure turbine and three double-flow low-pressure turbines coupled directly to a 1400-MWe generator. Each low-pressure turbine exhausts to a multipressure three-shell, single pass, surface condenser.
The turbine generator is equipped with an electrohydraulic control system that performs two basic functions: (1) turbine speed control for a variety of operating load conditions for which a digital control and monitoring (DCM) system is used and (2) turbine overspeed protection. The design functions of the turbine speed control system are (1) to control turbine speed

throughout the normal range of load conditions and ensure that a full-load turbine trip will not cause the turbine to overspeed beyond acceptable limits and (2) to provide turbine overspeed protection to minimize the probability of the generation of turbine missiles, in accordance with General Design Criteria (GDC) 4. The turbine control system is, therefore, important to the overall safe operation of the plant.

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

The speed control unit is designed to provide a speed error signal input to a load control unit on the basis of differences between actual turbine speed and/or acceleration, and fixed speed/acceleration references. The load control unit combines the speed error signal with a load reference to determine desired steam flow signals. These steam flow signals go to a valve-flow-control unit that provides positioning signals to the hydraulic operating mechanisms of the control valves. These, in turn, operate to open or close the valve, as required, to maintain desired turbine steam flow. In the case of a large generator load rejection up to and including full load followed by an increase in turbine speed, the speed-control unit will close both the control and intercept valves to limit turbine overspeed as follows: (1) the control and intercept valves start to close at approximately 101-percent of rated speed and (2) the control and intercept valves are fully closed by the time the turbine reaches approximately 104-percent of rated speed. The control and intercept valves do not trip; they will reopen when the turbine returns to rated speed.

The turbine overspeed protection system is comprised of two overspeed control systems. The mechanical overspeed trip system consists of a steel ring mounted on the turbine shaft and held concentric with the turbine shaft by a spring. At a predetermined speed (110-percent of rated speed), centrifugal force acting on the ring exceeds the spring force and moves the ring to an eccentric position where it strikes a trip lever, causing the loss of hydraulic pressure to all the turbine valves, thus closing them. The electrical overspeed trip system is a backup to the mechanical overspeed trip. The electrical overspeed trip consists of three independent (separate from the load speed control unit) speed transducers, two 24-V dc solenoids, and an electrical trip valve. At a predetermined speed (111-percent of rated speed), the 24-V dc solenoids are deenergized. This, in turn, actuates the electrical trip valve to release hydraulic pressure to all turbine valve actuators, thus closing them. The electrical overspeed signal also energizes a 125-V dc solenoid that actuates the mechanical trip valve described above. Following a trip, all turbine valve actuators remain closed until the trips are reset and the operator takes action to reopen the turbine valves.

A number of turbine generator electrical and mechanical parameters are monitored during operation. An abnormal condition in these monitored parameters will also cause a trip of all turbine valves. These emergency trips further reduce the possibility of a turbine missile by shutting down the turbine before overspeed or mechanical failures can occur. Some parameters monitored include power/load imbalance, vibration, various temperatures and water levels, condenser vacuum, reverse power, lube oil pressure, and thrust bearing wear.

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

An inservice inspection program for the main steam stop and control valves and combined intermediate valves is provided and includes: (1) dismantling and inspection of at least one main steam stop valve, one main steam control valve, one stop valve, and one intercept valve at approximately 3-1/3-year intervals during refueling or maintenance shutdowns coinciding with the inservice inspection schedule; and (2) testing at least once a month the main steam stop and control, intermediate stop, and intercept valve and the extraction steam nonreturn valves. GE has

included preoperational and startup tests of the turbine generator in accordance with RG 1.68, "Initial Test Programs for Water Cooled Power Plants." The adequacy of the test program is evaluated in Chapter 14 of this report.

GE has committed to provide the turbine generator equipment shielding and access control for all areas of the turbine building that will meet the dose criteria required by 10 CFR Part 20 for operating personnel.

The connection joint between the low-pressure turbine exhaust and the main condenser is a stainless steel expansion joint. Since there is no safety-related equipment located in the turbine area, failure of this joint will cause no adverse effect on safety-related equipment.

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

The scope of review of the turbine generator included descriptive information in SSAR Section 10.2, flow charts, and diagrams. The basis for acceptance in the staff review was conformance of the design criteria and bases and design of the turbine-generator system to GDC 4 with respect to the prevention of the generation of turbine missiles, the additional guidance of SRP Section 10.2, and industry codes and standards.

On the basis of its review, the staff concludes that the turbine generator system meets GDC 4 and SRP Section 10.2, and can perform its design function, and is, therefore, acceptable.

10.3 Main Steam Supply System

The main steam supply system is designed to supply the required amount of steam at the required pressure and temperature to the turbine, reheaters, condenser evacuation system, turbine gland seal system, and the offgas system.

The main steam supply system extends from the reactor to the turbine stop valves and, additionally, includes connected piping up to and including the first shutoff valve on the connected lines. The safety-relief valves, which are mounted on the main steamlines upstream of the containment isolation valves for the system, are evaluated separately in Section 5.2.2 of this report. The main steam supply system was reviewed in accordance with SRP Section 10.3 (NUREG-0800). The acceptance criteria therein provided the basis for the staff's evaluation of the system. The acceptance criteria for the main steam supply system includes meeting the guidelines of RG 1.115 "Protection Against Low-Trajectory Turbine Missiles." Compliance with the guidelines of the regulatory guide are evaluated separately in Section 3.5.1.3 of this report.

The steam generated in the reactor vessel is routed to the turbine and power cycle auxiliary equipment via four 28-inch main steamlines. Each main steamline is equipped with a flow restrictor and two main steam isolation valves (MSIVs), thus assuring main steamline isolation in the event of a steamline break outside the containment and a concurrent failure of an MSIV. One MSIV is located immediately inside the drywell and the other immediately outside the drywell. The MSIVs are designed to provide positive isolation against steam flow associated with a main steamline break. They are pneumatic or spring-operated (to close), fast-closing (3-4.5 seconds), Y-pattern, globe valves. Operating fluid, that is, nitrogen, is supplied to the valves from the nitrogen supply system. Nitrogen accumulators are provided to supply backup operating nitrogen for the MSIVs in the event of loss of the normal nitrogen supply system. GE should clarify whether the backup nitrogen accumulators are seismic Category I.

Downstream of the outboard MSIVs, the main steamlines that are routed to the turbine contain no other shutoff valves except for the turbine stop valves. From the main steamline header, in addition to the four steamlines to the turbine, two steamlines are provided to supply steam to the power cycle auxiliary equipment. One of the branch steamlines supplies steam to the turbine gland seal system and to two reheaters. The other branch line provides steam to the offgas system, the steam jet air ejectors, the condenser sparger, and two other reheaters. Each of these steamlines contains a shutoff valve which is a power-operated gate valve. These valves are 16 inch diameter, Quality Group D, American National Standards Institute (ANSI) B16.1 design code, have a closure time of 2 seconds, and are equipped with air operators and spring closure mechanisms.

ABWR SSAR Section 5.4.9 and Table 3.2-1 indicate that the main steamlines from the reactor vessel, out to and including the outboard MSIVs, are designed to Quality Group A. From the outboard MSIVs to the turbine stop valves, the steamlines and associated equipment are designed to Quality Group B. The main steamlines that extend from the reactor vessel up to and including the seismic interface restraint, which is downstream of the outboard MSIVs and in the reactor building, are seismic Category I. Downstream of the seismic interface restraints, the main steam piping and equipment are classified as non-nuclear safety-related. This includes the shutoff valves on the two branch steamlines (inside the turbine building) supplying steam to the power cycle auxiliary equipment (Quality Group D).

ABWR SSAR Section 10.3.3, however, states that the main steamlines from the containment outboard isolation valves, up to and including the turbine stop valves and all branch lines 2.5 inches and larger, up to and including the first valve and its supports, will be designed by the use of an appropriate dynamic seismic system analysis to withstand the operating basis earthquake (OBE) and the safe shutdown earthquake (SSE) design loads in combination with appropriate loads, within the limits specified for Class 2 pipe in the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) Section III. This design for the main steamlines downstream of the outboard MSIVs up to the turbine stop valves, the connecting lines up to the turbine bypass valves, and all other connecting lines up to and including the first shutoff valves, does not comply with staff's requirement for seismic and quality group classifications as stated in SRP Section 10.3, Criterion III.3.b. That criterion requires that the subject portions of the main steam supply system be designed to seismic Category I and Quality Group B requirements. This remains an open item.

The steamlines in the reactor building (including the containment and some portion of the steam tunnel) and in the steam tunnel portion in the control building are located in seismic Category I, flood-protected and tornado-protected structures. Thus, the requirements of GDC 2 and the guidelines of RG 1.29, Positions C.1 and C.2 are met for these portions of the main steam supply system, with the exception of seismic and quality group classifications for certain portions of the main steam supply system identified above.

ABWR SSAR Appendix 3F indicates that since the safety-related portions of the condensate and feedwater system (CFS) and the main steam supply system (from the reactor up to and including the seismic interface restraints) are qualified for the "leak before break" (LBB) criterion, no high-energy pipe break need be considered in these portions of the above systems solely for considering the local dynamic effects associated with such breaks. Appendix 3F has provided generic LBB evaluations procedures and methodology for the systems in support of the claim. As stated in Section 3.6.3 of this report, 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. Therefore, the staff will evaluate the acceptability of the above LBB claim on a plant-specific basis to determine whether the main steam supply system essential portion is adequately protected against dynamic effects associated with high-energy pipe breaks. Regarding the other aspect of GDC 4, which deals with the environmental design basis for SSE important to safety, ABWR SSAR Appendix 3I indicates that the essential equipment of the system is environmentally qualified to function following a postulated high-energy pipe break. Specifically, this means that the main steam isolation valves (MSIVs) that are required to function in order to ensure main steam isolation, are qualified to function in the expected steam environment resulting from a steamline break. Further, GE has identified an interface requirement for a referencing applicant which requires the applicant to provide details of how the MSIV functional capability is protected against the effects of postulated pipe failures. GE has also addressed the issue of steam hammer and relief valve discharge loads (submittal dated February 28, 1990) and states that the system design accommodates steam hammer and relief valve discharge loads. The staff will require referencing applicants to have operating and maintenance procedures which include adequate precautions to avoid steam hammer and relief valve discharge loads. The system design includes drains to protect against water entrainment. The essential equipment of the system is located in tornado-missile-protected structures (as stated above) and is separated from the effects of internally generated missiles. On this basis, the staff finds that the safety-related portion of the system meets the requirements of GDC 4 and the guidelines of RG 1.117 "Tornado Design Classification," Appendix Positions 2 and 4, pending the staff's approval of the plant specific LBB analyses for the main steam supply system and the CFS (both of these systems are considered since portions of the main steam supply lines and the feedwater lines are in the steam tunnel).

Therefore, the staff concludes that the main steam supply system for the ABWR from the reactor to the turbine building satisfies the requirements of GDCs 2 and 4 with respect to protection against natural phenomena, floods, tornados, missiles, and environmental effects, and the guidelines of RG 1.29, Positions C.1 and C.2; and RG 1.117, Appendix Positions 2 and 4, relating to the system's seismic classification and protection against tornado missiles and thus meets SRP Section 10.3 acceptance criteria and is, therefore, acceptable, subject to (1) meeting the seismic and quality group classifications for certain portions of the system, (2) clarification of the seismic category classification for the backup nitrogen accumulators for the MSIV, and

(3) the staff's approval of the plant-specific LBB analysis for the essential portions of the main steam supply system and the CFS.

10.4 Other Features

10.4.1 Main Condenser

The main condenser is designed to function as a steam cycle heat sink for the turbine exhaust steam, the turbine bypass steam and other turbine cycle flows, and to receive and collect condensate flows for return to the reactor. The main condenser transfers heat to the circulating water system, which, in turn, rejects the heat to the power cycle heat sink.

The main condenser is not required to serve or support any safet function or safe shutdown of the reactor. The main condenser is comprised of three, multipressure, two-tube bundle, single-pass shells. Each of the shells is located under its respective low-pressure turbine.

The main condenser hot well is sized and designed to provide a normal retention of 4 minutes' duration for all condensate from the time it enters the hot well until it is removed by the condensate pumps. Condensate is retained in the condenser for a minimum of 2 minutes to permit radioactive decay before it enters the condensate system. Offgas from the main condenser is processed in the gaseous radwaste system, which is described in SER Section 11.3. The main condenser is designed to (1) deaerate the condensate to the required water quality, (2) remove air and noncondensable gases, and (3) remove hydrogen and oxygen formed in the steam as a result of disassociation of water in the reactor.

Circulating water on the main condenser tubeside is treated with chemicals to limit algae growth, and to prevent long-term corrosion of the tubes. Corrosion on the shell side of the condenser is controlled by adhering to strict water quality. The construction materials used for the main condenser are chosen so that corrosion due to galvanic and other effects can be kept at a minimum.

Condenser leakage will be inleakage since the main condenser is normally operated under vacuum. Tube leakage is monitored by measuring the conductivity of water samples taken beneath the tube bundles. Additionally, since the condensate is monitored at the condensate pump discharge, any tube leakage will also be detected at this monitoring point. Conductivity of the condensate is continuously monitored at selected locations in the condenser. Condenser vacuum is also monitored. The loss of the main condenser vacuum will cause a turbine trip and main steam isolation valve closure. Bypass valve closure is at 12 inches Hg vacuum, and main steam isolation valve closure occurs at 7 to 10 inches Hg vacuum. High condenser pressure turbine trip occurs at 22 inches Hg vacuum, and alarm at 24 inches Hg vacuum.

The condenser is designed to condense up to 33-percent of the full-rated turbine steam flow as bypass steam. The main condenser will be designed with plant-specific provisions to preclude direct impingement of bypass steam on the condenser tubes. Typical provisions consist of a horizontal, perforated, steam distribution pipe enclosed in a perforated guard pipe.

In the event of condensate system (condenser shell side) failure, the resulting flood level will be less than grade level and the flooding of the turbine building will not affect any safety-related equipment since no such equipment is located inside the turbine building. The failure of the associated circulating water system and its consequences are evaluated in Section 10.4.5 of this report.

Radioactivity monitoring provisions, including a capability to control effluent, are provided for the offgases from the main condenser. Also, the circulating water decant line is periodically sampled for radioactivity. Thus, the main condenser complies with GDC 60 with regard to control of releases of radioactive materials to the environment.

The review of the main condenser is based on the guidelines and review acceptance criteria outlined in SRP Section 10.4.1 and industry standards. Therefore, the staff concludes that the design of the main condenser is in conformance with the SRP review criteria, can perform its design function, and is acceptable.

10.4.2 Main Condenser Evacuation System

The review of the main condenser evacuation system (MCES) is based on the acceptance criteria outlined in SRP Section 10.4.2, which include the requirements of GDCs 60 and 64, and guidelines contained in RGs 1.26, "Quality Group Classifications and Standards for Water-Steam and Radioactive-Waste-Containing Components of Nuclear Power Plants;" 1.33, "Quality Assurance Program Requirements (Operation);" and 1.123, "Quality Assurance Requirements for Control of Procurement of Items and Services for

Nuclear Power Plants;" and in the Heat Exchanger Institute's "Standards for Steam Surface Condensers."

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

The MCES consists of a mechanical vacuum pump for use during startup, and two 100-percent capacity, double stage, steam jet air ejector (SJAE) units (complete with intercondenser) for normal operating conditions. During startup, the mechanical vacuum pump is used to establish vacuum in the main condenser, and the exhaust gas is vented to the turbine building ventilation exhaust system. High main steamline radioactivity will trip the mechanical vacuum pump. The turbine building ventilation exhaust passes through a medium efficiency filter and is monitored for radioactivity before it discharges to the plant vent.

After the mechanical vacuum pump creates an absolute pressure of about 10 to 15 inches Hg in the main condenser, and adequate nuclear steam pressure is available, one of the two SJAEs is put in service to remove noncondensable gases from the condenser. Steam supply to the second stage of the SJAE is kept at a minimum predetermined flow to ensure adequate dilution of hydrogen to limit the hydrogen offgas from reaching its flammable concentration. The hydrogen concentration of the outlet of the second stage air ejector is maintained below 4-percent by volume by the dilution steam. The offgas portion of the MCES downstream of the second-stage of the SJAEs and interconnecting piping is designed to withstand the effects of hydrogen detonation. Also, low-flow of the dilution steam will result in automatic isolation of the main condenser from the offgas system. Two independent hydrogen analyzers are provided to monitor hydrogen concentration in the offgas steam downstream of each offgas condenser, and the signals are indicated, recorded, and alarmed in the main control room in case of high hydrogen concentration in the offgas. The hydrogen analyzers are also designed to withstand a hydrogen detonation.

The MCES is designed to Quality Group D in accordance with RG 1.26. The ABWR SSAR also provides assurance that the MCES will be designed to meet the quality assurance requirements for design, construction, and operation as outlined in RGs 1.33 and 1.123.

The MCES includes equipment and instruments that establish and maintain condenser vacuum and prevent uncontrolled release of gaseous radioactive material to the environment. The scope of the review includes the capability of the MCES design to remove radioactive gases from the main condenser and exhaust them to the offgas system or the turbine building ventilation exhaust system, and the design provisions that incorporate the capability of monitoring and controlling releases of radioactive materials. The staff concludes that the MCES design meets the requirements of GDCs 60 and 64 as related to the control and monitoring of release of radioactive materials to the environment and thus meets the SRP Section 10.4.2 acceptance criteria and is, therefore, acceptable. The ABWR SSAR provides assurance that the MCES design will also conform to the guidelines given in RGs 1.26, 1.33 and 1.123.

10.4.3 Turbine Gland Sealing System

The review of the turbine gland sealing system (TGSS) is based on the criteria outlined in the SRP Section 10.4.3.

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

The annular space between the turbine shaft and the casing is sealed with sealing steam supplied to the shaft seals. At all gland seals, the vent annulus is kept under a slight vacuum condition and also receives air inleakage from the outside. The steam mixture from the vent annulus is then pulled to the gland steam condenser which is operated under a slight vacuum condition created by one of the two exhauster blowers. The steam mixture is condensed in the gland steam condenser and the condensate is returned to the main condenser. The blower is designed to discharge the air inleakage and the noncondensable gases to the turbine building ventilation exhaust system, which eventually discharges to the plant vent. As mentioned above, the TGSS is also designed to provide sealing steam to the turbine stop and control valve, and combined intermediate valve (CIV) packings.

As stated in GE's letter dated February 28, 1990 (response to Question 430.82), the TGSS exhaust is monitored for radiation before it discharges to the turbine building ventilation system. However, SSAR Table 11.5-2 does not indicate any separate process radiation monitoring solely for turbine gland sealing system exhaust. GE should clarify the above contradiction.

During startup, sealing steam is provided from the main steamline or the plant auxiliary steam header. Plant auxiliary steam, which is "clean steam," is supplied by a conventional plant startup package boiler. The use of main steam as sealing steam would not pose a significant long-term average release of radioactive material to the environment since the startup time is relatively short, and plant startup radioactivity is relatively low. Even in the case where abnormally high radioactivity content of the sealing steam occurs due to the use of main steam, the sealing steam supply can be switched to the auxiliary steam system, which provides clean steam (i.e., radioactivity free).

During normal operation (above approximately 50-percent load), the sealing steam supply is switched to process steam. Process steam is provided from the high-pressure heater drain tank vent header which is elaborately designed to provide relatively high-purity steam (i.e., practically free of radioactivity) for the gland sealing purpose. Again, if this normal source of sealing steam is observed to have high radioactivity content, the source for the sealing system will be switched to the plant auxiliary steam system which provides 100-percent backup to the normal source. Thus, the long-term average amount of radioactive material released to the environment will be minimal. The staff agrees with this approach for providing sealing steam, subject to the following interface requirement.

Interface Requirement: Utility applicants who reference the ABWR design required to provide the necessary procedures for switchover to the auxiliary steam system when monitored radiation level in the gland sealing system exhaust exceeds an acceptable preset level.

The TGSS is designed to Quality Group D as outlined in RG 1.26. The ABWR SSAR provides assurance that the TGSS will be designed to conform to the quality assurance requirements for design, construction, and operation according to the guidance of RGs 1.33 and 1.123.

The review of the TGSS includes the source of sealing steam and the provision incorporated into the design that will allow monitoring and controlling releases of radioactive material in the effluent in accordance with GDCs 60 and 64. The staff concludes that the TGSS design for the ABWR meets the requirements of GDCs 60 and 64 with respect to the monitoring and control of radioactive material released to the environment, and the guidelines of RG 1.26 with regard to quality group classification for the TGSS components, subject to the clarification of the monitoring issue and identification of the interface requirement mentioned above. The staff also finds GE's commitment to design the TGSS in compliance with the guidelines of RGs 1.33 and 1.123 acceptable. On this basis, the staff concludes that the TGSS design for the ABWR meets the acceptance criteria of SRP Section 10.4.3, subject to resolution of the above open item.

10.4.4 Turbine Bypass System

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

The TBS consists of three control valves (as stated in ABWR SSAR Section 10.4.4.2.2) that are housed in a common valved chest that is connected to the main steamlines upstream of the turbine stop valves, and three dump lines that separately connect each regulating valve outlet to one condenser shell. Each bypass valve is a 9 inch, globe-type valve operated by hydraulic fluid pressure with spring action to close. The valve chest assembly includes a hydraulic supply and drain piping, hydraulic accumulators, servo valves, fast-acting servo valves, and position transmitters. The bypass valves are equipped with a separate hydraulic fluid power unit.

The turbine bypass valves are designed to close on loss of main condenser vacuum, loss of electrical power, or loss of hydraulic system pressure. The bypass valves are designed to open whenever the actual steam pressure exceeds the preset steam pressure. Fast-acting servo valves are incorporated into the design of the bypass valves to allow the bypass valves to open rapidly in case a turbine trip or a generator load rejection occurs.

The TBS can be tested during operation. Periodic inspections will be performed on a rotating basis within a preventive maintenance program recommended by the manufacturer. As stated in GE's submittal dated February 28, 1990 (response to Question 430.84), the detailed design of the bypass valves will follow standard industry practice and reduce the bypassed steam pressure sequentially through orifices prior to entering the condenser.

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

The review of the TBS was based on the acceptance criteria outlined in SRP Section 10.4.4 and industry standards. On the basis of the description of the system summarized above, the staff concludes that the TBS is in conformance with the acceptance criteria of SRP Section 10.4.4, by meeting the requirements of GDC 4 with respect to failure of the TBS not adversely affecting essential systems or components, and GDC 34 as related to the ability to use the system for shutting down the plant during normal operation, and is, therefore, acceptable.

10.4.5 Circulating Water System

The acceptance criteria of SRP Section 10.4.5 (NUREG-0800) formed the basis for the staff's evaluation of the circulating water system (CWS) for the ABWR.

The CWS is designed to remove the power cycle water heat from the main condenser and transfer this heat to the power cycle heat sink. The CWS is not required to maintain the reactor in a safe shutdown condition or support any safety-related systems or components. The system is non-seismic Category I and is Quality Group D.

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

The CWS pumps are vertical, wet pit-type, capable of delivering about 200,000 gpm per pump. The discharge of each pump is equipped with a butterfly valve to allow isolation and maintenance of any one pump while the others are in operation. The CWS pumps are tripped and the pump and the condenser isolation valves are closed on a high-high level condenser pit signal. A condenser pit high-level alarm also exists in the control room.

The power cycle heat sink is designed to maintain the temperature of the water entering the CWS within the range of 32°F to 100°F. The CWS is designed to deliver water to the main condenser within a temperature range of 40°F to 100°F. The 40°F minimum temperature is maintained by recirculating warm water from the discharge side of the condenser back to the screenhouse.

The analysis of a complete rupture of a single expansion joint in the CWS indicates that the CWS and associated facilities are designed so that all credible spillage inside the turbine building will remain inside the condenser pit. No safety-related equipment is located inside the turbine building. It is not clear how the SSCs important to safety in the reactor building are protected against flooding in the turbine building, since there are access openings and penetrations below design flood level between the reactor building and turbine building (SSAR Table 3.4-2). Subject to resolution of this concern, the CWS design for the ABWR meets the requirements of GDC 4 with respect to protection of safety-related systems from failures of non-safety-related components of the CWS.

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

On the basis of the description of the system summarized above, the staff concludes that the CWS for the ABWR meets the SRP Section 10.4.5 acceptance criteria by complying with GDC 4 with respect to protection of safety-related systems or components from failures of non-safety-related CWS components, subject to resolution of the staff's concern relating to turbine building flooding identified above.

10.4.6 Condensate Cleanup System

This section will be provided in a supplement to this report.

10.4.7 Condensate and Feedwater System

The condensate and feedwater system (CFS) was reviewed in accordance with SRP Section 10.4.7 (NUREG-0800). The CFS consists of the piping, valves, pumps, heat exchangers, and associated controls and instrumentation that extend from the main condenser outlet to the reactor and to the heater drain system. The function of the system is to receive condensate from the main condenser hotwells, supply condensate to the condensate cleanup system, and deliver high-purity feedwater to the reactor, at the required flow rate, pressure, and temperature. The major equipment in the CFS includes: (1) four identical, fixed-speed, motor-driven, condensate pumps, of which three are normally operating and one is on standby; (2) three identical and independent, 33-50-percent capacity reactor feed pumps; (3) three parallel and independent trains of four closed, low-pressure feedwater heaters; (4) two parallel and independent trains of two high-pressure feedwater heaters; (5) two heater drain tanks; and (6) two independent, motor-driven, heater drain pumps which take suction from a heater drain tank and discharge into the suction side of the feedwater pumps. The CFS is described in Sections 10.4.7 and 5.4.9 of the ABWR SSAR.

The CFS flow begins from the main condenser hot well drawn by three normally operated condensate pumps which pump it through the condensate filters and demineralizers. The condensate is discharged into a common header that feeds five parallel auxiliary condenser coolers (one gland steam exhauster condenser, two steam jet air ejector condensers, and two offgas recombiner coolers). The condensate is then fed to three parallel trains of low-pressure feedwater heaters which discharge into a common header that is routed to three parallel reactor feedwater pumps. The reactor feedwater pumps then discharge into two parallel high-pressure feedwater heater trains. Downstream of the high-pressure feedwater heaters, the feedwater is combined into a common header which discharges into the reactor through two parallel 22-inch-diameter feedwater lines, as stated in SSAR Section 5.4.9.3.

On each of the feedwater lines from the common feedwater header to the reactor, there is a seismic interface restraint, then a remote manual motor-operated gate valve powered by a non-safety grade bus which also serves as a feedwater shutoff valve. Downstream of the gate valve, there is a spring closing check valve held open by air as the outboard containment isolation valve. On the other side of the containment, there is a check valve which serves as the inboard containment isolation valve, and downstream of this check valve is a manual maintenance valve. The staff finds the provision of a non-safety-grade power source for the remote manual shutoff gate valve inappropriate since the valve and the piping portion, in which it is located are both seismic Category I. Additionally, the valve serves as a long-term isolation barrier. Therefore, subject to provision of an emergency bus for supplying power to the above valve, the staff concludes that the CFS will be isolated in the event of an accident coincident with a single failure in any isolation valve.

As indicated in Section 5.4.9 and Table 3.2-1 of the ABWR SSAR, the feedwater piping is Quality Group A from the reactor pressure vessel out to and including the outboard isolation valve, Quality Group B from the outboard isolation valve up to and including the seismic interface restraint, and Quality Group D beyond the shutoff valve. The feedwater piping and all connected piping $2\text{-}^1/_2$ inch or larger nominal size is seismic Category I from the reactor pressure vessel out to and including the seismic interface restraint. Table 3.2-1 states that piping beyond the seismic interface restraint is Quality Group D. On this basis, the staff concludes that the design of the CFS meets the requirements of GDC 2 and the guidelines of RG 1.29, Positions C.1 and C.2.

The essential equipment is separated from the effects of internally generated missiles. ABWR SSAR Appendix 3F indicates that since the safety-related portions of the main steam supply system and the CFS (from the reactor up to and including the seismic interface restraints) are qualified for the "leak before break" (LBB) criterion, no high-energy pipe break need be postulated in the safety-related portion of these systems solely for considering the local dynamic effects associated with such breaks. ABWR Appendix 3F has provided generic LBB evaluations (one for the steam supply system and the other for the CFS), procedures, and methodology in support of this claim. As stated in Section 3.6.3 of this report, the staff considers a bona fide LBB analysis to be the one that uses plant-specific data such as piping geometry, materials, fabrication procedures, loads, degradation mechanisms, and pipe support locations. Therefore, the staff will evaluate the acceptability of the LBB claim on a plant-specific basis to determine whether the CFS is adequately protected against dynamic effects associated with high-energy pipe breaks. Regarding the other aspect of GDC 4, which deals with the environmental design basis for SSC important to safety, ABWR SSAR Appendix 3I indicates that the essential equipment of the system is environmentally qualified to function following a postulated high-energy pipe break. GE has also addressed the issue of water-hammer loads due to hydraulic transients that can be caused by feedwater control valves rapidly interrupting feedwater flow (submittal dated February 28, 1990, response to Questions 430.89 and 430.90). GE plans to use a modified CFS design that will have only a low feedwater flow control valve specifically designed to minimize cycling in feedwater nozzles. The valve will be used only at low-power operating conditions. During normal power operations, feedwater flow will be varied as needed by using adjustable speed, motor-driven feed pumps, thus eliminating the need for any flow control valve. Furthermore, the staff will require the referencing applicant's commitment to review operating and maintenance procedures to ensure that precautions taken will minimize, or avoid, water-hammers. On the basis, the staff finds that the CFS design complies with GDC 4 pending the staff approval of plant-specific LBB analyses for the main steam supply system and the CFS (both of these systems are considered, since portions of the main steam supply lines and the feedwater lines are in a steam tunnel).

The feedwater system is not required to transfer heat under accident conditions; therefore, GDCs 44, 45, and 46 are not applicable.

The staff concludes that the CFS for the ABWR meets the requirements of GDCs 2 and 4 with respect to protection against natural phenomena, and missiles, and thus meets the SRP Section 10.4.7 acceptance criteria and environmental effects, and is, therefore, acceptable pending (1) staff approval of plant-specific LBB analysis for the main steam supply system and the CFS, and (2) provision of an emergency power supply source to the feedwater system shutoff gate valve.

10.4 8 Steam Generator Blowdown System (PWR)

This section is not applicable to the ABWR.

10.4.9 Auxiliary Feedwater System (PWR)

This section is not applicable to the ABWR.

11.0 RADIOACTIVE WASTE MANAGEMENT

The ABWR radioactive waste management systems are designed to provide for the controlled handling and treatment of liquid, gaseous, and solid wastes. The liquid radioactive waste system collects and processes liquid wastes from equipment and floor drains; sampling, decontamination and laboratory wastes; reactor water cleanup phase separator decant wastes; chemical wastes; and detergent wastes. The gaseous waste system provides catalytic recombiners to reduce the volume of offgases from the main condenser air ejector. It also provides holdup capacity in the form of charcoal delay beds to allow decay of short-lived noble gases from the main condenser air ejector and to adsorb radioiodines. It further includes high-efficiency particulate air (HEPA) filters to retain particulates present in the offgas stream. Thus, the system controls the release of gaseous radioactive effluents to the site environs so as to maintain the exposure of persons in unrestricted areas to as low as reasonably achievable (ALARA) in accordance with 10 CFR Part 20, 10 CFR 50.34a, and 10 CFR Part 50, Appendix I. The solid waste system provides for packaging spent resins and backwash slurries, solidifying concentrator bottoms, incinerating and packaging combustible dry radioactive materials, compacting and packaging noncombustible materials, and providing storage space for processed solid wastes before their shipment off site to a licensed facility for burial. The radioactive waste management review area includes the process and effluent radiological monitoring and sampling systems provided for the detection and measurement of radioactive materials in plant process and effluent streams.

Acceptance Criteria

The staff has reviewed the applicant's design, design criteria, and design bases for the radioactive waste management systems for the ABWR design. The acceptance criteria used as the basis for its evaluation are given in Section II of SRP Sections 11.1, 11.2, 11.3, 11.4, and 11.5. These acceptance criteria include the applicable general design criteria (GDC) (Appendix A to 10 CFR Part 50) and 10 CFR 20.106.

Because specific compliance with Appendix I to 10 CFR Part 50 and the guidelines given in American National Standards Institute (ANSI) Standard N13.1, "Guide to Sampling Airborne Radioactive Materials in Nuclear Facilities;" RG 1.21, "Measuring and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents From Light-Water-Cooled Nuclear Power Plants;" and RG 4.15, "Quality Assurance for Radiological Monitoring Programs (Normal Operation)--Effluent Streams and the Environment," is not totally within the scope of the ABWR, the staff will review individual utility license applications that refer to the ABWR design to ensure their conformance with Appendix I to 10 CFR Part 50, ANSI Standard N13.1, and RGs 1.21 and 4.15. Conformance with the acceptance criteria applicable to the ABWR design provides the bases for concluding that the radioactive waste management systems meet the requirements of 10 CFR Part 20.

Method of Review

The ABWR design has three radioactive waste management systems: the liquid waste management system, the gaseous waste management system, and the solid waste management system. Additionally, the

ABWR has several radiation-monitoring subsystems used for the process and effluent radiological monitoring and sampling systems. Each of these is reviewed in accordance with the applicable portions of SRP Sections 11.1 through 11.5. The findings of these reviews are discussed below.

11.1 Source Terms

The estimated releases of radioactive materials in gaseous effluents was calculated using the BWR GALE Code methodology described in NUREG-0016, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents From Boiling Water Reactors (BWR GALE Code)," Revision 1, January 1979. The calculations in the code for estimating the liquid and gaseous effluents during normal plant operation, including anticipated operational occurrences, are based on (1) data from operating reactors, (2) field and laboratory tests, (3) standardized coolant activities derived from American Nuclear Society (ANS) 18.1 working group recommendations, (4) release and transport mechanisms that result in the appearance of radioactive material in liquid streams, and (5) the plants radwaste system design features used to reduce the quantities of radioactive materials ultimately released to the environs. The principal parameters used in these calculations are given in Table 11.1. The gaseous source terms are given in Table 11.2. Capacities of principal components of the liquid and gaseous waste management systems are listed in Table 11.3 for a single-unit plant. The liquid effluent source terms will be reviewed on a plant-specific basis as discussed in Section 11.2 of this SER.

11.2 Liquid Waste Management System

11.2.1 System Description and Review Discussion

The liquid radioactive waste management system consists of process equipment and instrumentation necessary to collect, process, monitor, and recycle or discharge the processed radioactive liquid wastes. Treatment of liquid waste is dependent on the source, activity, and composition of the particular liquid waste and on the intended disposal procedure. The liquid wastes generated during operation will be collected and processed in three major liquid radwaste management subsystems and miscellaneous supporting subsystems. The three major subsystems are (1) the low-conductivity waste (LCW) (high purity) subsystem, (2) the high-conductivity waste (HCW) (low purity) subsystem, and (3) the detergent waste subsystem. These systems are described in detail in Section 11.2 of the ABWR SSAR. The ABWR uses hollow fiber filters in the LCW subsystem which require less backwash water than the precoat filters that are used in current designs. Also, it does not have recirculation pumps or associated valves and does not regenerate the deep bed condensate demineralizers or use ultrasonic resin cleaning. For these reasons, the staff expects less generation of waste in the LCW subsystem than that in older boiling water reactors (BWRs), and no condensate demineralizer regeneration waste.

The liquid radwaste treatment systems are designed to permit complete recycling of processed liquids from the low- and high-conductivity subsystems during normal operation. Processed liquids will be handled on a batch basis to permit optimum control and release of radioactive materials from the low-conductivity, high-conductivity, and detergent waste subsystems.

Discharge of processed LCW or HCW water will be solely governed by the plant water balance considerations. Before being released, samples will be analyzed to determine the types and amounts of radioactivity present. On the basis of the results of the analyses, the waste from the high and low conductivity subsystems will be recycled for eventual reuse in the plant, retained for further processing, or released under controlled conditions to the environment through the liquid pathway. All detergent wastes are expected to be released. A common radiation monitor in the discharge line will automatically terminate liquid waste discharges to the discharge canal from the LCW subsystem, HCW subsystem or the detergent waste subsystem if radiation measurements exceed a predetermined level in the applicable subsystem.

The LCW subsystem collects and processes clean wastes such as those from equipment drains (from the drywell, reactor, radwaste, and turbine buildings) and spent resin backwash transfer water. The wastes are collected in one or two parallel LCW collector tanks, filtered in one or two parallel hollow fiber filters (for the normal waste generation rate, one collector tank and one filter are used) for removal of insolubles, and demineralized in a mixed-bed demineralizer and a backup polishing demineralizer. Conductivity instrumentation on the demineralizer discharge routes the effluent either to the LCW sample tanks or back to the LCW collector tanks for reprocessing. From the sample tanks, the liquid stream is normally routed to the condensate storage tank for reuse. However, a small fraction of the processed waste may be discharged from one of the sample tanks should plant water balance considerations dictate such a discharge. The staff estimates that approximately 1 percent of the processed LCW will be discharged for the ABWR. The staff estimates that the normal waste generation rate for the LCW system will be about 15,200 gpd (GE's estimate: 14,530 gpd). The capacity of the limiting processing equipment is 190,080 gpd. The difference between the expected normal waste generation rate and the design process flow rate provides adequate reserve for processing a surge in LCW generation rate. The HCW subsystem collects and processes water of relatively high conductivity and solids content, such as the wastes from the floor drains (from drywell, reactor, radwaste, turbine, and service buildings). Additionally, it will collect and process chemical wastes (chemical lab wastes and lab drains). The wastes are collected in one or two parallel HCW collector tanks; they are chemically adjusted to a suitable pH for evaporation; are concentrated in one or two parallel forced-circulation concentrators or evaporators to reduce the volume of water and decontaminate the distillate. The distillate is demineralized in the HCW demineralizer and normally routed to the LCW system upstream of the polishing demineralizer. During normal processing, the LCW polishing demineralizer will be bypassed and the processed HCW will be directed to the LCW sample tanks from where the processed stream will be routed to the condensate storage tank for reuse. A small fraction of the processed HCW may be discharged from one of the LCW sample tanks should plant water balance dictate such a discharge. The staff estimates that approximately 10 percent of the processed HCW will be discharged. The distillate from the HCW demineralizer can also be routed to an HCW distillate tank to be reprocessed by the HCW demineralizer if required. The concentrated waste from the evaporator is routed to the concentrated waste storage tank for further processing by the solid waste system. The staff estimates that the normal waste generation rate for the HCW system will be approximately 6300 gpd (GE's estimate: 4000 gpd). The capacity of the limiting processing equipment in this system is 37,440 gpd. This provides adequate reserve for processing a surge in the HCW generation rate.

The detergent waste subsystem collects and processes detergent wastes from personnel showers and laundry operations. Detergent wastes are collected in the single hot shower drain (HSD) receiver tank, processed through one or two HSD filters, and routed to the HSD sample tank before discharge. GE estimates that the normal generation of detergent waste will be approximately 3000 gpd (staff's estimate: 1000 gpd). GE further states (June 7, 1990, submittal) that if storm drains are included, the waste generation for this subsystem will be approximately 8300 gpd. The storm drain water will be normally non-radioactive, but can become radioactive on contact with radioactive liquids. The storm drain water is collected in one of the two HSD sample tanks and is discharged after processing by the HSD filters if needed. The staff estimates that all of the detergent wastes and storm drains will be discharged. The capacity of the limiting processing equipment in this system is 75,000 gpd. This in combination with the tanks provided in the system ensures adequate margin to collect and process any surge in the waste generation in this system.

The liquid radwaste system provides one discharge line to the canal for the release of liquid waste. Radiation-monitoring equipment is placed on this line to measure the activity discharged. At any one time, this line can be fed only by one HSD sample tank or one of the two LCW sample tanks. GE states (June 29, 1990, submittal) that the total plant release per year will be limited by administrative controls to 0.1 Ci (excluding tritium). The staff estimates a total annual release of about 0.2 Ci for the liquid wastes (primarily due to 0.09 Ci of untreated detergent wastes and 0.1 Ci due to adjustment for anticipated operational occurrences such as operator error) and 59 Ci for tritium. Since administrative controls to limit the liquid wastes to 0.1 Ci/yr are within the scope of the referencing applicant, the staff will evaluate the acceptability of such specific administrative controls and liquid effluent source terms on a plant-specific basis for the ABWR referencing applicants.

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

In its evaluation of the liquid radioactive waste management system, the staff considered (1) the capability of the system to maintain releases below the limits in 10 CFR Part 20 during periods of fission-product leakage at design levels from the fuel, (2) the capability of the system to meet the processing demands of the station during anticipated operational occurrences, (3) the quality group and seismic design classification applied to the equipment, components, and structures housing the system, and (4) the design features that are incorporated to control the release of radioactive materials in accordance with GDC 60.

11.2.2 Evaluation and Findings

On the basis of its review of the ABWR SSAR section up to and including Amendment No. 15 and GE submittals dated June 7 and 29, 1990, in response to the staff's request for additional information (RAI), the staff concludes that the liquid radwaste system includes the equipment necessary to control the releases of radioactive materials in liquid effluents in accordance with GDCs 60 and 61 of Appendix A to 10 CFR Part 50 except as noted in this section. The staff concludes that the design of the liquid waste management system is acceptable and meets the requirements of 10 CFR 20.106, and GDCs 60 and 61, as referenced in the Standard Review Plan, for BWRs that discharge liquid wastes from the HSD sample tank or the LCW sample tank to the discharge canal having at least a dilution flow of 1500 gpm as stated in the ABWR SSAR Section 11.2.

The staff considers that demonstration of compliance with 10 CFR Part 50 (Appendix I), is within the scope of the referencing applicant and consequently the staff will evaluate this compliance individually for each license application. However, the staff has evaluated the ABWR design to determine whether there is reasonable assurance that any individual applicant referencing the ABWR in a license application will be capable of meeting the 10 CFR Part 50, Appendix I dose guidelines for radioactive materials released through liquid effluents. On the basis of its evaluation, the staff concludes that there is reasonable assurance that BWRs based on the ABWR design will meet the 10 CFR Part 50 (Appendix I) dose guidelines for radioactive materials released through liquid effluents for sites that satisfy the criteria stated in the ABWR SSAR with regard to the minimum discharge canal flow rate (1500 gpm) and the additional dilution credit of at least a factor of 5 between the point of release and the region in the unrestricted area where the water is used. For these reasons, the staff concludes that the design of the liquid waste management system is acceptable except as noted in this section.

The conclusions stated above are based on the following findings:

(1) On the basis of ABWR parameters that govern reactor coolant system concentrations of radionuclides and design of the liquid radwaste treatment systems, as stated in Section 11.2.1 of this report, the staff estimates that the total of radioactive wastes discharged via the liquid effluent during plant operation including anticipated operational occurrences will be no more than 0.2 Ci/yr (excluding tritium) and 59 Ci/yr for tritium. This finding, in conjunction with a minimum dilution flow rate of 1500 gpm and at least an additional credit of a factor of 5 between the point of discharge and the region in the unrestricted area where water is used, provides reasonable assurance that the ABWR liquid waste management system will meet the applicable Appendix I dose guidelines for liquid effluents.

(2) GE meets the requirements of 10 CFR Part 20 for ABWRs that have the minimum discharge canal flow rate of 1500 gpm and for which liquid waste can only be discharged from either the HSD sample tank or the LCW sample tank. The staff has considered the potential consequences resulting from ..actor

operation and has determined that the concentrations of radioactive materials in liquid effluent averaged over a year as permitted by 10 CFR 20.106 for the above case will be well below the limits in 10 CFR Part 20, Appendix B, Table II, Column 2. The staff has further determined that the instantaneous discharge concentrations of the radionuclides in liquid effluents to an unrestricted area will also be within these limits since GE has indicated by a submittal dated November 5, 1990, that the discharge rate via the single discharge line will be administratively controlled to conform to these limits. The staff will review such administrative controls on a plant-specific basis for the referencing applicants.

(3) GE has met the requirements c GDCs 60 and 61 with respect to controlling releases of radioactive materials to the environment. The staff has considered the capabilities of the proposed liquid radwaste treatment system to meet the demands of the plant resulting from anticipated operational occurrences and has concluded that the system capacity and design flexibility are adequate to meet the anticipated needs of the plant. It has reviewed GE's quality group classifications used for the system components and the seismic design applied to structures housing these systems (the quality assurance provisions of the liquid radwaste systems will be reviewed individually for each license application referencing the ABWR). The design of the systems and structures housing these systems meets the applicable criteria given in RG 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures, Components Installed in Light-Water-Cooled Nuclear Power Plants," Revision 1. The staff has reviewed the provisions

incorporated in GE's design to control the release of radioactive materials in liquids resulting from inadvertent tank overflow and concludes that the measures proposed by GE are consistent with the criteria given in RG 1.143 except for the lack of a local alarm capability for the condensate storage tank. This capability should be added to the design.

11.3 Gaseous Waste Management System

11.3.1 System Description and Review Discussion

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

The principal sources of gaseous wastes in the plant are the effluents from the offgas system, condenser mechanical vacuum pump, turbine gland seal system, and ventilation exhausts from the radwaste building, containment purge, reactor building, drywell, and turbine building. All these effluents are dumped into the plant stack either directly or indirectly (turbine gland seal system and mechanical vacuum pump) and monitored continuously. The gaseous effluent source terms identified above are described in detail in ABWR SSAR Sections 9.4, 10.4, and 11.4.

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

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

Fission and activation gases are held up for decay in the charcoal adsorber system downstream from the offgas condensers. Before entering the delay beds, these gases decay for 2.5 minutes during their transit from the main condenser to the delay beds. The charcoal adsorber beds consist of one guard bed adsorber followed by four parallel trains of two adsorber beds in series. The total mass of charcoal is 250,000 lb (GE submittal dated June 29, 1990).

Before discharge, the effluent passes through a HEPA filter assembly which removes particulates from the offgas system effluent stream. The holdup times in the ambient offgas treatment system charcoal beds (100°F, Dew Point 65°F) were calculated according to NUREG-0016, Revision 1 methodology and GE proprietary report NEDO-10751, "Experimental and Operational Confirmation of Offgas System Design Parameters," January 1973. The staff estimates these to be approximately 30 days for xenon, and 44 hours for krypton, and 18 hours for argon (GE's estimate: 42 days for xenon, 46 hours for krypton). Also, these beds remove iodines from the treatment system effluent by adsorption. The offgas system is designed to withstand a hydrogen explosion.

The ventilation exhausts from all plant areas such as the reactor building, which includes the primary containment (when it is vented or purged), fuel handling area, the area housing the emergency core cooling system (ECCS) equipment, other areas (e.g., standby gas treatment system (SGTS) rooms, fuel pool cooling system equipment rooms, areas housing nonessential equipment), the radwaste building, and the turbine building, are dumped into the plant vent and monitored continuously before their release to the environs. The reactor building areas mentioned above are serviced either directly or indirectly (for the primary containment purging or venting) by the secondary containment ventilation system during normal plant operation. However, ventilation exhausts from reactor building areas serviced by the essential electrical equipment heating, ventilation, and air conditioning (HVAC) subsystem, essential diesel generator HVAC subsystem, and reactor internal pump control panel HVAC subsystem; turbine building battery room exhaust subsystem, switchgear and air compressor rooms exhaust subsystem, and lube oil area exhaust subsystem; and service building exhaust air system are discharged directly to the environs unmonitored. Although ABWR SSAR Section 9.4.8.1.2 indicates that the service building ventilation exhaust will be monitored before its release. SSAR Table 11.5.1 does not indicate any monitoring provision for the subject exhaust. Therefore, the staff concludes that this discharge will be unmonitored. GE has not justified why these exhausts are considered as nonradioactive and, therefore, released to the environs unmonitored. The staff requires GE to address this open issue.

From ABWR SSAR Section 9.4 and GE's responses dated May 23 and August 22, 1990, to the staff's request for additional information relating to power conversion systems and building ventilation systems, the staff finds that neither the mechanical vacuum pump exhaust nor any building normal ventilation exhaust system includes charcoal adsorbers and/or HEPA filters to remove elemental and organic forms of iodines and particulates from the applicable effluent stream. Therefore, the staff has assumed that the mechanical vacuum pump exhaust and all building ventilation system exhausts are discharged to the environs untreated. The calculated release values for iodines and particulates in Table 11.2 of this report reflect the above assumption. ABWR SSAR Section 9.4.5.1.3, however, states that on detection of high radiation in the secondary containment exhaust or in the refueling floor atmosphere, the secondary containment normal ventilation system is secured and the exhaust is discharged through the safety-related standby gas treatment system comprised of charcoal adsorbers and HEPA filters. The

SGTS exhaust also goes through the plant vent which is monitored before release (as stated above).

The plant stack and the major streams feeding the plant stack (offgas system and building ventilation systems mentioned above) are monitored to facilitate appropriate corrective action in a timely manner to prevent offsite release exceeding applicable limits. Additionally, the offgas treatment system includes an automatic control feature to terminate the post-treatment release if it exceeds a preset radiation level in the effluent. However, it is not clear whether the monitors provided for monitoring the secondary containment exhaust will be sufficiently sensitive to detect a high radiation level in the primary containment purge exhaust as required by Branch Technical Position (BTP) CSB 6-4, "Containment Purging During Normal Plant Operations." Further, while ABWR SSAR Section 9.4.4.2.1.1 states that the turbine building ventilation exhaust will be monitored before its discharge to the monitored plant vent, SSAR Table 11.5-2 does not indicate any such process monitoring provision for the subject exhaust. The staff requires GE to address these concerns.

As stated above, all the airborne radioactivity releases will be through the plant vent. The plant vent is located on the reactor building, at 76 meters above grade, and is the tallest point on the site.

In its evaluation of the gaseous radwaste management system, the staff used the SRP criteria pertaining to (1) the capability of the system to maintain releases below the limits in 10 CFR Part 20 during periods of fission-product leakage at design levels from the fuel, (2) the capability of the system to meet the processing demands of the station during anticipated operational occurrences, (3) the quality group and seismic design classification applied to the equipment and to components and structures housing the system, (4) the design features that are incorporated to control the releases of radioactive materials in accordance with GDC 60, and (5) the potential for gaseous releases resulting from hydrogen explosion in the gaseous radwaste system. The staff has also reviewed the capability of the offgas system to limit the dose to 500 mrem due to individual exposure for 2 hours at the nearest exclusion area boundary resulting from radioactive releases from offgas system leak or failure as stated in BTP ETSB 11-5, Revision 0, July 1981. For its evaluation, the staff reviewed all the applicable information provided in the ABWR SSAR up to and including Amendment 15, as well as GE's submittals dated June 29, September 14, and October 26, 1990.

11.3.2 Evaluation and Findings

The staff concludes that the gaseous radioactive waste management system design for the ABWR meets the applicable requirements of 10 CFR 20.106 and GDCs 3, 60, and 61 with respect to radioactivity in gaseous effluents released to unrestricted areas, fire protection, control of releases of radioactive materials and radioactivity control in gaseous waste management system and ventilation system associated with fuel storage and handling areas, except as noted in this section.

Regarding the system design compliance with 10 CFR Part 50, (Appendix I) numerical guidelines for offsite radiation doses due to gaseous or airborne radioactive effluents during normal plant operation including anticipated operational occurrences, the staff considers that demonstration of such compliance lies within the scope of the referencing applicant. Therefore, the staff will evaluate the compliance on a plant-specific basis for the referencing applicant. However, the staff has evaluated the ABWR design to determine whether there is reasonable assurance that a referencing applicant will be capable of meeting the Appendix I dose guidelines for design objectives.

The ingestion, inhalation, and external irradiation due to ground contamination pathway doses to applicable organs resulting from release of radioactive iodines, radioactive material in particulate form, tritium and carbon-14 via airborne effluents depend upon a number of site-dependent parameters. Also, the population exposures (person-rem) and associated cost-benefit analysis are site dependent. Therefore, the staff's evaluation is limited to determination of the standard design's compliance for the gaseous waste management system with Appendix I guidelines for external doses to any individual in an unrestricted area due to noble gas radionuclides in gaseous effluents. On the basis of its evaluation, the staff concludes that there is reasonable assurance that ABWRs at sites that have an atmospheric dispersion factor ($^{X}/_{Q}$) equal to or less than 9.8×10^{-6} sec/meter will meet the above dose guidelines (5 millirem to the total body).

The staff also concludes that the ABWR offgas system design complies with BTP ETSB 11-5, "Postulated Radioactive Releases Due to a Waste Gas System Leak or Failure," for all sites that have a 0-2-hour $^{X}/_{Q}$ of equal to or less than 8.9×10^{-4} sec/meter at the nearest exclusion area boundary. The limiting $^{X}/_{Q}$ value given above corresponds to a total body exposure of 500 millirem and thus meets the acceptance criterion for dose at the nearest exclusion boundary specified in the BTP. The conclusions referred to above are based on the following findings:

(1) Subject to resolution of staff's concerns identified in Section 11.3.1 of this report relating to unmonitored plant releases directly to the environs and process monitoring provisions for the turbine building and primary containment purge/vent exhausts, the ABWR meets the requirements of GDCs 60 and 61 by ensuring that the design of the gaseous waste management system includes the equipment and instruments necessary to detect and control the release of radioactive materials in gaseous effluents.

(2) The staff has determined the releases of radioactive materials (noble gases, iodines, particulates, tritium and carbon-14) in gaseous effluents resulting from normal operation including anticipated operational occurrences, based on expected radwaste inputs over the life of the plant. It has used the calculated releases for noble gases (Table 11.2 of this report) to determine the bounding value for $^{X}/_{Q}$. In such a determination, the staff has assumed a 4-minute decay of the noble gas radionuclides during transit from the release point to the unrestricted area and has used the dose models and values for parameters given in RG 1.109, "Calculation of Annual Doses to Man From Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance With 10 CFR Part 50, Appendix I." For calculating the external dose due to noble gas radionuclides, the staff assumed a semi-infinite cloud model for the gaseous effluents. For the bounding $^{X}/_{Q}$ value quoted above, the staff calculated a total body dose (the limiting external

dose) of 5 millirem/yr which meets the applicable Appendix I dose guideline.

(3) ABWR meets the requirements of 10 CFR Part 20 because the staff has considered the potential consequences resulting from reactor operation with "a fission product release rate consistent with an offgas noble gas release rate of 100 Ci/MWt-sec at 30 minutes' decay" for a BWR and determined that under these conditions, the concentration of radionuclides in gaseous effluents in unrestricted areas with a $^{X}/_{Q}$ that is equal to or less than 9.8×10^{-6} sec/meter will be well below the concentration limits specified in 10 CFR Part 20, Appendix B, Table II, Column 1.

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

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

(6) The staff has reviewed the provisions incorporated in ABWR design to control releases resulting from hydrogen explosions in the gaseous waste management system (ABWR SSAR Section 11.3; GE submittal dated June 29, 1990). On the

basis of its review, the staff concludes that the features built into the design are adequate to prevent the occurrence of an explosion or to withstand the effects of an explosion in accordance with GDC 3.

On the basis of the above, the staff finds that the gaseous waste management system for the ABWR meets the acceptance criteria of SRP Section 11.3 and is, therefore, acceptable, subject to resolution of all the concerns identified above.

11.4 Solid Waste Management System

11.4.1 System Description and Review Discussion

The solid radioactive waste management system consists of the equipment and instrumentation necessary for the collection, solidification, incineration, packaging, and storing of radioactive wastes resulting from the operation of the reactor water cleanup system. the fuel pool cooling and cleanup system, the suppression pool cleanup system, the condensate polishing system, the liquid radwaste system, the building ventilation systems, the standby gas treatment system, the offgas system and miscellaneous solid wastes (e.g., paper, rags, contaminated clothing, gloves, shoe coverings) arising from the operation and maintenance of the plant. The solid radwaste management system is located in the radwaste building.

The ABWR solid waste system is designed to process two general types of solid wastes: "wet" solid wastes, which require solidification before being shipped off site, and "dry" solid wastes which require incineration, compaction, and/or packaging before being shipped.

Combustible dry wastes (e.g., rags, uniforms, paper) are burned in an incinerator and discharged to an ash storage drum. The offgas from the incinerator is passed through two ceramic filters in series and a HEPA filter before it is released into the atmosphere through the main plant vent. On the basis of GE submittal dated June 29, 1990, the staff estimates this release to be 0.016 Ci/yr and to be in particulate form. Incinerated ash is discharged to an ash storage drum by ash discharge equipment located on the bottom of the incinerator. Noncombustible dry solid wastes are compacted and placed in dry active waste (DAW) drums for shipment.

Wet wastes are of two forms: (1) slurries associated with spent resins and sludges associated with filter and backwash of filter demineralizer and (2) concentrated wastes from the HCW concentrators of the liquid radwaste treatment system. As stated in Section 11.2.1 of this report, the spent resins and the sludges are dewatered and the resulting slurry is loaded in high integrity containers (HICs) for eventual shipment.

The concentrated waste from the HCW concentrators is routed through a thin-film dryer which dewaters the concentrate. The water from this operation is routed back to the HCW collector tanks. Air is exhausted through the radwaste building HVAC exhaust. The dewatered, powdered waste is pelletized, and the pellets are mixed with a solidifying agent (cement glass) in drums for eventual offsite shipping. Air from the pelletizing and solidification process is routed to the radwaste building HVAC exhaust via a particle filter and a HEPA filter. Interface requirement 11.4.3.1 states that the first applicant referencing the ABWR design should provide detailed information to demonstrate that the wet waste solidification process will result in a product that complies with 10 CFR Part 61, Section 61.56. This should be modified to indicate that all applicants are required to provide this information.

For the ABWR, based on the Electric Power Institute report EPRI-NP-5528, Volume 1 (February 1988) "Radwaste Generation Survey Update - Boiling Water Reactors" and NUREG/CR-2907 "Radioactive Materials Released from Nuclear Power Plants," Annual Reports for 1986 and 1987 (Volumes 7 and 8 - only BWRs were considered), the staff estimates the processed wet wastes requiring shipment to be about 13,000 cubic feet per year containing approximately 1200 Ci. The spent resin and filter and filter/demineralizer sludge slurries will be stored in high-integrity containers before shipment. The solidified concentrates will be stored in 55-gallon drums before shipment. Based on the cited NUREG report and GE's submittal dated June 29, 1990, the staff estimates the processed drywastes requiring shipment to be about 12,000 cubic feet per year containing approximately 12 Ci. However, with incineration of combustible dry wastes, the shipment volume will be less than 12,000 cubic feet. The processed dry wastes will be stored in boxes or in 55-gallon drums.

In its evaluation of the solid radioactive waste management system, the staff considered (1) system design objectives in terms of expected types, volumes, and activities of wastes processed for offsite shipment, (2) provisions for onsite storage of processed solid wastes before shipment, (3) procedures for disposal of incinerated waste, (4) system design to meet acceptance criteria of SRP Section 11.4, and (5) piping and instrumentation diagrams for the system. Since the establishment and implementation of a process control program (PCP) for solidifying the evaporator concentrates using an approved solidification agent, and processing of the spent resins and filter sludges by dewatering are outside the scope of the ABWR, the staff will review the PCP and the dewatering process for each referencing applicant, separately.

11.4.2 Evaluation Findings

On the basis of its review of ABWR SSAR Section 11.4 and GE submittals dated June 7, June 29, October 26, and November 5, 1990, the staff concludes that the solid waste management system design meets the requirements of 10 CFR 20.106, 10 CFR 20.302(a), 10 CFR Part 50.34(a), 10 CFR 61.56, 10 CFR Part 71 and GDCs 60, 63, and 64, except as noted below. This conclusion is based on the following findings:

(1) The design includes equipment and instrumentation for processing, packaging and storage of radioactive solid wastes before shipment off site, except as discussed below. Dedicated radwaste storage rooms in the radwaste building accommodate storage of 385 fifty-five gallon drums and 15 boxes corresponding to storage of approximately 4000 cubic feet of processed wet and dry solid wastes. Further the GE submittal dated October 26, 1990, indicates that, if needed, additional space in the rooms can easily be provided to accommodate storage of more processed wastes. The staff is, however, concerned that GE has not specified the capacity and the maximum number of HICs which will be temporarily stored and shielded in the truck area. It may be noted that since this waste is Type B per 10 CFR 61.55 classification and is not intended to be processed to a stable form, it has

to be placed in HICs (ABWR SSAR Figure 11.2-29 shows that this waste will be put in the HICs) to comply with 10 CFR 61.56 and 10 CFR Part 71 requirements. Subject to resolution of this concern, the available storage space for the dry wastes (one full offsite waste shipment) and the processed wet wastes (30 days' waste generation at normal generation rate) meet the applicable criteria for storage of the wastes as identified in BTP ETSB 11-3, "Design Guidance for Solid Radioactive Waste Management Systems Installed in Light-Water-Cooled Nuclear Power Plants," Positions B.III.2 and 3. Also, the capacities of tanks accumulating spent resins and filter sludges meet the Position B.II.1 of the BTP.

(2) The system has the capability to process the types and volumes of wastes expected during normal plant operation including anticipated operational occurrences in accordance with GDC 60 and has provisions for the handling of wastes relative to the requirements of 10 CFR Part 20, 10 CFR Part 71 and applicable Department of Transportation regulations as they relate to the scope of the ABWR. Specifically, the staff has determined that the offgases (resulting from incineration of combustible wastes) exhausted through ceramic filters and a HEPA filter to the monitored plant vent will have minimal effect on demonstration of compliance with 10 CFR 20.106(b) relating to concentrations of radionuclides in gaseous or airborne effluents in unrestricted areas for sites with a $^{X}/_{Q}$ equal to or less than 9.8×10^{-6} sec/meter. As stated in Section 11.3.2, Item 3 of this report, the staff has concluded that the subject concentrations will be well below the applicable regulatory limit. By identifying the type, the expected

quantity, the Curie content, and the manner of disposal of the combustible wastes that will be incinerated, GE complies with 10 CFR 20.302 as it relates to the ABWR scope.

(3) The system complies with 10 CFR 50.34(a) and GDCs 63 and 64 as they relate to design provisions for monitoring radiation levels and leakage. Specifically, besides the effluents resulting from the system inputs to the liquid radwaste and gaseous radwaste management systems being monitored by the respective monitors for the liquid and gaseous waste management systems (in conjunction with other effluents from the systems), radiation monitors are placed at the end of a drum conveyor to monitor the radiation resulting from mixture in the drums and surface contamination of the drums. Devices such as position switches, weight elements and level sensors are used to prevent spillage while filling and pouring (solidification of vapor concentrates) and overfilling the drums. Additionally, safety interlocks provided for the solidification process system ensure that solidification is performed only when certain conditions are met (these conditions are identified in ABWR SSAR Section 11.4.2.3.4).

(4) ABWR SSAR Section 11.4.1.2 and GE submittals dated June 7 and 29, 1990, indicate that the quality group classification, seismic design, and other design features (e.g., heat tracing of concentrate pipings and tanks, flushing connections for all components and piping which contain slurries) meet the guidelines of RG 1.143 and BTP ETSB 11-3, Position B.V.

(5) With regard to the dewatering method for spent resin and

filter sludges, namely, treatment by a thin-film dryer or by a vendor-supplied mobile dewatering system, the staff finds the proposed method acceptable. However, as stated in Section 11.4.1 of this report, the staff will review the details demonstrating compliance of the dewatering process and solidification process with applicable positions of BTP ETSB 11-3 on a plant-specific basis for each referencing applicant. With regard to solidification of evaporator concentrate waste, to date, the staff has taken no position on the acceptability of the proposed cement glass as a solidifying agent (i.e., there has been no NRC approved topical report for cement glass as the solidifying agent). Therefore, contingent upon resolution of the staff concern identified in Section 11.4.1 of this report, the system design complies with 10 CFR 61.56 requirements as they relate to the solidification process and the dewatering process within the ABWR scope.

On the basis of these findings, the staff concludes that the solid radwaste management system design for the ABWR meets the acceptance criteria of SRP Section 11.4 and complies with the applicable requirements of 10 CFR 61.56 and is, therefore, acceptable, subject to resolution of the staff's concerns identified above and the following concerns relating to the information provided so far on the system:

(1) GE should identify the specific fire protection features available in the applicable area to prevent any undue fire hazard resulting from incineration.

(2) ABWR SSAR Section 11.4.2.3.5 (last paragraph), Table 11.4-2, and response to Question 430.171 are inconsistent and

confusing. For example, the section states that Table 11.4-2 represents the shipped volume of solid wastes and that Table 11.4-3 gives the corresponding Curie content; however, these two tables cannot be correlated since Table 11.4-2 gives the shipped volume only for the solidified concentrates (not the total volume of all solid wastes) and Table 11.4-3 gives Curie content for the spent resin and filters sludges. Therefore, the staff requires GE to correct the section and Table 11.4-2 as appropriate. Specifically, the staff requires GE to provide in the ABWR SSAR, the volumes of various kinds of solid wastes expected to be shipped annually and their corresponding total Curie content (for the processed wet wastes by radionuclides).

11.5 Process and Effluent Radiological Monitoring and Sampling Systems

11.5.1 System Description and Review Discussion

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

The GE submittal dated June 2, 1989, states that all airborne radioactive releases from the plant to the environment are exhausted through the plant vent. The major sources which are combined and routed to the plant vent are the offgas exhaust, the radwaste building exhaust, the radwaste building control room and unit substation exhaust, the reactor building (secondary containment) exhaust, and the turbine building exhaust, which includes the gland seal system and the mechanical vacuum pump exhausts. A radiation-monitoring system monitors the plant vent discharge for gross radiation level and collects halogen and particulate samples.

In addition to the main plant vent gaseous effluent monitor and samplers, as indicated in ABWR SSAR Table 11.5-1, radiation monitors are provided for monitoring the offgas post-treatment exhaust, the reactor building (secondary containment) exhaust, the radwaste building vent exhaust, and the radwaste building control room and unit substation exhaust. These monitors are used to identify sources of airborne activity before mixing in the main plant vent. Gaseous process stream monitoring includes the offgas pretreatment radiation-monitoring system (RMS), and the carbon bed vault RMS and control rod drive (CRD) maintenance area exhaust RMS.

The liquid effluent and process monitoring systems include the liquid radwaste effluent and the reactor building closed cooling water system RMS.

The RMSs which monitor the discharges from the gaseous and liquid radwaste treatment systems (i.e., offgas post treatment effluent, processed liquid radwaste effluent) have provisions to alarm and initiate automatic closure of the waste discharge valve of the affected treatment system before exceeding the normal operation limits to be specified in radiological technical specifications. Before being discharged from the radwaste treatment systems, liquid in the tanks is sampled and analyzed. Release and dilution rates are specified on the basis of the results of these analyses.

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

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

The ABWR design includes accident monitoring instrumentation for monitoring noble gases, iodines, and particulates in gaseous or airborne effluent streams during an accident. As stated in ABWR SSAR Sections 1.A.2.15 and Section 7.5, GE considers that such instrumentation provided for the ABWR is generally in accordance with RG 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants To Assess Plant and Environs Conditions During and Following an Accident," Revision 3, and therefore, meets the guidelines of NUREG-0737, "Clarification of TMI Action Plan Requirements," Item II.F.1, "Additional Accident-Monitoring

Instrumentation", Attachments 1, "Noble Gas Effluent Monitor," and 2, "Sampling and Analysis of Plant Effluents."

The staff has reviewed the ABWR SSAR up to and including Amendment 11 and GE submittals dated June 2, 1989, and February 28, 1990, with regard to the process and effluent radiological monitoring and sampling systems for the ABWR. The review included piping and instrumentation diagrams for the liquid and gaseous radwaste systems, SSAR Tables 11.5-1 through 11.5-7 which list the liquid and gaseous process and effluent radiation-monitoring systems and summaries of radiological analysis for liquid and gaseous process and effluent stream samples, information provided in SSAR Section 7.5.1.1 and Tables 7.5-1 and 7.5-2 (i.e., tables comparing ABWR design provisions for monitoring radioactive gaseous effluents during an accident with applicable RG 1.97 guidelines), and descriptions of the various building ventilation systems, the main condenser evacuation system, and the turbine gland sealing system insofar as they relate to the radiation-monitoring provisions for these systems. However, the staff could not review the location of the monitoring points relative to the effluent release points in the gaseous effluent streams, since GE has not provided the corresponding P&IDs. As stated in Section 11.0 of this report, the staff will review specific compliance of sampling program and quality assurance for radiological monitoring programs with ANSI Standard. N13.1 and RGs 1.21 and 4.15 on a plant-specific basis for the referencing applicants. Therefore, the staff's review of the sampling and analysis program for the ABWR standard design was limited to the identification of the streams required to be sampled.

11.5.2 Evaluation and Findings

On the basis of its review, the staff concludes that the design of the ABWR process and effluent radiological monitoring and sampling systems complies with 10 CFR 20.106 and GDCs 60, 63, and 64, except as noted below. The staff's conclusion is based on the following findings:

(1) The design includes monitoring the radioactivity of effluents to unrestricted areas except as noted below. The service building ventilation system exhaust and the exhausts from certain areas of the reactor building (serviced by the essential electrical equipment, essential diesel generators and reactor internal pump control panel HVAC subsystems), and the turbine building (serviced by the battery room, switchgear, and air compressor rooms, and the lube oil area exhaust subsystems) are directly released to the environs unmonitored. GE has not provided sufficient information in its submittal dated April 26, 1991 to clarify that the service building ventilation system exhaust is routed through the plant vent where radiation monitoring occurs.

(2) The design includes provisions for monitoring process streams (e.g., offgas post-treatment exhaust, secondary containment exhaust, reactor building cooling water system) and provisions for initiating appropriate action in case of postulated accidents. Automatic control features include automatic termination of liquid effluent release or the offgas system release as appropriate when the preset radiation level for the applicable stream is exceeded. The automatic control features also include securing the normal

secondary containment ventilation system and initiation of SGTS under certain conditions identified in the SSAR. Also, since there is a single monitor for the gland seal exhausts and normally totally clean steam is not supplied for sealing the turbine gland seals, the staff requires that plant procedures include manual switchover to the backup clean steam source whenever the monitor indicates that the exhaust stream concentration exceeds a preset level. Besides these, the staff is also concerned about whether the secondary containment ventilation exhaust monitor is sufficiently sensitive to detect high radiation level in the primary containment purge exhaust and trigger automatic initiation of appropriate corrective action.

(3) As stated in Section 11.5.1 of this report, the design includes provisions for sampling and analysis of radioiodines, particulates, and tritium in the process and effluent streams (for tritium only in the effluent stream). The staff is, however, concerned that SSAR Table 11.5-5 does not include grab sampling and analysis provisions for the gland seal process stream.

(4) ABWR SSAR Tables 7.5-1 and 7.5-2 provide design and qualification criteria for accident-monitoring instrumentation and the concentration ranges covered by the instrumentation. The staff finds that the ABWR design complies with RG 1.97 with regard to ranges and design and qualification criteria. However, neither these tables nor SSAR Section 11.5 contain sufficient information. Specifically, GE has not provided the following information:

(a) type of instrumentation to be used, their calibration frequency and technique

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

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

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

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

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

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

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

On the basis of these findings, the staff concludes that the design of the ABWR process and effluent radiological monitoring and sampling systems meets the acceptance criteria of SRP Section 11.5 and is, therefore, acceptable, subject to resolution of all the staff's concerns identified above.

Table 11.1
Principal Parameters Used in the Calculation
Of Gaseous and Liquid Effluents from ABWR

Parameter	Value
Thermal power (MWt)	3926
Total steam flow rate (10^6 lb/hr)	16.8
Mass of water, reactor coolant system (10^6 lb)	0.674
Steam/water concentration, reactor vessel	
Halogens	0.015
Particulates	0.001
RWC demineralizer flow rate (10^6 lb/hr)	0.334
Fraction of FW through condensate demineralizer	0.67
Reactor building iodine release fraction	1.0
Reactor building particulate release fraction	1.0
Radwaste building iodine release fraction	1.0
Radwaste building particulate release fraction	1.0
Turbine building iodine release fraction	1.0
Turbine building particulate release fraction	1.0
Mechanical vacuum pump iodine release fraction	1.0
Charcoal delay system	
Kr dynamic adsorption coefficient (cm^3/g)	16.0
Xe dynamic adsorption coefficient (cm^3/g)	260.0
Ar dynamic adsorption coefficient (cm^3/g)	6.4
Mass of charcoal (10^3 lb)	250.0
Detergent waste decontamination factor (DF)	1.0
Liquid waste inputs	
High purity (low conductivity) subsystem	
Waste collection rate (gpd)	15,200
Reactor coolant activity fraction	0.23
Collection, proces time (days)	5.98, 0.96
DFs for halogens and Cs and R	1,000, 100
DF for others*	1,000
Fraction discharged	0.01
Low purity (high conductivity) subsystem**	
Waste collection rate (gpd)	6300
Reactor coolant activity fraction	0.0028
Collection, process time (days)	1.52 0.67
DF for halogens	10,000
DF for others *	100,000
Fraction discharged	0.1
Detergent wastes	
DF for all radionuclides	1.0
Fraction discharged	1.0

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

Table 11.2
Calculated Releases of Radioactive Materials
in Gaseous Effluents from ABWR (Ci/yr-Unit)

Nuclide	Building Vents[1]	Gland Seal	Mechanical Vacuum Pump	Offgas System	Total
AR-41	1.5 E1	0.0	0.0	1.0	1.6 E1
KR-83M	0.0*	4.0	0.0	0.0	4.0
KR-85M	2.9 E1	7.0	0.0	8.5 E1	1.2 E2
KR-85	0.0	0.0	0.0	2.7 E2	2.7 E2
KR-87	6.3 E1	2.4 E1	0.0	0.0	8.7 E1
KR-88	9.5 E1	2.4 E1	0.0	6.0	1.3 E2
KR-89	6.1 E2	1.3 E2	0.0	0.0	7.4 E2
XE-131M	0.0	0.0	0.0	3.6 E1	3.6 E1
XE-133M	0.0	0.0	0.0	0.0	0.0
XE-133	4.8 E2	9.0	1.3 E3	2.3 E3	4.1 E3
XE-135M	[illegible].9 E2	2.9 E1	0.0	0.0	1.0 E3
XE-135	7.4 E2	2.6 E1	5.2 E2	0.0	1.3 E3
XE-137	1.3 E3	1.5 E2	0.0	0.0	1.4 E3
XE-138	1.0 E3	9.5 E1	0.0	0.0	1.1 E3
I-131	1.6 E-1	1.6 E-3	8.8 E-2	0.0	2.5 E-1
I-133	2.3	5.8 E-3	9.7 E-1	0.0	3.3
C-14	0.0	0.0	0.0	9.5	9.5
H-3	5.9 E1	0.0	0.0	0.0	5.9 E1
CR-51	2.7 E-3	0.0	1.0 E-6	0.0	2.7 E-3
MN-54	6.0 E-3	0.0	0.0	0.0	6.0 E-3
CO-58	1.5 E-3	0.0	0.0	0.0	1.5 E-3
FE-59	7.9 E-4	0.0	0.0	0.0	7.9 E-4
CO-60	1.3 E-2	0.0	5.6 E-7	0.0	1.3 E-2
ZN-65	1.1 E-2	0.0	3.4 E-7	0.0	1.1 E-2
SR-89	6.1 E-3	0.0	0.0	0.0	6.1 E-3
SR-90	3.0 E-5	0.0	0.0	0.0	3.0 E-5
NB-95	1.0 E-2	0.0	0.0	0.0	1.0 E-2
ZR-95	1.8 E-3	0.0	0.0	0.0	1.8 E-3
MO-99	6.8 E-2	0.0	0.0	0.0	6.8 E-2
RU-103	4.3 E-3	0.0	0.0	0.0	4.3 E-3
AG-110M	2.4 E-6	0.0	0.0	0.0	2.4 E[illegible]
SB-124	2.2 E-4	0.0	0.0	0.0	2.2 [illegible]
CS-134	7.3 E-3	0.0	3.2 E-6	0.0	7.3 E-3
CS-136	6.0 E-4	0.0	1.9 E-6	0.0	6.0 E-4
CS-137	1.1 E-2	0.0	8.9 E-6	0.0	1.1 E-2
BA-140	3.2 E-2	0.0	1.1 E-5	0.0	3.2 E-2
CE-141	1.1 E-2	0.0	0.0	0.0	1.1 E-2

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

[1] Does not include the HEPA filtered offgases resulting from incineration of certain types of drywastes. The total release is expected to be 0.016 Ci/yr in particulate form.

Table 11.3
Design Capacities of Principal Components
in the Liquid and Gaseous Radwaste Treatment
Systems for ABWR Single Unit

Component	Number	Capacity and/ or flow rate
Liquid Systems*		
High purity (low conductivity) subsystem		
Low conductivity (collection) tank	2	114,000 gal
Waste hollow fiber filter	2	66 gpm
Waste demineralizer (mixed bed)	2	130,160 gpm
Sample tank**	2	114,000 gal
Low purity (high conductivity) subsystem		
High conductivity (collection) tank	2	12,000
Waste evaporator	2	13 gpm
Distillate demineralizer	1	26 gpm
Sample tank**	2	114,000
Distillate tank	1	4200 gal
Detergent waste subsystem		
Hot shower drain receiver	1	8700 gal
Hot shower drain sample tank	2	55,500 gal
Detergent filter	2	26 gpm
Gaseous Systems***		
Ambient temperature RECHAR system		
Catalytic recombiner	2	350 psig design pressure, 450°F design temp. temperature
Condenser	1	350 psig, tube side design pres. 900°F design temperature
Charcoal adsorber beds	9	Carbon steel; 350 psig design pressure; 40°F to 250°F design temperature; 125 tons of activated charcoal

* Quality group and seismic design (Regulatory Guide 1.143).
** Shared by high and low conductivity subsystems. Acts as a surge tank for both systems when condensat storage of the processed liquids is unavailable. In addition, serves as a sample tank for the high conductivity or low conductivity subsystem from where discharge to the environment can occur.
*** Quality group in accordance with Regulatory Guide 1.143.

13 CONDUCT OF OPERATIONS

13.1 Organizational Structure of Applicant

This is out of the ABWR standard plant scope.

13.2 Training

This section will be included in a supplement to this report.

13.3 Emergency Planning

This section will be included in a supplement to this report.

13.4 Review and Audit

This is out of the ABWR standard plant scope.

13.5 Plant Procedures

This section will be included in a supplement to this report.

13.6 Physical Security

13.6.1 Preliminary Planning

The NRC Standard Review Plan (SRP) NUREG-0800 Section 13.6 (Revision 2, July 1981) addresses preliminary planning for physical security as a subject to be reviewed at the preliminary safety analysis review (PSAR) stage. The SRP indicates that preliminary planning is considered acceptable if it provides reasonable assurance that provisions required to be addressed in the applicant's security plan are expected to be achieved. The ABWR SSAR states that preliminary planning is not required. Since the security plan will be the responsibility of the future applicants who reference the ABWR standard design, the staff concurs.

13.6.2 Security Plan

The SSAR states that this is beyond the scope of the ABWR standard plant. The staff concurs. In addition to the interface requirements listed in Section 13.6.3 in the ABWR SSAR, the utility applicant must provide plant specific security, contingency, and guard training plans in accordance with 10 CFR 50.34, and 10 CFR Part 73.

Interface Requirement: Section 1.9 of Chapter 1 of the ABWR SSAR states that there are a number of interfaces between the ABWR standard plant design and the remainder of the plant that must be addressed by the parties that reference the ABWR design.

The staff has reviewed the interfaces and finds them to be acceptable subject to the addition of the requirement that the utility applicant provide plant specific security, contingency and guard training plans in accordance with 10 CFR 50.34 and Part 73.

13.6.3 Control of Access to Areas Containing Vital Equipment

13.6.3.1 Introduction

Section 13.6.3.1 of the SSAR is an introduction to 13.6.3 that says that 13.6.3 is concerned with the control of access to areas containing vital equipment. It states that discussion of the capability for detecting loss of operability of vital equipment, which helps protect against radiological sabotage, is included in ABWR SSAR Chapter 7. The staff considers the discussion contained in Appendix 19C to adequately address timely detection of tampering with vital equipment.

13.6.3.2 Design Bases

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

In response to staff comments, GE acceptably modified ABWR SSAR Appendix 19B, Sections 19B.2.4 and 19B.3, to clarify which version of the Electric Power Research Institute (EPRI) Advanced Light Water Reactor (ALWR) Requirements Document the ABWR security requirements follow.

The ALWR Requirements Document (Volume 2, Revision 1, Chapter 11, Section 8.4.1) specifically requires the protected area lighting to be powered from an uninterruptable power source. The ABWR response to NRC Question 910.18 identified a site security load on the non-Class 1E vital (uninterruptable) load list (Table 20B-1), but the staff considered the description of this interface to be insufficiently defined. In response to staff comments, GE added Section 19B.3.12, which clarified the interface between the security system uninterruptable power requirements (to be later so determined by the site-specific security system designer as to meet required security system performance) and the non-Class 1E vital power supply capacity.

Section 19B.3.12 allows the protected area lighting to be powered from an interruptible power source. Although a difference remains between EPRI's ALWR Requirements Document and the ABWR SSAR, the staff considers this revision resolves the need for clarification. The staff finds that the ABWR security power requirements are not in conflict with NRC requirements for security power.

NRC Information Notice 83-83 suggested that new plant designs that make extensive use of solid state devices in instrument and control circuits may experience reactor system malfunctions and spurious actuations due to keying of portable radios in their vicinity. In Questions 910.10 and 910.17, the staff asked that radio frequency interference design criteria be established to ensure that security personnel within the reactor and control building could maintain radio communication without adversely affecting plant operation. The ABWR response to Question 910.17 referenced discussions of system tolerance to electromagnetic interference (EMI). The staff considers the auxiliary systems interface criteria in Section 9.5.13.11 of Chapter 9 and the response to NRC Question 430.315 to adequately resolve staff's concern.

13.6.3.3 Vital Areas

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

Questions 910.9, 910.11 and 910.20 addressed the completeness of the list of vital equipment in Section 13.6.3.3. Additional clarification was provided following discussions with staff. The staff is satisfied that the list of vital equipment in Section 13.6.3.3 includes all active and passive plant equipment essential to safe shutdown of the reactor, including necessary support systems; the reactor vessel and the remainder of the reactor coolant system pressure boundary within primary containment; the suppression pool; spent fuel in the fuel pool; and any associated piping, equipment, and controls whose failure could result in an offsite release in excess of 10 CFR Part 100 limits. The staff finds this to be compatible with the NRC's Review Guideline 17. The staff approves the listing of these vital equipment as an interface.

Protection of the diesel fuel storage tank as vital equipment is not addressed in SSAR Section 13.6. The staff considers this to be acceptable since each diesel fuel day tank provides at least 8 hours of capacity for its diesel engine at full load. In addition, Chapter 9 of the ABWR SSAR states that each diesel generator fuel oil storage tank fill connection is capped and locked to prevent entry of moisture; each vent is goose-necked with fine mesh screen to prevent access.

The staff's review of the designation of equipment as vital in site-specific applications will focus on that plant support equipment which was outside the scope of the certified design. In addition, 10 CFR 73.55(e) requires the central alarm station to be considered a vital area, and secondary power supply systems for alarm annunciator equipment and non-portable communications equipment to be located in vital areas. The secondary alarm station is also typically on site and treated as a vital area. These are open items that will be resolved at the time of the site-specific security plan review.

The site-specific licensing review of the security and contingency response plan will also include an evaluation of whether the security response force capability to interdict the violent external assault postulated in 10 CFR 73.1(a)(1)(i) properly accounts for the minimum penetration delay provided by the vital area barriers and doors.

13.6.3.4 Methods of Access Control

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

SSAR Section 13.6.3.4 describes, in general terms, the types of door controls that would be used to control access to vital areas. In response to staff Questions 910.12, 910.21, and 910.22, statements were added that all doors and hatches connecting vital to non-vital areas are to be alarmed and emergency egress will not require keys or card readers. Section 13.6.3.5 describes the specific security measures at portals into the reactor and control buildings from exterior areas and facilities of the remainder of the plant.

The types of door controls specified in SSAR Section 13.6.3.4 are generally acceptable, but detail provided is insufficient to determine compatibility with RG 5.12, "General Use of Locks in the Protection and Control of Facilities and Special Nuclear Material," which is specified in the Standard Review Plan (SRP) (NUREG-0800, July 1981). This interface on access control methods also does not address the positive control requirement of 10 CFR 73.55(d)(7)(i)(B) and the record-keeping requirement of 10 CFR 73.70(d), which requires logging individuals' times of entry to and exit from each vital area. This does not need to be addressed in the ABWR SSAR but remains open to review during review of the site-specific security plan.

In Question 910.19, the staff asked why the environmental conditions parameters of ABWR SSAR Appendix 3I should not apply to the design and qualification of security access control components. TMI Action Item II.B.2 (NUREG-0737), "Design Review of Plant Shielding and Environmental Qualification of Equipment," identifies areas for which environmental qualification of equipment necessary to ensure post-accident access may need to be considered. Although the "security center" is not safety related, it is included in NUREG-0737 since access to it may be necessary to give access to the rest of the plant.

NRC Information Notice 86-106, Supplement 2, discussed an event at the Surry Power Station in which condensed steam saturated a security card reader and shorted out the card reader system for the entire plant. As a result, key cards would not open doors controlled by the security system. In the same event, the performance of a security communications system radio repeater was temporarily degraded as a result of a thick layer of ice formed on it from actuation of a carbon dioxide discharge nozzle. In response to Question 910.19, GE stated that (1) this equipment is not safety related and is not required to operate under accident conditions, (2) the card reader design is required to preclude the possibility of failure of one card reader affecting the operation of any other card reader, (3) card reader doors are required to have a key-operated override, and (4) emergency exits are required to be provided for exiting without using keys or card readers. The staff considers this response to be generally consistent with currently accepted industry practice.

Furthermore, the interface requirement of ABWR SSAR Chapter 19, Appendix 19B.3.10, requires an applicant to evaluate the impact of the security system on required operator actions during all emergency modes of operation. The staff suggests that this analysis include consideration of an emergency requiring evacuation of the control room in the control building to the remote shutdown panel in the reactor building. Evaluation of compliance with the vital equipment prompt access requirements of 10 CFR 73.55(d)(7)(ii) is an open item that can be resolved during review of the site specific security plan.

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

SSAR Section 13.6.3.6 discusses bullet-resistant walls and doors, and security grills and screens incorporated into the building design, with the stated intent of minimizing forcible access to the control room. Responses to staff questions (910.13, 910.23, 910.24) did not resolve staff uncertainty as to adequacy of barriers in all man-sized openings in physical barriers that separate other vital from non-vital areas. Also, the staff position on the effectiveness of the ventilation system barriers described in 13.6.3.6 remains as described in Question 910.13, that is consideration may need to be given to how accessible, isolated, and hidden from view these barriers will be, as well as whether they can be penetrated with hand tools available on site. While SSAR Section 13.6.3.6 only addresses the main control room heating, ventilation, and air conditioning (HVAC) ducting and exterior air exhaust systems, ABWR SSAR Chapter 19, Appendix 19B.2.4(13), includes ALWR requirements on utility port openings (e.g., HVAC, cooling, and piping) through all vital or protected area boundaries. With the change in Appendix 19B.2.4 that clarifies that the ABWR design will comply with the ALWR requirements as defined, the staff considers this issue to be satisfactorily resolved.

13.6.3.7 Compatibility With the Remainder of the Plant

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