UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

November 5, 1991

Docket No. STN 50-605

Mr. Patrick W. Marriott, Manager
Licensing & Consulting Services
GE Nuclear Energy
General Electric Company
175 Curtner Avenue
San Jose, California 95125

Dear Mr. Marriott:

SUBJECT: DRAFT SAFETY EVALUATION REPORT ON THE ADVANCED BOILING WATER REACTOR DESIGN

Enclosed is a copy of the draft safety evaluation report (DSER) relating to the review of your application for certification of the Advanced Boiling Water Reactor (ABWR) design. The DSER discusses the results of the review of General Electric's (GE) Standard Safety Analysis Report Chapters 1, 2, 3, 5, 6, 8, 9, 10, 12, 13, 14, and 15.

Chapters 1, 2, 3, 5 6, 9, 10, and 13 are additions to previously issued evaluations. Chapters 8, 12, 14, and 15 are newly issued chapters. As noted in Chapter 1, the DSER phase of the ABWR review is completed with the issuance of the enclosed document. The staff will continue working with GE to resolve the open issues which have been identified in all DSERs. Final resolution of all issues will be included in the staff's final safety evaluation report (FSER) for the ABWR.

Please review the DSER for proprietary information. If no feedback is provided within ten working days, we will assume that it includes no proprietary information.

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Copies of this report have been sent to the Advisory Committee on Reactor Safeguards for its review, and will be placed in the Public Document Room at the Gelman Building, 2120 L Street, N.W., Washington, D.C. 20555 following GE review for proprietary information. If you have any questions regarding this report, please contact Chet Poslusny at (301) 429-1132 or Vic McCree at (301) 492-1121.

Sincerely,

Original Signed By

Dennis M. Crutchfield, Director
Division of Advanced Reactors
and Special Projects
Office of Nuclear Reactor Regulation

Enclosure:
As stated

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Copies of this report have been sent to the Advisory Committee on Reactor Safeguards for its review, and will be placed in the Public Document Room at the Gelman Building, 2120 L Street, N.W., Washington, D.C. 20555 following GE review for proprietary information. If you have any questions regarding this report, please contact Chet Poslusny at (301) 429-1132 or Vic McCree at (301) 492-1121.

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Dennis M Crutchfield

Dennis M. Crutchfield, Director
Division of Advanced Reactors
and Special Projects
Office if Nuclear Reactor Regulation

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As stated

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Mr. Patrick W. Marriott
General Electric Company

Docket No. STN 50-605

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POLICY ISSUE
(Information)

October 31, 1991 SECY-91-355

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, 2, 3, 5, 6, 8, 9, 10, 12, 13, 14, AND 15 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's DSER addresses open items needing closure that have been identified by the staff's review of GE's Standard Safety Analysis Report (SSAR) Chapters 1, 2, 3, 5, 6, 8, 9, 10, 12, 13, 14, and 15.

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," the staff provided the previous DSER sections to the Commission and discussed the ABWR review process and the Commission's guidance that is being followed.

Discussion: The enclosed DSER addresses additions to the previously issued Chapters 1, 2, 3, 5, 6, 9, 10, 13, and addresses Chapters 8, 12, 14, and 15 in their entirety. As noted in Chapter 1 of the DSER, with the issuance of the enclosed document, the DSER phase of the ABWR review is essentially complete. The staff will continue working with GE to resolve DSER open items. The staff recently held meetings with GE and received additional information which covers many of the issues in the DSER. The

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D. Scaletti, NRR
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492-1121

NOTE: THIS SECY PAPER AND THE ATTACHED DSER WILL BE MADE PUBLICLY AVAILABLE IN 10 WORKING DAYS AND FOLLOWING A REVIEW BY GE TO ENSURE THAT NO PROPRIETARY MATERIAL HAS BEEN INCLUDED IN THE PSAR staff will document its review of this information and the resolution of open items in the final safety evaluation report. The staff will provide copies of this DSER to the Advisory Committee on Reactor Safeguards.

Conclusion: The enclosed DSER contains no new policy issues. The staff will forward the report to GE to inform it of the staff's current findings and request that they review the DSER for proprietary information. The staff plans to issue the enclosed DSER on October 31, 1991 to comply with the advanced reactor review schedules of SECY-91-161, and place it in the NRC Public Document Room within 10 working days of the date that this paper is issued to GE.

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

James M. Taylor
Executive Director
for Operations

Enclosures: - Commissioners, SECY, OGC only
DSER for Chapters 1, 2, 3, 5, 6, 8, 9, 10, 12, 13, 14, and 15

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DRAFT SAFETY EVALUATION REPORT

ON

CHAPTERS 1, 2, 3, 5, 6, 8, 9, 10, 12, 13, 14, and 15 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

October 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 DSER summarizes the results, to date, of the staff's safety review of Chapters 2, 3, 6, 9, 10, and 13, in part, and Chapters 8, 12, 14, and 15 of the ABWR Standard Safety Analysis Report (SSAR). Parts of Chapters 2, 3, 6, 9, 10, and 13 were covered in previously issued DSERs. Staff review of ABWR SSAR Chapters 1, 2, 3, 4, 5, 6, and 17, and Chapters 3, 9, 10, 11, and 13 were issued in May 1991 and August 1991, respectively. The third, forth, and fifth DSERs covering Chapter 7, Chapter 19, and Chapter 18, respectively, of the SSAR were issued in October 1991. 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.

The chapters of this DSER addresses those portions of the ABWR SSAR that were reviewed against the corresponding chapters of the Standard Review Plan (SRP) (NUREG-0800), 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).

Presently all of the staff's acceptance criteria as defined in the 18 chapters of the SRP have been addressed in this and in previously issued DSERs. The only SRP chapter not addressed is Chapter 16, "Technical Specification." The ABWR Technical Specifications (TS) are being reviewed in parallel with the Boiling Water Reactor Owners Group's development of the standard TS for the BWR/6 designs. Areas outside the SRP still needing staff review are USIs/GSIs, reliability assurance, TMI and CP/ML issues, and operational experience. Completion of the staff's review of those areas outside of the SRP reviews are dependent, in part, on the closure of a number of open items identified in the DSERs as well as a need for additional information on some of these areas. Both TS and the other issues will be addressed at a later date. The staff's review of severe accident mitigative design alternatives (SAMDAs) for the ABWR will be carried out (when submitted by GE) consistent with the requirements of the National Environmental Policy Act.

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 the final safety evaluation.

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 the final safety evaluation.

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) which are identified in the version of NUREG-0933, "A Prioritization of Generic Safety Issues," current on the date six months prior to the application. GE has committed to identify and address any of these issues that are applicable to the ABWR design and any new prioritized 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 the final safety evaluation for the ABWR.

1.8 Outstanding Issues

Certain outstanding regulatory review issues for the ABWR, many of which involve the need for greater design detail, had not been resolved with GE when the NRC issued this report. The staff will discuss the resolution of the issues in the final safety evaluation. The issues and the SER sections in which they are discussed are provided below. This section of this report identifies all issues that have been identified as outstanding in the SSAR sections included in this report.

Issue	Status
(1) Accessibility to ASME Class 1, 2, and 3 components (5.4.2, 6.6)	Awaiting information
(2) Modeling of ABWR containment drywell volumes (6.2.1.2.1)	Awaiting information
(3) ABWR containment design testing (6.2.1.2.1)	Awaiting information
(4) Drywell depressurization (6.2.1.5.1)	Awaiting information
(5) Suppression pool SRV loading tests (6.2.1.6)	Awaiting information
(6) Data for suppression pool test conditions (6.2.1.6)	Awaiting information
(7) Clarification of assumptions (6.2.1.6)	Awaiting information
(8) Clarification of methods used to calculate loads (6.2.1.6)	Awaiting information
(9) Justification for scaling laws (6.2.1.6)	Awaiting information

Issue	Status
(10) Additional modeling assumptions (6.2.1.6)	Awaiting information
(11) Subcompartment pressure analysis (6.2.1.7)	Awaiting information
(12) Subcompartment pressure analysis inconsistencies (6.2.1.7)	Awaiting information
(13) Steam bypass of the suppression pool (6.2.1.8)	Awaiting information
(14) Administrative control of openings, doors and hatches (6.2.3) ...	Awaiting information
(15) Response to questions regarding containment isolation system (6.2.4) ..	Awaiting information
(16) Commitment to GDCs 1, 2, 4, 16 and 56 (6.2.4)	Awaiting information
(17) Containment isolation valve information (6.2.4)	Awaiting information
(18) Containment purge system design criteria (6.2.4.1)	Awaiting information
(19) Atmospheric control system design (6.2.5)	Awaiting information

Issue	Status
(20) Capability of post-LOCA purging of the containment (6.2.5)	Awaiting information
(21) Availability of hydrogen recombiners following a LOCA (6.2.5) ..	Awaiting information
(22) SSAR inconsistencies re offsite power system description (8.2)	Under review
(23) Physical and electrical separation concerns (8.2)	Under review
(24) Clarification of interface requirements for offsite circuits (8.2)	Under review
(25) Circuit testing requirements (8.2)	Under review
(26) Circuit capacity information (8.2)	Under review
(27) Identification of Standard Review Plan (SRP) criteria applicable to offsite systems (8.2)	Under review
(28) Single failure General Design Criterion 17 concerns (8.2.1)	Awaiting information
(29) Trip of all reactor internal pumps-power systems reliability concerns (8.2.2) ..	Awaiting information

Issue | Status

(30) Clarification of SRP criteria and application of IEEE-279 (8.3.1) .... Under review

(31) Protection of safety systems (8.3.1.1) .... Under review

(32) Separation of conduits from non-enclosed raceways (8.3.2.1) .... Awaiting information

(33) Class 1E penetration separation (8.3.2.2) .... Awaiting information

(34) Class 1E/non 1E penetration separation (8.3.2.2) .... Awaiting information

(35) Class 1E penetration/non 1E cables (8.3.2.2) .... Awaiting information

(36) Clarification of divisional separation criteria 8.3.2.3) .... Awaiting information

(37) Separation and color coding of cables within cabinets or panels (8.3.2.4) ... Awaiting information

(38) Clarification of criteria for associated circuits (8.3.2.5) .... Awaiting information

(39) Clarification of method for identifying cables and raceways (8.3.2.6) .... Awaiting information

Issue	Status
(40) Clarification of cable routing criteria (8.3.2.7)	Awaiting information
(41) Compliance with IEEE-384 1974 version (8.3.2.8)	Awaiting information
(42) Clarification of fire barrier separation design basis (8.3.2.8)	Awaiting information
(43) Clarification of interface requirements for protection of penetrations and definition of current limiting devises (8.3.3.1)	Awaiting information
(44) Effectiveness of safety bus grounding interlocks (8.3.3.2)	Awaiting information
(45) Clarification of qualification design information (8.3.3.3)	Under review
(46) Clarification of SSAR information related to submergence of electrical equipment (8.3.3.4)	Under review
(47) Clarification of effects of fire suppressant on electrical equipment (8.3.3.5)	Under review

Issue | Status

(48) Isolation between Class 1E and non-1E loads (8.3.3.6) .... Awaiting information

(49) Clarification of diesel generator protective relaying (8.3.3.7) .... Awaiting information

(50) Testing of thermal overload bypass devise (8.3.3.8) .... Awaiting information

(51) Clarification of SSAR regarding breaker coordination (8.3.3.9) .... Awaiting information

(52) Protective relaying setpoint information (8.3.3.10) .... Under review

(53) Clarification of design criteria for fault interrupting capacity and related interface requirements (8.3.3.11) .... Awaiting information

(54) Protection of valve motors from overloads (8.3.3.12) .... Awaiting information

(55) Clarification of the SSAR discussion of provision of two electrical protection assemblies (8.3.3.14) .... Awaiting information

(56) Clarification of SSAR regarding independence of divisions (8.3.4.1) .... Under review

Issue | Status

(57) Clarification of information on constant voltage supplies (8.3.4.2) .... Under review

(58) Clarification of independence of power circuits for safety/relief valves (8.3.4.3) .... Under review

(59) Design information for lighting systems (8.3.5) .... Awaiting information

(60) Clarification of design control information (8.3.6.1) .... Under review

(61) Design bases control-consistency throughout the SSAR (8.3.6.2) .... Awaiting information

(62) Equipment testing concerns (8.3.7) .... Awaiting information

(63) Clarification of SSAR regarding shutdown capability (8.3.8.1) .... Under review

(64) Description of auxiliary DC power system (8.3.8.2) .... Awaiting information

(65) Clarification of SSAR discussion of Class 1E 125 v battery capacity (8.3.8.3) .... Awaiting information

Issue	Status
(66) Use of silicone diode in DC system (8.3.8.4)	Awaiting information
(67) Design information on diesel generator (8.3.8.5)	Awaiting information
(68) Station blackout concerns (8.3.9)	Awaiting information
(69) Compressed Air System (CAS) Valve number and valve operator inconsistencies (9.3.1)	Awaiting information
(70) Identification of CAS component safety classification (9.3.1)	Awaiting information
(71) Identification of CAS valve failure modes (9.3.1)	Awaiting information
(72) Failure position of CAS valves AO F018A & B (9.3.1)	Awaiting information
(73) ANSI compliance of High Pressure Nitrogen Gas Supply System (9.3.1)	Awaiting information
(74) Discrepancy in identification of Instrument Air (IA) System (9.3.1)	Awaiting information

<u>Issue</u> | <u>Status</u>

(75) Justification for particulate size for IA system (9.3.1) .... Awaiting information

(76) Discrepancy in identification of Radioactive Drain Transfer System containment isolation valves (9.3.8) .. Awaiting information

(77) Classification of Radioactive Drain Transfer System check valves (9.3.8) .. Awaiting information

(78) Discrepancy resolution and component qualification requirements (9.3.8) .... Awaiting information

(79) Safety related designation of drain system (9.3.8) .... Awaiting information

(80) Provisions for tornado missiles Auxiliary Support Systems (9.5.4.1) ... Awaiting information

(81) Interface requirement for DG fuel oil transfer pump motive power (9.5.4.2) .... Awaiting information

(82) Verification of interfaces and instrumentation discrepancies (9.5.4.2) .... Awaiting information

Issue	Status
(83) Figure 9.5-6 discrepancies (9.5.4.2) ..	Awaiting information
(84) Interface requirement for verifying day tank full (9.5.4.2)	Awaiting information
(85) Discrepancies in SSAR regarding jacket circulating water pump (9.5.5)	Awaiting information
(86) Interface requirement for temperature sensor (9.5.5)	Awaiting information
(87) DG cooling water heat removal capacity (9.5.5)	Awaiting information
(88) Documentation of DG starting air system filter arrangement (9.5.6)	Awaiting information
(89) DG starting air system interface requirements (9.5.6)	Awaiting information
(90) Omission of reference to air compressor discharge coolers (9.5.6)	Awaiting information
(91) Omission of DG lubrication system level indication (9.5.7)	Awaiting information

Issue	Status
(92) Identification of design criteria as interface requirements (9.5.7)	Awaiting information
(93) Classification of components in the DG lubrication system (9.5.7)	Awaiting information
(94) Identification of ASME and ANSI components (9.5.7)	Awaiting information
(95) Selection of a combustion air flow capacity (9.5.8)	Awaiting information
(96) Diesel Generator Combustion Air Intake and Exhaust System provisions for tornado missiles (9.5.8)	Awaiting information
(97) Design information on areas not included in SSAR (9.5.8)	Awaiting information
(98) Adequacy of process sampling system (9.3.2.1)......................	Awaiting information
(99) Adequacy of post-accident sampling system (9.3.2.2)......................	Awaiting information
(100) Adequacy of hydrogen water chemistry system (9.3.9)	Awaiting information

Issue	Status
(101) Adequacy of zinc injection system ...	Awaiting information
(102) Fire hazards analysis (9.5.1.1)..............	Under review
(103) HVAC smoke removal-system description (9.5.1.2) (9.5.1.4.4)	Awaiting information
(104) Water distribution and fire extinguishing systems (9.5.1.3.3)	Under review
(105) Adequacy of condensate cleanup system (10.4.6)	Awaiting information
(106) Upper drywell shielding concerns (12.1.2)	Under review
(107) Identification and description of contained and airborne radioactive sources in SSAR (operation and accident conditions) (12.2.1)	Awaiting information
(108) Plant layout drawing deficiencies regarding identification of radiation sources, legibility, and consistency (12.3.1)	Awaiting information
(109) Identification of post-loss-of-coolant-accident (LOCA) vital areas (12.3.1)	Awaiting information

Issue | Status

(110) Justification of high radiation zone above spent fuel pool during operation 12.3.1) .... Awaiting information

(111) Identification of "highly radioactive systems" (12.3.1) .... Awaiting information

(112) Drywell and reactor vessel shielding design information (12.3.2) .... Under review

(113) Airborne contamination information (12.3.3) .... Awaiting information

(114) Airborne radiation monitoring information (12.3.4) .... Awaiting information

(115) Dose assessment background, bases, consistency, justification (12.4) ... Awaiting information

(116) Power-to-flow operating map (14.2.11) .... Awaiting information

(117) Table listing startup tests and test conditions (14.2.11) .... Awaiting information

(118) Generic interfacing support system availability individual test abstracts 14.2.12 .... Awaiting information

Issue | Status

(119) GE commitment to perform tests (14.2.12) .... Awaiting information

(120) Modifications to individual test abstracts (14.2.12) .... Awaiting information

(121) Clarify acceptance criteria and modify startup test abstracts (14.2.12) .... Awaiting information

(122) Screening to identify tests not essential to demonstrate conformance (14.2.12) .... Awaiting information

(123) Conformance of the ABWR with RG 1.68 Revision 2 (14.2.12.3) ...... Awaiting information

(124) TMI Items (14.2.12.3) .... Awaiting information

(125) Functioning of conductivity meters (14.2.12.4) .... Awaiting information

(126) Modification to feedwater control test description (14.2.12.4) .... Awaiting information

(127) Clarification of feedwater control test acceptance criteria (14.2.12.4) Awaiting information

Issue		Status
(128)	Total Air Demand for Instrument Air and Station Air (14.2.12.4)	Awaiting information
(129)	Functional testing of instrument and control air systems (14.2.12.4)	Awaiting information
(130)	HVAC preoperational testing requirements (14.2.12.4)	Awaiting information
(131)	Design, maintenance, and testing criteria for ventilation systems (14.2.12.4)	Awaiting information
(132)	RHR system isolation (14.2.12.4)	Awaiting information
(133)	Verification of relief valve settings by vendor bench tests (14.2.12.4) ...	Awaiting information
(134)	Approval of REDYA and ODYNA. (15.1) .	Awaiting information
(135)	Loss of Feedwater heater transient, (15.1 Item (1)) ...	Awaiting information
(136)	Software reliability in determining limiting faults. (15.1 Item (3))	Awaiting information
(137)	Rod block algorithm and setpoint (15.1 Item (4)(b))	Awaiting information

Issue | Status

(138) Credit for non-safety grade equipment
in safety analysis (15.1 Item (6)) .. Awaiting information

(139) Slow turbine control valve
closure event (15.1 Item (6)) .... Awaiting information

(140) Compliance of ATWS Rule
10 CFR 50.62.(15.4) .... Awaiting information

(141) Pressure Suppression Pool as a Fission
Product Cleanup System. (15.3 (1)) .. Awaiting information

(142) Radioactive Iodine Deposition
in the Main Steam Lines
and Condensers (15.3 (3)) .... Awaiting information

(143) Containment leak rate (15.3.1) ...... Awaiting information

(144) Compliance of ATWS Rule
10 CFR 50.62.(15.4) .... Awaiting information

(145) Compliance with 10 CFR 50.55a(g)
(5.2, 6.6) .... Awaiting information

(146) Design basis tornado analyses (2.3.2) .Awaiting information

1.9 Confirmatory Issue

(1) GE commitment to add additional references to RGs (14.2.7).
(2) Completion of pre-operational testing (14.2.10).

1.10 Interface Information

(1) Primary Containment Leakage Rates (15.3(2))

(2) Primary coolant activity limits (15.3(4))

(3) Main steam isolation valve leak rate (15.3(3))

(4) Inspection and surveillance requirements for suppression pool bypass (15.3(1))

2 SITE CHARACTERISTICS

2.3 Meteorology

2.3.2 Local Meteorology

The applicant proposed in Table 2.0-1, "Envelope of ABWR Standard Plant Site Design Parameters" that the maximum tornado wind speed of 260 miles per hour and the tornado recurrence interval of one million years (10^{-6} per year tornado strike probability) be utilized for the design basis tornado. These parameters are based on ANSI/ANS 2.3, "Standard for Estimating Tornado and Extreme Wind Characteristics at Nuclear Power Sites."

The current NRC regulatory position with regard to design basis tornados is contained in two 1974 documents, WASH-1300, "Technical Basis for Interim Regional Tornado Criteria," and Regulatory Guide 1.76, "Design Basis Tornado for Nuclear Power Plants." WASH-1300 states that the probability of occurrence of a tornado that exceeds the Design Basis Tornado (DBT) should be on the order of 10^{-7} per year per nuclear power plant and the regulatory guide delineates the maximum wind speeds of 240 to 360 miles per hour depending on the regions.

The staff has not endorsed the ANSI/ANS 2.3. However, the staff has reevaluated the regulatory positions in Regulatory Guide 1.76 using current tornado data. The staff documented the results of its evaluation in a report, U.S. Nuclear Regulatory Commission, "Tornado Climatology of the Contiguous United States," NUREG/CR-4461 (PNL-9697), dated May 1986 (Reference 1). At the heart of this study is the tornado data tape prepared by the National

Severe Storm Forecast Center (NSSFC) with 30 years of data, 1954 through 1983. This data tape contains the data for the approximate 30,000 tornados that occurred during the period.

The staff found that the tornado strike probabilities range from near 10^{-7} per year for much of the western United States to about 10^{-3} per year in the central United States. The wind speed values associated with a tornado having an annual strike probability of 10^{-7} range from less than 153 mph to 332 mph. These wind speed estimates are 50 to 100 mph lower than the speed estimates presented in WASH-1300 and Regulatory Guide 1.76 for most of the United States. On the basis of this analysis, the staff concluded in its report (Reference 1) that it would appear to be reasonable to reduce tornado design basis wind speeds to 200 mph for the United States west of the Rocky Mountains and to 300 mph for the United States east of the Rocky Mountains.

The staff will accept the tornado design basis proposed by GE, but the use of the specific design criteria may limit the number of acceptable sites. Furthermore, the tornado design basis requirements have been used in establishing structural requirements (minimum concrete wall thicknesses) for the protection of nuclear plant safety related structures, systems, and components against the effects not covered explicitly in review guidance such as Regulatory Guides or the Standard Review Plan. Specifically, some aviation (general aviation light aircraft) crashes, nearby explosions, and explosion debris or missiles have been reviewed and evaluated routinely by the staff by taking into account the existence of the tornado protection requirements. Hence, the staff's acceptance will also necessitate a concurrent review and evaluation of their effect on the protection criteria for some external impact hazards, such as general aviation or nearby explosions. This item remains open pending the satisfactory evaluation of a submittal for the ABWR design which addresses the above external phenomena.

Reference 1 U.S. Nuclear Regulatory Commission, "Tornado Climatology of the Contiguous United States," NUREG/CR-4461 (PNL-9697), May 1986

3 DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

3.5 Missiles

3.5.1 Missile Selection and Description

3.5.1.5 Site Proximity Missiles (Except Aircraft)

In the SSAR Section 3.5.1.5, GE states that "external missiles other than those generated by tornados are not considered design basis...," since the resultant event probability is $\leq 10^{-7}$. SSAR Section 3.5.4.3 includes an interface which requires a utility applicant to provide an analysis which demonstrates that the probability of missiles impacting the ABWR Standard Plant and causing consequences greater than 10 CFR Part 100 guidelines is $\leq 10^{-7}$. The staff will determine the acceptability of such an analysis on a site specific basis.

3.5.1.6 Aircraft Hazards

In the SSAR Section 3.5.1.6, GE states that "aircraft hazards are not a design basis event...," since the resultant event probability is $\leq 10^{-7}$. SSAR Section 3.5.4.3 includes an interface which requires a utility applicant to provide an analysis which demonstrates that the probability of aircraft impacting the ABWR Standard Plant and causing consequences greater than 10 CFR Part 100 guidelines is $\leq 10^{-7}$. The staff will determine the acceptability of such an analysis on a site specific basis.

5 REACTOR COOLANT SYSTEM

5.2 Compliance With Code and Code Cases

5.2.4 Inservice Inspection and Testing of Reactor Coolant Pressure Boundary

The staff has reviewed this section and finds the information pertaining to compliance with 10 CFR 50.55a(g), design access, preservice inspection requirements and proposed methodology for inservice inspections unacceptable. The information in SSAR Section 5.2.4 contains inspection requirements from a specific edition of ASME Section XI, i.e., a "reference edition." The actual requirements for preservice and inservice inspections are dependent upon the edition of ASME Section XI in effect at the time of the construction of the particular component, as defined in 10 CFR 50.55a(g). Therefore, the staff reviewed the information in SSAR Section 5.2.4 as the concepts that GE intends to apply.

5.2.4.1 Compliance with the Standard Review Plans

The staff is conducting its review according to SRP Section 5.2.4.

Paragraph 5.2.4.2 - Accessibility states:

> "All items within the Class 1 boundary are designed, to the extent practicable, to provide access for the examinations required by ASMC [sic] Section XI, IWB-2500. Items for which the design is known to have inherent access restrictions are described in Subsection 5.2.4.8."

The staff finds that the proposed standards for accessibility are not consistent with 10 CFR 50.55a(g)(3) which states:

> "(i) Components which are classified as ASME Code Class 1 shall be designed and be provided with access to enable the performance of inservice examination of such components and shall meet the preservice examination requirements set forth in Section XI of editions of the ASME Boiler and Pressure Vessel Code and Addenda applied to the construction of the particular component."

The staff concludes that the design of the ABWR must be in compliance with NRC regulations.

5.2.4.2. Examination Requirements

GDC 32, "Inspection of Reactor Coolant Pressure Boundary," requires that components that are part of the RCPB be designed to permit periodic examination and testing of important areas and features to assess their structural and leaktight integrity. To ensure that no deleterious defects develop during service, selected welds and weld heat-affected zones are to be examined periodically.

The design of the ASME Code Class 1 and 2 components of the RCPB must incorporate provisions for access for inservice examinations, as required by Paragraph IWA-1500 of Section XI of the ASME Code.

The design of the RCPB may incorporate exclusions from examination as defined in paragraphs IWB-1220 and IWC-1220 of ASME Section XI for ASME Class 1 and 2 components, respectively. However, for the ABWR, GE states in Paragraph 5.7.4.1.2 [sic] Exclusions:

"Portions of systems within the reactor coolant pressure boundary, as defined in 5.2.4.1.1, that are excluded from the Class 1 boundary are as follows:

(1) those components where, in the event of postulated failure of the component during normal reactor operation, the reactor can be shut down and cooled down in an orderly manner, assuming makeup is provided by the reactor coolant makeup system only; and

(2) components which are or can be isolated from the reactor coolant system by two valves (both closed, both open, or one closed and one open). Each such open valve is capable of automatic actuation and if the other valve is open its closure time is such that, in the event of postulated failure of the component during normal reactor operation, each valve remains operable and the reactor can be shut down and cooled down in an orderly manner assuming makeup is provided by the reactor coolant makeup system only."

The staff finds that the proposed exclusion criteria described in paragraph 5.7.4.1.2 [sic] if intended to be used to develop examination requirements are not consistent with 10 CFR 50.55a(g)(3). The staff position is that if the design of the ABWR incorporates exclusions from examination criteria, then those exclusions shall be in compliance with NRC regulations in effect at the time of the construction of the particular component.

5.2.4.2.1 Examination Methods

Paragraph 5.2.4.3.2.1 Ultrasonic Examination of the Reactor Vessel states:

> "Ultrasonic examination of the RPV will be conducted in accordance with ASME Section XI, IWA-2232 (a), and Section V, Article 4. In addition the ultrasonic examination system shall meet the requirements of Regulatory Guide 1.150 as described in Table 5.2-9. RPV welds and nozzles subject to examination are shown in Figure 5.2-7a."

The staff finds the commitment to use Regulatory Guide 1.150 for the ultrasonic examination of the reactor vessel to be acceptable. However, GE's proposal to use ASME Section V, Article 4 is not consistent with the plans of the nuclear industry and ASME Code for future inservice inspections. ASME Section XI has published Appendix VII, "Qualification of Nondestructive Examination Personnel For Ultrasonic Examination," and Appendix VIII, "Performance Demonstration For Ultrasonic Examination Systems." The NRC has published in the Federal Register its intent to reference in 10 CFR 50.55a(b) the ASME Section XI edition that includes the publ ned Appendix VII. In addition, the NRC staff has established a technical contact to coordinate the implementation of Appendix VIII. Therefore, the preservice inspection program for the ABWR should include provisions that ultrasonic testing be performed in accordance with Appendices VII and VIII pursuant to 10 CFR 50.55a(g)(3).

5.2.4.3 Evaluation of Compliance with 10 CFR 50.55a(g)

The staff concludes that the information in SSAR Section 5.2.4 is not in compliance with 10 CFR 50.55a(g). GE should revise SSAR Section 5.2.4 in its entirety to include information and commitments that are consistent with NRC regulations and conditions that are anticipated to occur during actual inspections.

Interface Requirement: Utilility applicants referencing the ABWR design must: (1) Docket a complete and acceptable preservice inspection (PSI) program. The PSI program must be in compliance with 10 CFR 50.55a(g)(3), and include reference to the edition and addenda of ASME Section XI that will be used for the selection of components for examinations, lists of the components excluded from examination by the applicable code, and isometric drawings. (2) Submits plans for preservice examination of the reactor pressure vessel welds to address the degree of compliance with RG 1.150, "Ultrasonic Testing of Reactor Vessel Welds During Preservice and Inservice Examinations."

Interface Requirement: Utility applicants referencing the ABWR design will need to submit the complete plant-specific inservice inspection (ISI) program. This program will be evaluated after the applicable ASME Code edition and addenda are determined based on 10 CFR 50.55a(b), but before inservice inspection begins during the first refueling outage.

The staff considers the review of the ABWR plant specific PSI/ISI programs an interface requirement. Each utility applicant's submittal will need to address the information noted above.

5.2.4.6 Conclusions

Periodic examinations and hydrostatic testing of pressure-retaining components of the RCPB, in accordance with the requirements of Section XI of the ASME Code and 10 CFR Part 50, will provide reasonable assurance that structural degradation or loss of leaktight integrity during service will be detected in time to permit corrective action before the safety functions of a component are compromised. Compliance with the preservice and inservice examinations required by the ASME Code and by 10 CFR Part 50 constitutes an acceptable basis for satisfying the inspection requirements of GDC 32.

The staff finds the information in SSAR Section 5.2.4 unacceptable. The staff will complete the evaluation in a future revision of this report after GE provides supplemental information.

6 ENGINEERED SAFETY FEATURES

6.2 Containment Systems

The containment systems for the ABWR include a containment structure as the primary containment, a secondary containment (reactor building) surrounding the primary containment and housing equipment essential to safe shutdown of the reactor and fuel storage facilities, and supporting systems. The primary containment is designed to prevent the uncontrolled release of radioactivity to the environment with a leakage rate of 0.5 percent by weight per day at the calculated peak containment pressure related to the DBA. The secondary containment is designed to confine the leakage of airborne radioactive materials from the primary containment. Figure 6.2.1 shows the principal features of the ABWR containment.

6.2.1 Primary Containment Functional Design

The ABWR primary containment design has the following main features:

1. A drywell comprised of two volumes: (a) an upper drywell (UD) volume surrounding the reactor pressure vessel (RPV) and housing the steam and feedwater lines and other connections of the reactor primary coolant system, safety/relief valves and the drywell HVAC coolers, and (b) a lower drywell (LD) volume housing the reactor internal pumps, control rod drives and under vessel components and servicing equipment.

The upper drywell is a cylindrical steel-lined reinforced concrete structure with a removable steel head and a reinforced concrete steel diaphragm floor. The cylindrical RPV pedestal, which is connected rigidly to the steel diaphragm floor, separates the lower drywell from the wetwell. Ten UD to LD connecting vents (DCVs), approximately 1M X 2M in cross-section, are built into the RPV pedestal. The DCVs are extended downward via 1.2M inside diameter steel pipes, each of which has three horizontal vent outlets into the suppression pool.

The drywell, which has a net free volume of 259,563 ft^3, is designed to withstand design pressure and temperature transients following a loss of coolant accident (LOCA) and also the rapid reversal in pressure when the steam in the drywell is condensed by emergency core cooling system flow during post LOCA flooding of the RPV. A wetwell-to-drywell vacuum relief system is provided to prevent back-flooding of the suppression pool water into the lower drywell and to protect the integrity of the steel diaphragm floor slab between the drywell and wetwell and the drywell structure and liner. The drywell design pressure and temperature are 45 psig and 340°F, respectively. The design drywell-to-wetwell differential pressures are (+) 25 psid and (-)2 psid. The design drywell-to-reactor building negative differential pressure is (-)2 psid.

2. A system of drywell-to-wetwell vents which channel blowdown from the drywell and discharge into the suppression pool following a LOCA.

There are 30 vents in the vertical section of the lower drywell below the suppression pool water level, each with a nominal diameter of 2.3 feet. These vents are arranged in 10 circumferential columns, each containing three vents. The three vent centerlines in each column are located at 11.48 feet, 15.98 feet and 20.48 feet below the suppression pool water level when the suppression pool is at the low water level.

3. A wetwell, comprised of an air volume and suppression pool, with a net free air volume of 210,475 ft^3 and a minimum pool volume of 126,427 ft^3 at low water level.

   The wetwell is designed for an internal pressure of 45 psig and a temperature of 219°F. The design wetwell-to-reactor building negative differential pressure is (-)2 psid. The suppression pool, located inside the wetwell annular region between the cylindrical RPV pedestal wall and the outer wall of the wetwell, is a large body of water which serves as a heat sink for postulated transients and accidents and as a source of cooling water for the emergency core cooling system (ECCS). In the case of transients that result in a loss of the ultimate heat sink, energy would be transferred to the pool by the discharge piping from the reactor system's safety/relief valves (SRVs). In the event of a LOCA within the drywell, the drywell atmosphere is vented to the suppression pool through the system of drywell-to-wetwell vents.

This primary containment design basically uses combined features of the Mark II and Mark III designs, with the exception that the drywell is composed of upper and lower drywell volumes. The vents to the suppression pool are a combination of the vertical Mark II and horizontal Mark III systems. The wetwell is similar to a Mark II wetwell. Table 6.2.1 provides a comparison of the design parameters for Mark I, II, III, and ABWR containments.

6.2.1.1 LOCA Chronology

Following a postulated LOCA, the drywell pressure increases as a result of blowdown of the reactor coolant system. Pressurization of the drywell causes the water initially in the vent system to be accelerated into the pool until the vents are cleared of water. During this clearing process, the water leaving the horizontal vents forms jets in the suppression pool and causes water jet impingement loads on the structures within the suppression pool and on the containment wall opposite the vents.

During the vent clearing transient, the drywell is also subjected to a pressure differential, and the RPV pedestal wall experiences a vent-clearing reaction force.

Immediately following vent clearing, an air and steam bubble forms at the exit of the vent. The bubble pressure initially is assumed equal to the existing drywell and wetwell differential pressure. This bubble transmits a pressure wave through the suppression pool water and results in a loading on the suppression pool boundaries and on equipment located in the suppression pool.

As the air and steam flow from the drywell becomes established in the vent system, the initial vent exit bubble expands to equalize the suppression pool hydrostatic pressure. GE's large-scale Pressure Suppression Test Facility (PSTF) tests show that the steam fraction of the flow is condensed, but continued injection of drywell air and expansion of the air bubble result in a rise of the suppression pool surface. During the early stages of this process, the pool swells in a bulk mode (i.e., a slug of solid water is accelerated upward by the air pressure). Structures close to the pool surface will experience loads as the rising pool surface impacts the lower surface of the structure. In addition to these initial impact loads, these same structures will experience drag loads as water flows past them. Equipment in the suppression pool will also experience drag loads.

Data from PSTF air tests indicate that after the pool surface has risen approximately 1.6 times the initial submergence of the top vent (which translates to 12 feet above the initial pool surface for the Mark III design, the thickness of the water ligament could be as small as 2 feet or less, and the impact loads would then be significantly reduced. This phase is referred to as "incipient breakthrough," i.e., the ligament begins to break up. To account for possible non conservatisms in the test facility arrangement and instrumentation error bands, the staff has determined that the breakthrough height should be set at 18 feet above the initial pool surface for the Mark III design. The staff's evaluation of the breakthrough height for ABWR design is discussed in Section 6.2.1.6 of this SER.

As the drywell air flow through the horizontal vent system decreases and the air/water suppression pool mixture experiences gravity-induced phase separation, pool upward movement stops and the fallback process starts. During this process, floors and other flat structures experience downward loading, and the containment wall theoretically can be subjected to a small pressure increase. However, this pressure increase has not been observed experimentally.

As the reactor blowdown proceeds, the primary system is depleted of high-energy fluid inventory and the steam flow rate to the vent system decreases. This reduced steam flow rate leads to a reduction in the drywell-to-wetwell pressure differential that, in turn, results in a sequential recovering of the horizontal vents. Suppression pool recovery of a particular vent row occurs when the vent stagnation differential pressure corresponds to the suppression pool hydrostatic pressure at that row of vents.

Toward the end of the reactor blowdown, the top row of vents is capable of condensing the reduced blowdown flow and the two lower rows will be totally recovered. As the blowdown steam flow decreases to very low values, the water in the top row of vents starts to oscillate, causing what has become known as vent chugging. This action results in dynamic loads on the top vents and on the RPV pedestal wall opposite the upper row of vents. In addition, an oscillatory pressure loading condition can occur on the drywell and wetwell. Because this phenomenon is dependent on a low steam mass flux (the chugging threshold appears to be in the range of 10 $lb/sec/ft^2$), it is expected to occur for all break sizes. For smaller breaks, it may be the only mode of condensation that the vent system will experience.

The staff's evaluation of this LOCA-related pool dynamic loads is discussed in Section 6.2.1.6 of this SER.

Shortly after onset of a DBA, the ECCS pumps automatically start and pump suppression pool water into the reactor pressure vessel. This water floods the reactor core and, if the operator fails to follow the emergency procedure guidelines requiring ECCS flow to be throttled, the water starts to cascade into the drywell from the break. When this occurs depends on the size and location of the break. Because the drywell is full of steam at the time of vessel flooding, the sudden introduction of cool water causes rapid steam condensation and drywell depressurization. When the drywell pressure falls below the wetwell airspace pressure, air from the wetwell redistributes between the drywell and wetwell via the wetwell-to-drywell vacuum relief system. Eventually enough air returns to equalize the drywell and wetwell pressures, however, during this drywell depressurization transient, there is a period of negative pressure on the drywell structure. A negative load condition of (-)2 psid is, therefore, specified for drywell design. The staff's evaluation of this drywell to wetwell negative differential pressure is discussed in Section 6.2.1.5.1 of the SER.

6.2.1.2 Containment Analysis

The staff's review of the containment design included the temperature and pressure responses of the drywell and wetwell to a spectrum of LOCAs, the capability to withstand the effects of steam bypass from the drywell directly to the air region of the suppression pool, the external pressure capability of the drywell and wetwell and the negative drywell-to-wetwell differential pressure. In addition, the review considered GE's proposed design bases and criteria for the containment, the analyses and test data in support of the criteria and bases, and the loads resulting from pool dynamic phenomena.

6.2.1.2.1 Containment Analytical Model

GE's calculation of the short term and long term containment pressure-temperature response to postulated high energy line breaks used the same analytical models and conservative assumptions that were previously presented and reviewed for the Mark III containment in GESSAR II. The staff found these to be acceptable using independent confirmatory analyses with the CONTEMPT-LT28 computer code. These models and assumptions are discussed in the ABWR SSAR and NEDO-20533 and its Supplement 1, "The G.E. Mark III Pressure Suppression Containment Analytical Model." In response to the staff's request for additional information (RAI), GE stated that the analytical models described in NEDO-20533 are appropriate to calculate the ABWR containment responses to postulated accidents. Though originally written for prediction of Mark III transients, these models, which simulate the transient conditions in the containment, can be adapted for the ABWR containment configuration. These models simulate the drywell, vent systems, and wetwell (suppression pool and airspace). They are, therefore, adaptable to other containment configurations having the same basic components.

As indicated in Section 6.2.1 of this DSER, the ABWR containment design uses combined features of Mark II and Mark III designs, with the exception of a unique feature of two drywell volumes (upper and lower). The vent system is a combination of vertical (Mark II design) and horizontal (Mark III design) drywell-to-wetwell vent systems and the wetwell (suppression pool and airspace) is similar to a Mark II. However, GE has not provided a detailed discussion to describe how the two ABWR drywell volumes and the combination vertical and horizontal vent system are modeled in the computer code to represent the physical geometry of the containment, and how the air carryover from the two drywell volumes to the wetwell is treated in the computer code. The impact of any difference in the hydrodynamic force, caused by venting, between the Mark III design (vent annulus) and ABWR design (pipe vents) is also unclear. The staff requires this information for its review of the ABWR containment analysis. In addition, the staff will require tests to verify that:

1. Following a LOCA, the combination of vertical and horizontal drywell to wetwell vent system will perform (to demonstrate venting clearing, condensation and chugging) as predicted.

2. Following a LOCA, the containment will perform (air carryover, and containment pressure and temperature responses) as predicted by the analytical model.

Based on its review, the staff has not been able to conclude that the assumptions and analytical models used to predict the containment pressure and temperature transients following a LOCA in the ABWR containment are acceptable. In a telephone conversation with the staff on August 9, 1991, GE indicated that additional information in this area was not warranted because the Mark III containment configuration described in NEDO-20533 is similar to the ABWR containment. Subsequently, GE provided a figure to justify its position. The staff reviewed this figure and concluded that information which describes the results of additional tests, identified above, is needed to adequately address the staff's concerns. This is an open item.

6.2.1.3 Short-Term Pressure Response

The maximum drywell-to-wetwell differential pressure occurs during the blowdown phase (short-term) of a LOCA. GE has performed analyses of various postulated primary system breaks, including a double-ended rupture of the main feedwater line, a double-ended rupture of the main steam line, and small break accidents. Results of the analyses indicate that the main feedwater line break (FWLB) yields the limiting drywell-to-wetwell differential pressure and peak drywell and wetwell pressure and is, therefore, the design-basis accident for the drywell and wetwell. The main steam line break (MSLB) yields the limiting drywell temperature. GE has provided comparative plots of drywell and wetwell short term pressure and temperature response to design basis, 0.5 ft^2, 0.1 ft^2, and 0.01 ft^2 breaks in both the main feedwater and main steam line piping inside the drywell. These figures substantiate the large guillotine breaks resulting in the highest drywell and wetwell pressure and temperature. However, these figures comparing different size pipe breaks do not indicate the same value of peak drywell and wetwell pressure as reported in Table 6.2-1 of the ABWR SAR. The staff will require these discrepancies to be clarified. Table 6.2.2 of this SER shows the maximum calculated and design pressure and temperature in drywell and wetwell.

Standard Review Plan (SRP) Section 6.2.1.1.c, "Pressure-Suppression Type BWR Containments," states that for Mark III plants at the construction permit stage, the containment design pressure should provide at least a 15 percent margin above the peak calculated containment pressure, and the design differential pressure between drywell and containment should provide at least a 30 percent margin above the peak calculated differential pressure. GE's calculated drywell peak pressure for the FWLB is 39 psig and maximum calculated temperature is 338°F resulting from the MSLB. The design pressure for the drywell is 45 psig which provides a margin of 15 percent above the peak calculated pressure in the drywell and is equal to the margin recommended in the SRP. Therefore, the staff finds this design margin for containment pressure acceptable.

The calculated wetwell peak pressure and maximum temperature are 26 psig and 207°F (which is 12°F below the design temperature of 219°F) resulting from the FWLB. The design pressure for the wetwell is 45 psig which provides a margin of 42 percent above the peak calculated pressure in the wetwell.

The calculated drywell-to-wetwell peak differential pressure is 16 psid and the design drywell-to-wetwell differential pressure is 25 psid which provides a design margin of 56 percent.

Based on its review and pending the acceptability of GE's analytical models as described in Section 6.2.1.2.1, the staff concludes that the containment pressure and temperature transients following a LOCA in the ABWR containment are acceptable. The staff will report the resolution of this matter in the final safety evaluation for the ABWR.

6.2.1.4 Long-Term Response

Following the short-term blowdown phase of the accident, the suppression pool temperature and containment pressure continuously increase because of the input of decay heat and sensible energy into the containment. During this period, the emergency core cooling system (ECCS) pumps, which take suction from the suppression pool, reflood the reactor pressure vessel up to the level of the main steam nozzles. Subsequently, ECCS water flows out of the break and fills the drywell establishing a recirculation flow path for the ECCS. The relatively cold ECCS water condenses the steam in the drywell and brings the drywell pressure down rapidly. After approximately 10 minutes, the residual heat removal (RHR) heat exchangers are automatically activated to remove energy from the containment via recirculation cooling of the suppression pool with the RHR service water system. This is a conservative assumption since the RHR design permits automatic initiation of containment cooling well before a 10 minute period. The containment spray is also conservatively assumed not to be used.

In the long-term analysis, GE accounted for potential post-accident energy sources. These included decay heat, pump heat rate, sensible heat, and metal-water reaction energy. GE's long-term model also assumed that the containment atmosphere would be saturated and equal to the suppression pool temperature at any time. Therefore, the containment pressure is equal to the sum of the partial pressure of air and the saturation pressure of water corresponding to the pool temperature.

Based on the above assumptions, GE calculated a peak suppression pool temperature of 206.46°F. The calculated long-term secondary peak containment drywell and wetwell pressures are well below the calculated short term peak pressures. Based on its review, and pending the acceptability of GE's analytical models as described in Section 6.2.1.2.1 of this SER, the staff finds GE's analysis for long-term response following a LOCA in the ABWR containment acceptable.

6.2.1.5 Reverse Containment Pressurization

Certain events in the primary containment cause depressurization transients that can create negative drywell-to-wetwell, drywell-to-reactor building, or wetwell-to-reactor building pressure differentials. Therefore, vacuum relief provisions may be necessary in order to limit these negative pressure differentials within design values. The events which cause containment depressurization are:

1. Inadvertent drywell/wetwell spray actuation during normal operation,

2. Post-LOCA drywell depressurization as a result of condensation of the steam by the spilled ECCS subcooled water, and

3. Wetwell spray actuation following a stuck open relief valve.

6.2.1.5.1 Drywell Depressurization

Drywell depressurization, which will create a negative drywell-to-wetwell pressure differential and/or a negative drywell-to-reactor building pressure differential, is caused by two major events:

1. Post-LOCA drywell depressurization as a result of condensation of the steam by the spilled ECCS subcooled water, and

2. Inadvertent drywell spray actuation during normal operation.

GE indicates that drywell depressurization following a feedwater line break results in the most severe negative pressure transient in the drywell. Without the provision of vacuum relief, this negative pressure transient may create a drywell-to-wetwell negative pressure differential of (-)40 psid. This pressure differential is much greater than the design negative drywell-to-wetwell pressure difference of (-)2 psid. Therefore, this transient is used to determine the size and the number of wetwell-to-drywell vacuum breakers.

Based on its analysis, GE further indicates that with a typical vacuum breaker diameter of 20 inches, a loss coefficient, K, of 3, and one single failure, eight wetwell-to-drywell vacuum breakers are required to maintain the negative pressure differentials of drywell-to-wetwell and of drywell-to-reactor building below the design negative pressure differentials of (-)2 psid.

Based on the staff's review, GE has not identified the specific arrangement of vacuum breakers (e.g., lower drywell or upper drywell 2 valves in series for bypass single failure protection), and has not proposed a test program to demonstrate that they will perform as predicted. In a telephone conversation regarding this subject on August 9, 1991, GE indicated that analyses were performed using first principle analytical models. These analyses were similar to analyses performed for other BWRs and assumed that the spray efficiency was 100 percent. The staff has not been able to conclude that the number and arrangement of wetwell-to-drywell vacuum breakers fo the ABWR are acceptable. The additional information identified above (i.e., vacuum breaker number, location and performance demonstration results), is required to allow the staff to determine the acceptability of the design. This is an open item.

6.2.1.5.2 Wetwell Depressurization

Wetwell depressurization, which will create a negative wetwell-to-reactor building negative pressure differential, is caused by the following events:

(1) Drywell and wetwell spray actuation during normal operation,

(2) Wetwell spray actuation subsequent to stuck open relief valve, and

(3) Drywell and wetwell spray actuation following a LOCA.

GE indicates that the limiting negative pressure transient in the wetwell corresponds to wetwell spray actuation following a stuck open relief valve. The effect of relief valve discharge on the suppression pool is to heat the wetwell airspace, thus increasing its pressure. When the pressure in the wetwell becomes greater than the drywell pressure, the wetwell-to-drywell vacuum relief system allows the flow of air from the wetwell to the drywell, thereby pressurizing both drywell volumes. Wetwell pressure and temperature peak when the reactor decay heat decreases below the heat removal capability from continued pool cooling and wetwell spray. Wetwell temperature and pressure decrease, but the drywell pressure remains at its peak value. When the pressure difference between the two volumes becomes greater than the hydrostatic head of water above the top vent, air flows back into the wetwell airspace, slowing down wetwell depressurization. The pressure differential between the drywell and the wetwell is maintained constant at the hydrostatic head above the top row of horizontal vents. The final pressure in the wetwell is lower than the drywell pressure because more air is transferred to the drywell during wetwell pressurization than is received during wetwell depressurization.

Inadvertent drywell or wetwell spray actuation during normal operation can cause depressurization of the sprayed volume due to the resultant condensation of vapor present in the air space. However, the magnitude of this depressurization is less than the post-LOCA or stuck-open relief valve cases because of the relatively smaller mass of condensable gas present during normal operation.

Calculation of the peak wetwell-to-reactor building negative differential pressure is based on an energy balance of the containment atmosphere before and after spray activation, assuming that the final air-vapor mixture is at 100 percent relative humidity and that there are no reactor building-to-wetwell vacuum breakers. Using these assumptions, the peak calculated wetwell-to-reactor building negative differential pressure was determined by GE analysis assuming worse accident conditions to be -1.77 psid. This is 10 percent less than the ABWR design value of -2.0 psid. The staff has reviewed the initial conditions, assumptions, and the methodology used in the GE analysis and finds them acceptable.

6.2.1.6 Suppression Pool Dynamic Loads

GE submitted proprietary Appendix 3B, "Containment Hydrodynamic Loads" to address the issues of suppression pool dynamic loads for the ABWR. Appendix 3B encompasses the areas of: SRV actuation and LOCA phenomena, SRV discharge loads, LOCA loads, submerged structure loads, and loads combinations. Each of these topics will be discussed later in this section. It should be noted that, although similar to the Mark III containment design, the ABWR has several distinctive features which affect suppression pool dynamic loads. These features are: wetwell airspace pressurization, a lower drywell volume, a smaller number of horizontal vents (30 in the ABWR vs. 120 in the Mark III), horizontal vent extension into the pool, vent submergence, and suppression pool width.

Both SRV actuation and LOCAs constitute the events which can result in the imposition of dynamic loads on the suppression pool. SRVs discharge steam from the Reactor Pressure Vessel (RPV) through discharge piping which is routed into the suppression pool and fitted at its suppression pool end with a quencher to enhance heat transfer between the hotter SRV discharge fluid (steam and air) and the cooler suppression pool water.

SRV discharge into the suppression pool consists of the following three phases which are listed in the order they occur: water-clearing, air-clearing, and steam flow. The discharge pipe standing column of water is first pushed out or cleared into the pool by blowdown steam pressure. Water-clearing creates SRV pipe pressure and thermal loads, pipe reaction forces, drag loads on structures submerged in the pool, and pool boundary loads. Following water-clearing, air-clearing occurs as air above the water column in the pipe is forced out t e pipe and into the pool. The air-clearing phase generates expanding bubbles in the pool which causes transient drag loads on submerged structures due to both the velocity and acceleration fields and oscillating pressure loads on the pool boundary. Finally, the steam flow phase creates pipe reaction forces, quencher thrust forces, structure thermal loads, and oscillating pool boundary loads due to steam jet condensation at the quencher.

For the ABWR, the feedwater line break (FWLB) and main steam line break (MSLB) cause dynamic loads in the suppression pool. As with the SRV discharge, these events can be characterized by several phenomena which occur in the following order: vent clearing, pool swell, high steam flow, and chugging. After sufficient pressurization of the drywell due to an FWLB or MSLB, water in the vents is forced out into the pool. This vent water clearing causes submerged jet induced loads on nearby structures and the pool basemat. After vent clearing, air and steam bubble flow out the vents is initiated. The air component, originating from the drywell air, expands in the pool causing a rise in pool surface level which is called pool swell. Pool swell imposes loads on submerged structures and pool boundaries. After pool swell, a period of high steam flow occurs in which the steam is condensed in the pool vent exit area and no significant loads are imposed on the pool system. Later, as vent steam flow decreases, the steam condensation process causes a phenomena in which the vent exiting steam bubble first grows and then suddenly collapses creating oscillatory loads. This process is called chugging and imposes significant vent and suppression pool boundary loads.

The ABWR SRV discharge line exits into the suppression pool through X-quencher discharger devices. These discharge devices are generally designed to optimize heat transfer and stable condensation while minimizing pool boundary loads. However, it is not clear that the use of the X-quencher has additional benefits for the ABWR design. The X-quencher, consisting of a conical extension, capped plenum, and four arms with numerous small holes, is identical to the quencher used in the Mark II and Mark III designs. GE stated that it will calculate quencher discharge loads using the same methodology and test data that was previously used in similar analyses for Mark II and Mark III plants. These loads will be calculated after the exact SRV discharge piping arrangement is finalized. They will include pool boundary pressure from both single and multiple valve actuation. Based on test data that has shown that full steam flow is completely condensed in the pool without imposing any significant loads on the pool, no calculations of pool loads will be performed after the SRV is passing a steady flow of steam into the pool. GE should specifically address the SRV loads which result from these valves re-opening a second time before the SRV tailpipe is cooled and completely vented. In addition, GE should consider suppression pool temperature limits (NUREG-0763 or NUREG-0773) in analyzing steady state SRV steam flow conditions. This is an open item.

As previously discussed in this section, LOCA (i.e., FWLB or MSLB) loads comprise the following components: pool boundary, access tunnel (i.e., a submerged structure in the pool), impact and drag, diaphragm floor, steam condensation oscillation, and chugging. GE calculated the pool boundary, access tunnel, impact and drag, and diaphragm floor loads using analytical models that simulate the thermal-hydraulic phenomena that affect these loads. Steam condensation oscillation and chugging were analyzed using tests that were conducted using specific (but reduced scale) ABWR vent geometry and thermodynamic conditions. However, GE should present specific data which compares a complete set of test conditions to actual ABWR conditions. This is an open item.

The pool boundary load is due to pool swell which is caused after the vent clearing process when air is expelled out the vent. GE used the same model that was developed and used for Mark II and Mark III containments. This model includes the following conservative assumptions:

- o Non-condensable gases are treated as ideal gases.

- o After vent clearing, only non-condensable gases (NCG) flow out the vent; their flow rate is calculated as one-dimensional adiabatic, with pipe friction effects.

- o Initial drywell NCG are compressed isotropically and taken at the drywell temperature when modeled as bubbles in the suppression pool water.

- o After vent clearing, constant thickness pool water above the vent outlet is accelerated upward while neglecting friction and fluid viscosity.

- o Pool wetwell NCG undergo polytropic compression during pool swell with a polytropic index of 1.2 for swell height and 1.4 for wetwell pressurization.

- o Pool swell velocity is multiplied by a factor of 1.1 and effective pool surface area is 0.8 times actual pool surface area.

The above assumptions all maximize the pool swell loads on the suppression pool. GE should clarify which, if any, of these assumptions are unique to the ABWR design. This is an open item.

Two partially submerged personnel and equipment access tunnels run through the pool and are subject to drag load, air bubble pressure loading, and buoyancy loading during pool swell. GE has presented its methodology for calculating these loads which relies on three fundamental equations, one for each load. In addition, structures above the normal pool surface are subjected to impact and drag loads due to pool swell. GE presented its calculation methods for calculating these loads. This method includes an additional 35 percent factor on the maximum impact pressure load and the previously mentioned 1.1 multiplier on the swell velocity. A potential for upward differential pressure loads on the diaphragm floor exists during pool swell. However, analytical results have indicated that the wetwell pressure will not exceed the drywell pressure and, therefore, there will only be a downward differential pressure load on the diaphragm floor. GE should explain whether this methodology, and its associated factors, constitute a conservatism or accommodation for the ABWR, or merely represent carryover from previous Mark II and Mark III design analyses. This is an open item.

Steam condensation loads, unlike vent clearing and pool swell loads, have required sub-scale (SS) and partial full-scale (FS) tests which specifically simulate ABWR vent and pool geometry characteristics as well as thermodynamic conditions. A total of 24 separate blowdown tests were conducted, 11 FS and 14 SS, to confirm the condensation oscillation (CO) and chugging (CH) loads due to a LOCA in the ABWR. Of the 24 tests, 13 were for CO behavior and 11 for CH behavior. These tests included both steam and liquid line breaks and a range of initial drywell and wetwell thermodynamic conditions. These tests were instrumented at seven pool boundary locations for dynamic pressure along with structural instrumentation on the basemat, pedestal, and containment walls. GE did not provide sufficient test details and analysis to determine differences between the test and actual ABWR design features. The staff noted that scaling laws for condensation have not been previously accepted, thus they require further justification. This is an open item.

For CO phenomena, the tests provided data on pressure amplitudes and frequencies as well as other structural loads. GE states that these tests also confirmed scaling factors that were used in the test facility and measured results. However, as previously discussed, insufficient evidence is provided to support this assertion, especially in the light of the historical lack of acceptability of condensation scaling. The CO tests confirmed the expected ABWR CO behavior and defined the CO load on the ABWR pool. Using this test data, an alternate formulation of CO load was developed which is termed "Source Load Approach." This method, which has been previously used, simulates the actual CO load by imposing a series of oscillations in the pool which produce the same or greater loads on the pool.

The 11 CH behavior tests provided data on peak overpressure and associated amplitude data as well as information on the frequency and periods of pressure pulses associated with CH. It was found that a lower initial wetwell pool temperature resulted in a higher CH peak overpressure. The CH test data was also used to develop a source load for this phenomena. The CH tests also provided load data for the access tunnels and horizontal vents.

The final area of containment hydrodynamic loads which was evaluated by GE is submerged structure loads which are caused by either LOCA or SRV injection of fluids (i.e., air, water, and steam) into the suppression pool. Loads on submerged structures in the suppression pool can be induced by pool swell, condensation oscillations, chugging, and SRV discharge. GE has stated that it will use the same methodology for these loads as it has previously used for other BWR designs.

ABWR SAR Appendix 3B provides a detailed discussion of all the phenomena which could induce hydrodynamic loads on the ABWR containment. The methodology for calculating these loads, which is identical to that used for Mark III containments, is also presented. For steam condensation loads (e.g., condensation oscillations and chugging), 24 tests with sub-scale and partial full-scale facilities were conducted to measure suppression pool loads. GE claims that these tests confirmed scaling factors and provided data for an alternative source load methodology. With the exception of these test results, GE did not perform and present actual hydrodynamic load calculations for the ABWR pending the final design of the vent and SRV piping routing into the suppression pool.

Although the overall methodology for calculating hydrodynamic loads for the ABWR is similar to that used and approved for the Mark II and Mark III containments, this fact does not necessarily imply that this is acceptable for the ABWR. GE needs to provide the results of actual ABWR hydrodynamic load calculations; thus demonstrating that these loads are acceptable compared to design values. In addition, GE has not provided any justification for performing only 24 blowdown tests for the ABWR design whereas over 200 tests were conducted to verify the adequacy of the Mark III containment design. The test condition details need to be presented and compared to actual ABWR suppression pool design features and thermodynamic conditions. Specific modeling details for analyzing the ABWR loads need to be presented in terms of their applicability to the ABWR. The discussion of model assumptions is not complete. A . model assumptions need to be delineated and compared to Mark II, Mark III, and ABWR conditions. GE needs to provide detailed technical justification for the acceptability of scaling laws when applied to condensation loads. Two additional aspects of suppression pool loads which were not included in Appendix 3B, but need to be, are additional SRV actuation before the tailpipe is cooled and vented completely and the margin to suppression pool temperature limits during steady state SRV steam discharge.

6.2.1.7 Subcompartment Pressure Analysis

Internal structures within the drywell and wetwell form subcompartments or restricted volumes that are subjected to differential pressure subsequent to postulated pipe ruptures. In the drywell there are two such volumes: (1) the reactor pressure vessel annulus, which is the annular region formed by the reactor pressure vessel and the biological shield, and (2) the drywell head, which is a cavity surrounding the reactor pressure vessel head. There is also a main steam tunnel located in the drywell.

The design of the containment subcompartments was based on the postulated worst-case design-basis accident (DBA) occurring in each subcompartment. For each containment subcompartment in which high-energy lines are routed, mass and energy release data corresponding to a postulated line break were calculated. All breaks were considered to be full double-ended circumferential breaks.

In response to RAI Question 430.17 regarding subcompartment pressurization from high energy line breaks, GE submitted SSAR Tables 6.2-3, 6.2-4 and Figures 6.2-37a and 6.2-37b. These tables and figures present subcompartment node and vent path initial conditions, break conditions, and physical characteristics as well as a flow chart showing the volume and junction connections between each subcompartment. GE modelled a total of 23 subcompartments connected with 35 separate flow path vents for the subcompartment analysis. Most of the vents are blowout panels which have a characteristic opening pressure and time. The subcompartments enclose some compartments of the RHR, RCIC, ECCS, RWCU, main steam, and main turbine systems. GE presented the calculated peak differential pressure for each subcompartment in SSAR Table 6.2-3.

The staff evaluated the aforementioned information in accordance with the requirements and guidance set forth in the Standard Review Plan: RG 1.70, Rev. 3, Section 6.2.1.2, "Subcompartment Analysis." Based on its review, the staff concluded that the following additional information is required to adequately assess the ABWR subcompartment analysis:

1. mass and energy release rates assumed for the subcompartment analyses,

2. methodology (i.e., computer codes, if any used in calculating subcompartment pressurization,

3. nodalization sensitivity studies for the individual subcompartments to justify the final model,

4. basis for selecting subcompartment initial thermodynamic conditions, and

5. individual subcompartment design pressure differential.

This is an open item.

Within the limitations of the available information, the staff made the following additional observations:

1. The selected subcompartment initial humidity specified in SRP 6.2.1.2 Section II.B.1 is 0 percent. Due to the ability of water vapor to absorb more energy than dry air, a humidity level of 0 percent results in a maximum peak differential pressure during a high energy line break in a subcompartment. In GE's analysis, a higher value for initial humidity is used.

2. Based on subcompartment volume and relief vent properties, the trend of calculated peak differential pressure for rooms with the same pipe break was analyzed. A number of calculated subcompartment peak pressures do not follow the basic trend that is expected, i.e., for the same pipe break, peak pressure should increase with smaller room volume and/or smaller vent area. The subcompartments with questionable peak pressures include: SA7, SA4, SR5, SR4, and SR9.

3. Using the COMPARE MODE 1A computer code, subcompartment and vent properties from ABWR Tables 6.2-3 and 6.2-4 and main steam line break mass and energy (M&E) release data from ABWR Figures 6.2-24 and 6.2-25, the staff performed a review calculation for the pressurization of rooms SS1 and ST1 (steam tunnel and turbine

building). This analysis resulted in significantly different pressures for the steam tunnel and for the turbine building than are reported in the SSAR.

The observations identify inconsistencies in subcompartment peak pressure trends, subcompartment pressures and analytical assumptions. These differences, when considered collectively, may result in a less conservative structural design of containment subcompartments. Further staff discussion with GE is needed to assess the impact observations. This is an open item.

6.2.1.8 Steam Bypass of the Suppression Pool

The concept of the ABWR pressure-suppression containment is that steam released from the primary system will be condensed by the suppression pool and will therefore limit pressurization of the containment system. This is accomplished by channeling the steam into the suppression pool through a vent system. Bypass leakage paths could exist between the drywell and the wetwell airspace that might over-pressurize the containment. Potential sources of steam bypass include leakage through the vacuum relief valves, cracking of the drywell concrete structure, and penetrations through the drywell structure.

The ABWR containment structure design includes a steel liner on its primary containment boundary to minimize drywell-to-wetwell leakage. The only fluid lines that transverse the wetwell air space are the SRV discharge lines. These lines will be designed to preclude rupture and will not be equipped with guard pipe. To mitigate the consequences of any steam which may bypass the suppression pool, the wetwell spray system will be manually activated. The flow rate of the wetwell spray system is 500 gpm.

The allowable bypass leakage is defined as the amount of the steam which could bypass the suppression pool without exceeding the wetwell design pressure. The allowable value has been evaluated by the applicant for the complete spectrum of credible primary system pipe ruptures. It is expressed in terms of the parameter $(A/K^{(1/2)})$,

where:

A = flow area of leakage path (ft^2)
K = geometric and friction loss coefficient

The parameter $(A/K^{(1/2)})$ is dependent only on the geometry of the drywell leakage paths and is a convenient numerical definition of the overall drywell leakage capability.

GE evaluated the bypass capability of the primary containment for small primary system breaks, considering containment sprays and containment heat sinks as means of mitigating the effects of bypass leakage. GE stated, in response to the staff's RAI, that while large primary system ruptures generate high pressure differentials across the assumed leakage path and, therefore, high leakage flow rates, the large breaks also rapidly depressurize the reactor and terminate the blowdown. Small breaks, however, result in an increasingly longer reactor blowdown period that increase the duration of the leakage flow. Therefore, GE determined that the limiting case is a very small reactor system break which will not automatically result in reactor depressurization. The applicant's analysis resulted in an allowable drywell leakage capability $(A/K^{(1/2)})$ of 0.05 ft^2, which is identical to that for the Mark II design.

When the staff approved the leakage capability of the Mark II design, it recognized that the containment had not been designed with suppression pool bypass as a consideration. In light of that fact, the staff accepted the 0.05 ft^2 value for the Mark II design. However, the ABWR containment is a combination of the Mark II and Mark III designs and the BWR pressure suppression design is sensitive to relatively small bypass leakage areas. Additionally, as discussed in Section 19.6.4.2.1 of this DSER, the results of the staff's PRA review indicate that the amount of suppression pool bypass has a major influence on the CET results. GE should demonstrate that the ABWR is able to accommodate a spectrum of breaks, from small to large. The staff considers that the limiting case will be a drywell-to-wetwell leakage rate that is slightly above that which the wetwell sprays can accommodate. As a goal, therefore, advanced BWR's that use the pressure suppression design concept should demonstrate a capability to accommodate bypass leakage equivalent to that resulting from the single failure of one vacuum relief penetration. For the aforementioned reasons, the staff concludes that a drywell leakage capability of 0.05 ft^2 for the ABWR is unacceptable. This is an open item.

6.2.2 Containment Heat Removal System

The containment heat removal system is an integral part of the residual heat removal (RHR) system which consists of three redundant loops. Each loop is designed so that a failure in one loop cannot cause a failure in another. In addition, each of the loops and associated equipment is located in a separate protected area of the reactor building to minimize the potential for single failure, including loss of onsite or offsite power causing the loss of function of the entire system. The system equipment, piping, and support structures are designed to seismic Category I criteria.

The containment heat removal system encompasses several of the RHR operating modes, which are the low pressure flooder (LPFL) mode, the suppression pool cooling mode, and the containment (drywell and wetwell) spray modes.

a) LPFL Mode

Following a LOCA, containment cooling starts as soon as the LPFL injection flow begins. During this mode, water from the suppression pool is pumped through the RHR heat exchangers and injected into the reactor vessel. The LPFL mode is initiated automatically by a low water level in the reactor vessel or high pressure in the drywell. In addition, each loop in the RHR system can also be placed in operation by means of a manual initiation push button switch.

b) Suppression Pool Cooling Mode

Following a LOCA, the suppression pool cooling subsystem provides a means to remove heat released into the suppression pool. During this mode of operation, water is pumped from the suppression pool through the RHR heat exchangers and back to the suppression pool. This mode is initiated, as needed, manually, by closing the LPFL injection valves and opening the suppression pool return valves. In response to an RAI, GE indicated that the heat removal function would be initiated within 10 minutes following a LOCA. The staff found this to be sufficiently conservative and adequate to achieve the necessary containment cooling function.

c) Containment (Wetwell and Drywell) Spray Cooling Mode

Two of the RHR loops provide containment spray cooling subsystems. Each subsystem provides both wetwell and drywell spray cooling. This subsystem provides steam condensation and primary containment atmospheric cooling following a LOCA by pumping water from the suppression pool, through the RHR heat exchangers and into the wetwell and/or drywell spray spargers in the primary containment. The drywell spray mode is initiated by operator action as needed following a LOCA by closing the LPFL injection valves and opening the spray valves.

Provisions have been made in the RHR system to permit inservice inspection of system components and functional testing of active components.

The location of suction and return lines in the suppression pool facilitates mixing of the return water with the total pool inventory before the return water becomes available to the suction lines.

RG 1.1, "Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Sumps," prohibits design reliance on pressure or temperature transients expected during a LOCA for ensuring net positive suction head. The ABWR net positive suction head design assumes 0-psig containment pressure and the maximum expected fluid temperatures resulting from a LOCA and, therefore, is acceptable.

The suppression pool make-up system provides additional water from the condensate storage tank through the suppression pool cleanup system to the suppression pool by gravity flow during normal conditions. Following a LOCA, the ECCSs take suction from the suppression pool. The quantity of water is sufficient to account for all conceivable post-accident entrapment volumes (i.e., places where water can be stored while maintaining long-term drywell vent water coverage).

Based on its review of the information in the ABWR SSAR and the responses to the staff's RAIs concerning the containment heat removal systems, the staff concludes that the containment heat removal systems satisfy the guidelines described in SRP Section 6.2.2, "Containment Heat Removal Systems" and RG 1.1, and are, therefore, acceptable.

6.2.3 Secondary Containment Functional Design

The ABWR secondary containment region completely surrounds the primary containment and is designed to remove fission products released from the primary containment during a DBA to limit whole body and thyroid doses within the guidelines of 10 CFR Part 100 and 10 CFR Part 50 Appendix A General Design Criterion (GDC) 19. The components of the secondary containment are designed to withstand missiles, pipe whip, post accident environments, seismic events, a single active failure, and a loss of offsite power. The two systems that fulfill this function are the secondary containment heating, ventilating, and air conditioning (HVAC) and the standby gas treatment system (SGTS).

The HVAC maintains a negative pressure within the secondary containment during normal operation to prevent any radioactivity from escaping to the environment. The SGTS provides post-accident filtration and removal of airborne halogens and particulates from the secondary containment. The SGTS is designed to maintain at least -0.25 inches water gage negative pressure (secondary containment to environment) after any postulated accident. GE indicates that testing and inspection of the integrity of secondary containment will be part of the testing of the SGTS. The staff's evaluation of the SGTS is discussed in Section 6.5.3 of the DSER.

SRP Section 6.2.3, "Secondary Containment Functional Design," in part, indicates that all openings, such as personnel doors and equipment hatches, should be under administrative control. These openings should be provided with position indicators and alarms having readout and alarm capability in the main control room. The effect of open doors or hatches on the functional capability of the depressurization and filtration system should be evaluated. In response to RAI Question 430.34 regarding this issue, GE provided SSAR Table 6.2-9 which lists all secondary containment penetrations along with their elevation and diameter. In their response, however, GE did not address the staff concerns delineated above. This is an open item.

Based on its review, additional information is required to allow the staff to determine the acceptability of the secondary containment functional design.

6.2.3.1 Secondary Containment Bypass Leakage

Although the primary containment is enclosed by the secondary containment, there are systems that penetrate both the primary and secondary containment boundaries, creating potential paths through which radioactivity in the primary containment could bypass the leakage collection and filtration systems associated with the secondary containment. A number of the lines contain physical barriers or design provisions that can effectively eliminate leakage. These include water seals, containment isolation provisions, and vent return lines to controlled regions. The criteria by which potential bypass leakage paths are determined has been set forth in Branch Technical Position CSB 6-3, "Determination of Bypass Leakage Paths in Dual Containment Plants."

In RAI Question 430.33, the staff requested that GE provide additional information to justify the bypass leakage path barriers that are relied upon to preclude bypass flow. In their response, GE indicated that only valve leakage could bypass the secondary containment, and Type C containment leakage tests on the outboard containment isolation valves will be used to monitor this leakage. In response to related RAI Question 430.52c, GE stated that 140 SCFH is considered to be the bypass leakage rate through the MSIVs. GE provided the information in accordance with RG 1.70, Revision 3 in Table 6.2-10 of the ABWR SSAR. In addition, GE specifically addressed the guidelines which are specified in BTP 6-3 in the response to Question 430.50a.

Based on its review of the SSAR and GE's responses to the staff's RAI's, the staff concluded that GE adequately addressed the criteria described in BTP CSB 6-3 and that the design of barriers to preclude bypass flow is acceptable.

6.2.4 Containment Isolation System

The containment isolation system includes containment isolation valves and associated piping and penetrations necessary to isolate the primary containment in the event of a LOCA. The staff's review of this system considered the number and location of isolation valves, valve actuation signals and valve control features, positions of the valves under various plant conditions, protection afforded isolation valves from missiles and pipe whip, and environmental design conditions specified in the design of components. The design requirements for the containment isolation system are based upon GDCs 54, 55, 56, and 57.

The piping systems of the ABWR that penetrate containment can be classified into three areas:

1. Piping lines that meet the explicit requirements of GDCs 54, 55, 56, and 57,

2. Piping lines that do not meet the explicit requirements of GDCs 54, 55, 56, and 57 but are acceptable based on their meeting the specific guidelines given in SRP 6.2.4, which constitute acceptable alternatives design provisions, and

3. Other lines that must be reviewed on a case-by-case basis to determine if an acceptable alternative basis exists for allowing a deviation from the explicit GDC on grounds not previously articulated in the SRP.

During the course of its review, the staff requested GE to provide more detailed information with respect to the containment isolation system. Questions 430.31, 430.32, 430.34, 430.35, 430.36, 430.37, 430.39, 430.40, 430.41, 430.43, and 430.44 all involve issues affecting the containment isolation system design. GE has responded to all of these questions with the exception of 430.32. The GE response to these questions has been reviewed and found to be acceptable with the exception of 430.34 and 430.36.

In response to Question 430.34, GE provided a new Table 6.2-9 which lists all the secondary containment openings, but GE did not present any information on the instrumentation means by which each of these openings is assured to be closed during a postulated DBA. For Question 430.36, GE stated that all isolation valves are within the scope of the ABWR Standard Plan, but did not address the staff's request for information regarding essential and non-essential systems per RG 1.141 as well as non-essential system containment isolation requirements. Questions 430.32, 430.34, and 430.36 have not been satisfactorily resolved. This is an open item.

Although GE specifically commits to the requirements of GDCs 54, 55, 56, and 57, there is no commitment to GDCs 1, 2, 4, and 16 which is a requirement of NUREG-0800 (Standard Review Plan) 6.2.4. This is an open item. In addition, in response to Question 430.41, GE stated that instead of meeting the requirements of GDC 56 for the HPCS and RHR test and pump miniflow bypass lines, RCIC pump miniflow bypass line, RCIC turbine exhaust and pump miniflow bypass lines, and SPCU suction and discharge lines, the ABWR will use GE Safety Standard 20 No. 8 to No. 9. These standards do not meet the requirements of GDC 56, but instead provide less conservative criteria for containment isolation. This is not acceptable for the ABWR design. The ABWR design must conform to GDC 56 unless a more detailed justification is provided for this deviation. This is an open item.

In reviewing Table 6.2-7 which delineates containment isolation valve information in response to Question 430.35, some design features were identified which do not conform to SRP guidance for containment isolation. In this table, GE only states whether the normal position of each valve is "open" or "closed." There is no way to determine if closed is the same as locked closed as it is stipulated in SRP 6.2.4. Also, the isolation valve closure times of "< 30 seconds" for drywell atmosphere systems with relatively large diameter penetrations (e.g., atmospheric control system 22-inch valve T31-F004 and flammability control system 6-inch valve T49-F006A) are not justified by GE. SRP 6.2.4 requires a technical basis for the selection of the drywell atmosphere closure times which is based on radiological consequences analysis for the DBA. GE should provide information to clearly identify which valves are "open," "closed" and "locked closed." GE should also provide the technical basis for drywell atmosphere closure times. This is an open item.

6.2.4.1 Containment Purge System

Question 430.42 requested the design features of the ABWR which show conformance with Branch Technical Position (BTP) CSB 6-4, "Containment Purging During Normal Plant Operations." GE responded to this question and incorporated a new proprietary Section 9.4.5.6, "Containment Supply/Exhaust System" in the ABWR SSAR. Additional information is presented in SSAR Table 6.2-7 and on Figure 6.2-39a.

The containment purge supply and exhaust lines, connected to both the drywell and wetwell, consist of one supply and one exhaust penetration each for the drywell and wetwell. Both the purge supply and exhaust lines, each of which are connected to both the drywell and wetwell, have two parallel isolation valves which are located as close as possible outside of the primary containment. One valve (22 inch diameter) is used for high volume inerting and purging while the other valves (2 inch diameter) are used for necessary venting for pressure control during operation. All isolation valves are air operated, fail in the closed position, and are closed by high drywell pressure or Level III low reactor vessel water level. The large diameter valves are butterfly type valves with a closure time of less than 30 seconds. The small diameter valves are globe type and have closure times of less than 15 seconds. The above valve configuration does not comply with GDC 56, which requires one isolation valve inside and one isolation valve outside containment for each penetration.

A number of criteria which are delineated in BTP CSB 6-4 have not been addressed by GE. These areas include the following:

1. Radiological consequence analysis for a LOCA with the purge system initially open (BTP CSB 6-4, B.5.a).

2. System structural integrity design under LOCA thermal-hydraulic conditions (BTP CSB 6-4, B.5.b).

3. Design provisions to ensure that isolation valve closure is not prevented by debris entrained in escaping air and steam (BTP CSB 6-4, B.1.g).

4. ECCS backpressure containment pressure reduction analysis for a LOCA with an initially open purge system (BTP CSB 6-4, B.5.c).

5. Case-by-case purge isolation valve maximum allowable leak rate (BTP CSB 6-4, B.5.d).

6. Technical justification for purge system isolation valve closure time greater than the five seconds delineated in BTP CSB 6-4 Section B.1.f.

Based on the above review, the staff requires additional information from GE to complete its assessment of the ABWR containment purge system.

6.2.5 Combustible Gas Control in Containment

Following a LOCA, hydrogen may accumulate within containment as a result of the following phenomena: (1) metal-water reaction between the zirconium fuel cladding and the reactor coolant, (2) radiolytic decomposition of the water in the reactor core and containment, and (3) corrosion of metals by emergency core cooling and containment spray solutions. If a sufficient amount of hydrogen is generated, it may react with the oxygen present or generated in the containment following an accident. To monitor and control the buildup of hydrogen and oxygen within the containment, GE has incorporated the following systems and capabilities within the ABWR design: Atmospheric Control System, Containment Atmosphere Monitoring System, the capability of post-LOCA purging of the containment, and hydrogen recombiners.

(1) Atmospheric Control System (ACS)

The ACS is designed to maintain the primary containment oxygen concentration below the maximum permissible limit per RG 1.7 during normal, abnormal and accident conditions to assure an inert atmosphere. Inerting is accomplished with nitrogen storage tanks that are adequately sized and provided with make up capability.

The ACS is designed to withstand missiles, pipe whip, flooding, tornadoes, a safe shutdown earthquake, LOCA environment, and a single active failure. However, GE states that the ACS is non-safety grade, whereas the SRP Section 6.2.5 acceptance criteria regarding GDC 41 states that the combustible gas control system design should be safety-grade because this system is relied on to ensure that containment integrity is maintained following an accident. Based on this discrepancy, the staff has not been able to find the requirements for the design of the ACS acceptable. This is an open item.

(2) Containment Atmosphere Monitoring System (CAMS)

The CAMS is designed to monitor oxygen levels in the wetwell and drywell during accident conditions to confirm that the primary containment is inerted. The staff's evaluation of the CAMS is discussed in Section 7.0 of the DSER.

(3) Capability of Post-LOCA Purging of the Containment

Post-LOCA primary containment backup purging capability is provided in accordance with RG 1.7 and as an aid in containment atmosphere cleanup following a LOCA. During normal plant operation, the purge line also functions, in conjunction with the nitrogen purge line, to maintain primary containment pressure at about 0.75 psig and oxygen concentration below 4 percent by volume. This is accomplished by makeup of the required quantity of nitrogen into the primary containment through the makeup line or relieving pressure through the purge line. Flow through the bleed line will be directed through either the SGTS or the secondary containment HVAC and will be monitored for radiation release. However, GE has provided neither the purge rate that would be required to maintain the oxygen concentration below 4 percent by volume nor the radioactive consequences analysis for the staff to review. Based on its review, the staff has not been able to find the ABWR capability of post-LOCA backup purging of the containment acceptable. This is an open item.

(4) Hydrogen Recombiners

GE states that provisions are made for connection of two permanently installed recombiners in the secondary containment. However, GE has not provided information on dedicated redundant containment penetrations to demonstrate that the recombiners can perform their safety function assuming a single failure. Therefore, the staff has not been able to find that the design is acceptable. This is an open item.

With respect to the post-accident hydrogen generation analysis GE indicates that the analytical model described in GE report, NEDO-22155, "Generation and Mitigation of Combustible Gas Mixtures in Inerted BWR Mark I Containment," was used to compute the hydrogen and oxygen generation from radiolysis. The NEDO-22155 report is being reviewed by the staff for the EPRI requirements document certification. The staff will report its finding of this issue in the final safety evaluation for the ABWR.

Three questions on the combustible gas control system were transmitted to GE (430.45, 430.46, and 430.47). These questions dealt with the subjects of scope and interface, compliance with

RG 1.7, and BTP ASB 9-2 for hydrogen and oxygen production and accumulation. GE has responded to all of these questions and amended Section 6.2.5 of the SSAR. GE stated that the entire combustible gas control system is within the scope of the ABWR and that there are no outside system interfaces. GE also indicated that the LOCA hydrogen and oxygen production and accumulation were calculated using the appropriate section of RG 1.7, BTP ASB 9-2, and Section 6.2.5.3 of RG 1.70, Revision 3. The staff concluded that GE's responses and SSAR amendments adequately addressed the issues identified in the above referenced RAIs.

Based on the above review, the staff requires additional information from GE to complete its assessment of combustible gas control in the ABWR containment design.

6.2.7 Fracture Prevention Of Containment Pressure Boundary

The primary containment vessel of the ABWR is a reinforced concrete structure with ferritic parts (the removable head, personnel locks, equipment hatches and penetrations), which are made of material that has a nil-ductility transition temperature, RTNDT, of at least 30°F below the minimum service temperature. GDC 51 of 10 CFR Part 50 is only applicable to parts of containment that were made of ferritic materials.

The staff requested GE to clarify the applicability of GDC 51 because in the original SSAR it appeared that GDC 51 could be applicable to the concrete part of the containment (Q251.12). GE responded that GDC 51 is applicable to the removable drywell head, personnel locks, equipment hatches and penetrations which are made of ferritic materials. The staff concludes that GE has satisfactorily responded to the staff's RAI (Q251.12) and has revised Section 3.1.2.5.2.2 accordingly. Therefore, the design of the primary containment vessel complies with GDC 51.

6.2.8 Severe Accident Considerations

The containment performance in severe accidents is addressed in Chapter 19 of the ABWR SSAR. The staff review is documented in the corresponding Chapter of the DSER.

6.6 Inservice Inspection of Class 2 and 3 Components

The information in this section of the ABWR SSAR pertains to the design access, preservice inspection requirements and proposed methodology for inservice inspections of ASME Class 2 and 3 components. The staff's review of this section was based upon the requirements of 10 CFR Part 50.55a(g), and in accordance with SRP section 6.6.

SSAR section 6.6.1 contains a description of the Class 2 and 3 system boundaries. The SSAR describes systems, such as, the service air system, high pressure nitrogen gas supply system and the instrument air system, that are outside of the scope of ASME Section XI. The inservice inspection requirements for such systems are contained in the individual DSER sections that address the system function. GE stated its plans to use Regulatory Guide 1.26 for the Quality Groups B and C boundaries. The staff's evaluation of the classification of system boundaries is contained in DSER section 5.2.

The staff review of accessibility to ASME Class 2 and 3 components was based upon the requirements of 10 CFR Part 50.55a(g)(3)(ii), which states:

Components which are classified as ASME Code Class 2 and Class 3 and supports for components which are classified as ASME Code Class 1, Class 2, and Class 3 shall be designed and be provided with access to enable the performance of inservice examination of such components and shall meet the preservice examination requirements set forth in Section XI of editions of the ASME Boiler and Pressure Vessel Code and Addenda applied to the construction of the particular component.

GE addressed this requirement in Section 6.6.2 - Accessibility, which states:

"All items within the Class 2 and 3 boundaries are designed, to the extent practicable, to provide access for the examinations required by IWC-2500 and IWD-2500. Items for which the design is known to have inherent access restrictions are described in Subsection 6.6.9."

SSAR subsection 6.6.9 describes areas of the Class 2 and 3 vessel nozzle welds which may be inaccessible for ultrasonic examination. Based on the staff's review, "inherent access restrictions" are contrary to the requirements of 10 CFR Part 50.55a.

6.6.1 Examination Requirements

GDC 36, 39, 42, and 45 require, in part, that the Class 2 and 3 components be designed to permit appropriate periodic inspection of important components to ensure system integrity and capability. Section 50.55a(g) of 10 CFR 50 defines the detailed requirements for the PSI and ISI programs for light-water-cooled nuclear power facility components.

The design of the ASME Class 2 and 3 components may incorporate exclusions from examinations as defined in paragraph IWC-1220 and IWD-1220, respectively, of ASME Section XI. The systems or portions of systems that were excluded are specifically identified in SSAR Table 6.6.1.

The staff finds that the proposed exclusion criteria described in Table 6.6.1 is inconsistent with 10 CFR 50.55a(g)(3), in that exclusions from examination criteria shall comply with NRC regulations in effect at the time of construction of the particular component. The staff therefore concludes that exclusions from examination criteria should be addressed by utility applicants who reference the ABWR design. The staff therefore considers the identification of exclusions from inservice inspection of Class 2 and 3 components to be an interface requirement.

6.6.2 Evaluation of Compliance With 10 CFR 50.55a(g)

The staff concludes that the information in SSAR Section 6.6 is not in compliance with 10 CFR 50.55a(g) in that the inspection requirements reference a specific edition of ASME Section XI. GE should revise section 6.6 of the SSAR to include information which demonstrates that the inservice inspection of Class 2 and 3 components will comply with NRC regulations in effect at the time of construction of the particular component.

The staff notes, for example, that the examination methods described in SSAR Table 6.6-1 regarding ultrasonic testing are not consistent with the plans of the nuclear industry and the ASME Code for future inservice inspections. ASME Section XI has published Appendix VII "Qualification of Nondestructive Examination Personnel for Ultrasonic Examination" and Appendix VIII "Performance Demonstration for Ultrasonic Examination Systems." The NRC has published its intent in the Federal Register to reference in 10 CFR 50.55a(b) the ASME Section XI edition that includes the published Appendix VII and has established a technical contact to coordinate the implementation of Appendix VIII.

SSAR Section 6.6.7 Augmented Inservice Inspection addresses additional inspection requirements for all high energy piping between containment isolation valves. Based upon the staff's review, G.E. did not address the issue of erosion/corrosion-induced pipe wall thinning. From a design perspective, the ABWR should include design configurations and materials of construction that eliminate or minimize pipe wall thinning. However, in recognition of plans to revise ASME Section XI to include Subsection IWT "Requirements for Examination of Class 1, 2, and 3 Systems for Detection of Pipe Wall Thinning Due to Single-Phase Erosion-Corrosion," the staff concluded that GE should describe plans for inservice inspection of erosion/corrosion-induced pipe wall thinning as defined in NRC Bulletin 87-01. This is an open item.

Interface Requirement: Utility applicants who reference the ABWR design shall develop and submit for staff review the following plant-specific PSI/ISI program information:

(1) A complete and acceptable preservice inspection PSI/ISI program. The program must be in compliance with 10 CFR 50.55a(g)(3) and include reference to the edition and

addenda of ASME Section XI that will be used for the selection of components for examinations, lists of the components excluded from examination by the applicable code, and isometric drawings.

(2) Plans for preservice examination of the reactor pressure vessel welds to address the degree of compliance with RG 1.150, "Ultrasonic Testing of Reactor Vessel Welds During Preservice and Inservice Examinations."

The staff considers the above to be an interface requirement. The PSI/ISI will be evaluated before inservice inspection begins during the first refueling outage based upon the applicable ASME Code edition and addenda referenced in 10 CFR 50.55a(b).

6.6.3 Conclusions

Compliance with the preservice and inservice inspection requirements of the ASME Code and 10 CFR Part 50 constitutes an acceptable basis for satisfying the applicable requirements of GDCs 36, 39, 42 and 45. However the staff requires GE to address the aforementioned open items and interface requirements to allow the staff to determine the acceptability of section 6.6.

6.7 Main Steam Isolation Valve Leakage Control System

This system is not used on the ABWR.

6.8 High Pressure Nitrogen Gas Supply System

The ABWR design includes four compressed air systems: the Instrument Air System, the Service Air System, the High Pressure Nitrogen Gas Supply System, and the Atmosphere Control System. This section of the ABWR SSAR provided information on the design of the High Pressure Nitrogen Gas Supply (HPIN) System and was reviewed in accordance with SRP Section 9.3.1, "Compressed Air System."

The HPIN System is comprised of both non-essential (i.e., non-safety related) and essential systems. A single non-essential system provides continuous nitrogen supply to all pneumatically-operated components in the primary containment during normal operation. As noted in Section 6.5.2, during normal operation, the High Pressure Nitrogen Gas Supply System is supplied from the nitrogen gas evaporator/storage tank via the makeup line to the ACS.

The essential system is comprised of two independent divisions, with each division containing a safety-related emergency stored nitrogen supply capable of supplying 100 percent of the requirements of the division being serviced. Nitrogen gas for the essential system is supplied from high pressure nitrogen gas storage bottles. There are tielines between the non-essential and each division of the essential system. Each tieline has a motor operated shut-off valve.

Because the HPIN System is one of the four systems that perform functions addressed in SRP Section 9.3.1, the review of this system was performed as part of an integrated review of the ABWR compressed air systems. The results of this review are presented in Section 9.3.1 of this FSER.

8 ELECTRIC POWER SYSTEMS

8.1 Introduction

The primary bases for evaluating the adequacy of the Electric Power Systems for the GE-ABWR, Chapter 8 of the Standard Safety Analysis Report (SSAR) were the acceptance criteria and guidelines for Electric Power Systems contained in Table 8.1 of the United States Nuclear Regulatory Commission (NRC) Standard Review Plan-NUREG-0800, Revision 3-July 1983 (SRP).

8.2 Offsite Power System

Based on information presented on Figure 8.3-1 of Amendment 10 to the ABWR SSAR, it appears that the offsite power system consists of the following three sources:

1. A back feed from the transmission network through the main transformer, bus duct, and two unit auxiliary transformers to the Class 1E distribution system input terminals. To initiate this back feed, the main generator must be disconnected from this source by a generator breaker;

2. An offsite line from the transmission network through the reserve auxiliary transformer to the Class 1E distribution system input terminals; and

3. A combustion turbine generator to the Class 1E distribution system input terminals.

Section 8.2.3 of Amendment 10 to the SSAR indicated that these circuits, for the most part, are within the ABWR design scope; however, Section 3.1.2.2.8.2.2 of Amendment 10 to the SSAR indicated (and Sections 8.2.1 and 8.2.2 of Amendment 16 to the SSAR indicated) that these circuits are in total, out of the ABWR Standard Plant scope. Thus, description and analysis demonstrating compliance of the offsite circuits to regulatory requirements was initially not provided in Section 8.0 of the ABWR SSAR. By Amendment 17 to the ABWR SSAR, description and analysis were provided for the offsite circuits within the ABWR scope of supply.

To complete the staff's review of the offsite system in accordance with the criteria in the SRP, additional information is required for the following items:

1. The inconsistency between Sections 3.1.2.2.8.2.2, 8.2.1, 8.2.2, and 8.2.3 of the ABWR SSAR as to what part of the offsite system is within ABWR standard design scope.

2. The description of the offsite system in Section 8.1 versus 8.2 of the SSAR is not consistent with RG 1.70. Justification for this area of apparent non-compliance with the standard format for SSARs needs to be addressed.

3. Interface requirements for the offsite circuits that are outside the ABWR Standard plant scope.

4. Description and analysis of criteria relating to physical and electrical separation between the offsite circuits and between the offsite and the onsite Class 1E circuits.

5. Interface criteria relating to physical and electrical separation between offsite circuits and between offsite and onsite Class 1E circuits.

6. Physical lay out drawings which shows the physical separation of the offsite circuits and separation between onsite and offsite circuits. This shall include the instrumentation and control circuits associated with each offsit. circuit.

7. The physical and electrical separation between the circuits associated with the combustion turbine generator and other offsite circuits including instrumentation and control circuits.

8. Identification, analysis, and justification for each circuit or component part of the offsite system which will not be tested during normal plant operation.

9. Capacity and capability of each offsite circuit to supply nected loads.

10. Identification of SRP criteria applicable to offsite systems similar to Table 8.1-1.

8.2.1 Independence between Offsite and Onsite Systems

The following criterion, specified in Section 8.3.2.2.1 of Amendment 10 to the SSAR for the ABWR design, implies that a single failure of one 125 V DC system may jeopardize and thus cause loss of offsite and onsite power to one safety division but will not jeopardize or cause loss of offsite preferred alternating current (AC) power to any other safety divisions.

> "The unlikely loss of one 125 V DC system does not jeopardize the supply of preferred and standby AC power to the Class 1E buses of the other load groups."

This criterion implies that no single failure, ground fault, or other aberration in one offsite preferred circuit between the plant's switchyard and the Class 1E distribution system input terminals if caused by failure of one 125 V DC system will cause loss of offsite power to or challenge in any way more than one Class 1E AC distribution system, division, or load group. This criterion meets the requirements of GDC 17, and is, therefore, acceptable. However, the offsite system being proposed for the ABWR does not meet this criterion if, for example, the loss of a 125 V DC system causes:

a. Failure of the single main transformer supplying two of the safety divisions causing loss of offsite power to more than one safety division.

b. Failure of any one of the four unit auxiliary transformers causing loss of offsite power to more than one safety division.

To facilitate completion of the staff's review in this area, in accordance with Section 8.3.1 of the SRP, additional information is required for the following items.

1. The extent Class 1E direct current (DC) power is used for control and protection of the offsite circuits from the switchyard to the terminal connection on the Class 1E system.

2. Descriptive information or analysis demonstrating compliance of the ABWR design to the above quoted criterion.

3. Specific identification and documentation of the above and other exceptions to this criteria in the ABWR SSAR with justification.

8.2.2 Protective System for the Reactor Internal Pumps

Section 15.3.1.1.1 of Amendment 10 to the SSAR states that since four buses are used to supply power to the ten reactor internal pumps (RIPs), the worst single failure can only cause three RIPs to trip. Further down in this same section a statement is made that the probability of any additional RIP trips is low (less than 10^{-6} per year). Therefore, the event (i.e., the simultaneous trip of more than three RIPs) is classified as a limiting fault.

In order to establish that the probability of any additional RIP trips is less than 10^{-6}, additional information or analysis is required from GE in a SSAR amendment to address each of the following items.

(a) Probability analysis which demonstrates that a fault on the offsite circuit that occurs anywhere between and including the offsite switchyard and the reactor internal pumps will not cause loss of more than three reactor internal pumps (RIPs).

(b) Identify each component part of the power supply to the reactor internal pumps and/or protective systems that is expected to function to assure the assumptions used in the probability analysis of Item (a) above.

(c) Probability analysis which demonstrates that the combined probability of all events (including those described in Item (a) above is less than 10^{-6} for trip of more than three RIPs.

8.3 Onsite Power Systems

8.3.1 Compliance with General Design Criteria

Item (1)(b) of Section 8.3.1.2.2 of Amendment 10 to the SSAR indicated that the Class 1E Constant Voltage Constant Frequency (CVCF) power supply is in compliance with GDCs 2, 4, 17, and 18 in part or as a whole, as applicable. Response to Question 435.26 (also of Amendment 10) provided clarification that there are no non-compliances, but also indicated that some portions of the GDC's are not applicable at this level (for example, the statement in GDC 17 about two physically independent circuits from the transmission network). Based on the information presented, it was unclear as to what parts of these GDC's were considered not applicable to the CVCF power supplies. Also, it was unclear as to why two physically independent circuits from the transmission network were considered not applicable to the CVCF power supplies.

In discussions with the staff and in a draft submittal dated September 4, 1991, GE proposed modifying the response to Question 435.26 and Section 8.3.1.2.2 to indicate full compliance with the GDC's and to delete (1) certain conflicting statements in the SSAR, (2) the example of non applicability to GDC 17, and (3) the statement "the substance and intent of" from Section 8.3.1.4.2.1. Contingent on documentation of these proposed modifications, concerns addressed in requests for additional information (RAIs) are considered resolved with the exception of the two items described below which remain open until additional information or clarification is provided by GE.

- o Inconsistencies within Table 8.1-1 and between Table 8.1-1 and Section 8.1.3.1.2 as to applicable SRP criteria.

- o Clarification of the systems or components to which the Institute of Electrical and Electronics Engineers (IEEE) Standard 279, "Criteria for Protection Systems for Nuclear Power Generating Stations "(1971) applies. (Reference: Item 11 of Section 1.2.1.1.2)

8.3.1.1 Compliance with GDCs 2 and 4

Chapter 8 of the ABWR SSAR Amendment 10 contained the following statements in relation to the compliance of electrical system design to the requirements of GDC 2, "Design Bases for Protection Against Natural Phenomena," and GDC 4, "Environmental and Missiles Design Bases," of Appendix A to 10 CFR Part 50. It appeared during the staff review that each of these statements can be incorrectly interpreted to mean that protection need only be provided for two of the three (or four) independent safety related electrical divisions.

> "In some instances spacial separation is provided such that no single event may disable more than one of the redundant divisions or prevent safe shutdown of the plant."

> "Electrical equipment and wiring for the Class 1E systems which are segregated into separate divisions are separated so that no design basis event is capable of disabling more than one division of any engineered safety features (ESF) total function." (Reference: SSAR Section 8.3.1.1.5.1)

> "Redundant parts of the system are physically separated to the extent that a single credible event...can not cause loss of power to redundant load groups." (Reference: Section 8.1.3.1.1.1)

"Where spatial separation cannot be maintained in hazardous areas (e.g., potential missile areas), physical isolation between electrical equipment of different divisions is achieved by use of a 6-inch minimum thickness reinforced concrete barrier." (Reference: Section 8.3.1.4.1)

"Class 1E electric equipment and wiring is segregated into separate divisions so that no single credible event is capable of disabling enough equipment to hinder reactor shutdown, removal of decay heat from the core, or isolation of the containment in the event of an accident." (Reference: Section 8.3.1.4.1.1)

"Equipment arrangement and/or protective barriers are provided such that no locally generated force or missile can destroy any redundant reactor protection system (RPS), nuclear steam supply system (NSSS), emergency core cooling system (ECCS), or ESF functions. In addition, arrangement and/or separation barriers are provided to ensure that such disturbances do not affect both high pressure core flooder (HPCF) and reactor core isolation cooling (RCIC) systems." (Reference: Section 8.3.1.4.1.1)

"Containment penetrations will be so arranged that no design basis event can disable cabling in more than one division." (Reference: Section 8.3.1.4.2.3.2.(7)

"The protection system and ESF control logic, and instrument panels/racks shall be located in a safety class structure in which there are no potential sources of missiles or pipe breaks that could jeopardize redundant cabinets and raceways." (Reference: Section 8.3.1.4.2.2.3)

"In any compartment containing an operating crane...there must be a minimum horizontal separation of 20 feet or a 6 inch thick reinforced concrete wall between trays containing cables from different divisions." (Reference: Section 8.3.1.4.2.2.2(3))

"In rooms or compartments having heavy rotating machinery...or in rooms containing high-pressure feed water piping or high pressure steam lines...minimum separation of 20 feet or a 6-inch thick reinforced concrete wall is required between trays containing cables of different divisions." (Reference: Section 8.3.1.4.2.2.2(1))

Based on a review of the above statements, it appeared that the provision of barriers between redundant safety divisions (versus barriers from the effects of a credible event such as a locally generated missile) is the design basis for electrical systems meeting the protection requirements of Criteria 2 and 4 of Appendix A to 10 CFR Part 50. The design basis for protection of safety systems is not clear. It is not clear that following any design basis event with any resulting loss of equipment and single failure, sufficient remaining safety systems will be available to effect a safe plant shutdown for all allowable modes of plant operation. To resolve this issue, clarification is required in the ABWR SSAR as to the design basis for protection of safety systems.

8.3.2 Physical Independence

8.3.2.1 Conduits to open trays

Section 8.3.1.4.2.3.1 and response to Question 435.35 of Amendment 10 to the SSAR indicated that physical separation, for conduits containing scram solenoid group circuit wiring, will have a minimum separation distance of one inch from either metal enclosed raceways or non-enclosed raceways. The one inch of separation between a conduit and enclosed raceways complies with RG 1.75 separation guidelines and is therefore acceptable. The one inch of separation between a conduit and non-enclosed raceways, however, does not comply with separation guidelines of RG 1.75. The staff was, therefore, concerned that the proposed one inch of separation may not provide sufficient independence between redundant systems and/or protection to safety systems in accordance with the requirements of Criterion 17 of Appendix A to 10 CFR Part 50.

In the draft submittal dated September 4, 1991, GE proposed changing the one inch of separation to 1 foot horizontally and 3 feet vertically or providing a fire barrier. This item is considered resolved contingent on documentation of the above information in the SSAR.

8.3.2.2 Containment Penetrations

<u>Separation Between Class 1E to Class 1E Penetrations</u>

Item (7) of Section 8.3.1.4.1.2 of Amendment 10 to the SSAR indicated that electric penetration assemblies of different Class 1E divisions are separated by distance, separate rooms or barriers, and/or location on separate floor levels. Separate rooms or barriers and/or location on separate floor levels exceeds separation guidelines for penetrations and is acceptable. Separation by distance may also meet separation guidelines; however, information as to what constitutes the minimum allowable distance between penetrations had not been clearly defined in Amendment 10 to the SSAR.

In the draft submittal dated September 4, 1991, GE indicated that the minimum allowable distance between penetration assemblies of different Class 1E divisions is 3 feet horizontal and 5 feet vertical and is only allowed within the inerted containment. Outside containment separation is by separate rooms or barriers and/or location on separate floor levels. Section 8.3.1.1.5 of

Amendment 10 to the SSAR also stated that penetration assemblies are located around the periphery of the containment and at different elevations to facilitate reasonably direct routing to and from the equipment. The proposed design meets the guidelines of Section 5.5 of IEEE Standard 384-1974, "Criteria for Independence of Class 1E Equipment and Circuits," as endorsed by RG 1.75 and is therefore acceptable contingent on documentation of GE's proposed separation criteria in the SSAR and resolution of the following:

a. Identification, justification, and approval of the differences between the 1974 and 1981 versions of IEEE Standard 384 as they relate to containment electric penetration assemblies.

b. Clarification of what is meant by barriers as used above or where they are defined.

Separation Between Class 1E to Non-Class 1E Penetrations

In the draft submittal dated September 4, 1991, GE indicated in response to Question 435.33 that the minimum separation between penetrations containing non-Class 1E circuits and penetrations containing Class 1E or associated Class 1E circuits is by separate rooms, or barriers, or different floor levels outside containment and 3 feet horizontal and 5 feet vertical distance inside the inerted containment. The proposed design meets the guidelines of Section 5.5 of IEEE Standard 384-1974 as endorsed by RG 1.75 and is therefore acceptable contingent on documentation of the above design information in the SSAR and resolution of the following:

a. Identification, justification, and approval of the differences between the 1974 and 1981 versions of IEEE Standard 384 as they relate to containment electric penetration assemblies.

b. Clarification of what is meant by barriers as used above or where they are defined.

c. Inclusion of the proposed separation criteria contained in response to Question 435.31(a) in Section 8.0 of the ABWR SSAR.

Separation Between Class 1E Penetrations to Non-Class 1E Cables or to Other Divisional Cables

In the draft submittal dated September 4,1991,in response to Question 435.33, GE indicated that the minimum separation between penetrations containing Class 1E circuits and other divisional or non divisional cables is by separate rooms, or barriers, or different floor levels outside containment and 3 feet horizontal and 5 feet vertical distance inside the inerted containment. The proposed design meets the guidelines of Section 5.5 of IEEE Standard 384-1974 as endorsed by RG 1.75 and is therefore acceptable contingent on documentation of the above design information in the SSAR and resolution of the following:

a. Identification, justification, and approval of the differences between the 1974 and 1981 versions of IEEE Standard 384 as they relate to containment electric penetration assemblies.

b. Clarification of what is meant by barriers as used above or where they are defined.

c. Inclusion of the proposed separation criteria contained in response to Question 435.31(a) in Section 8.0 of the ABWR SSAR.

8.3.2.3 Class 1E Equipment

Section 8.3.1.1.5.1, Physical Separation and Independence, of Amendment 10 to the SSAR stated that divisional separation is achieved through the use of barriers, spatial separation, and totally enclosed raceways. This combination of methods for achieving separation meets the guidelines of Section 4.3 of IEEE Standard 384-1974 and is acceptable. GE in its draft submittal dated September 4, 1991, indicated that it had changed the separation to three hour fire rated barriers and totally enclosed raceways. This separation also meets 384-1974 and is acceptable.

Section 8.3.1.4 of Amendment 10 indicated that barriers (used to maintain divisional separation) are fire rated where feasible. Also Section 8.3.1.1.5.1 of Amendment 10 indicated that raceways embedded in concrete walls, ceiling, or floors will be used as barriers to maintain divisional separation. The use of fire rated barriers and embedded conduit meets the intent of IEEE Standard 384-1974 for separation of divisional cables and is acceptable. However, Section 8.3.1.4.2.2.2 of Amendment 10 to the SSAR indicated that there is an allowable exception to the combination of barriers, spatial separation, and totally enclosed raceways as the criteria for maintaining divisional separation. In plant areas with potential hazards (such as high-pressure feed water piping or high pressure steam lines) redundant raceways separated by 20 feet without barriers or being totally enclosed were allowed to be used to maintain divisional separation. Also, Item (9) of

Section 8.3.1.4.2.3.1 of Amendment 10 indicated that cables associated with the four redundant divisions of the start up range monitoring system and the two divisions of the rod control and information system located under the vessel would not use barriers, spatial separation, or totally enclosed raceways. However, Section 9A.5.5.5 of Amendment 10 to the SSAR indicated that flexible metallic conduit is allowed to be used on these cables under the vessel. To clarify or resolve these inconsistencies and to establish consistent separation criteria, additional information is required for the following items.

1. Clarification of the criteria to be used as the licensing and/or design basis for separation between (a) redundant divisional raceways (or cables) and (b) divisional or associated divisional and non-divisional raceways (or cables).

2. Identification of each exception to the licensing and/or design basis criteria for separation.

3. Detailed design description and analysis justifying each exception identified. For example, response to Question 435.35 in Amendment 10 of the SSAR stated that each scram conduit will be physically separated by at least one (1) inch from non-enclosed raceways. For any separation of 5 feet to one inch between a conduit and non-enclosed raceway the design does not meet separation guidelines of IEEE Standard 384-1974 and must be identified as an exception and justified by analysis.

8.3.2.4 Cables in Cabinets/Panels

Section 8.3.1.1.5.1 of Amendment 10 to the SSAR stated that divisional cables to and from the containment and to and from the dedicated divisional equipment in the reactor building are routed in separate cable raceways for each division. Section 8.3.1.1.5.1 further stated that divisional cable routing is maintained up to the terminal cabinets in the main control room. This statement implied that separate cable raceways for each division may not be maintained within cabinets and implied that non safety cables may be routed in the same raceway with divisional cables within cabinets or that redundant divisional cables may be routed in the same raceway within cabinets. This statement contradicted other sections of the ABWR SSAR which require separate raceways from terminal to terminal including inside of cabinets or other types of enclosures.

GE has indicated that the above described inconsistency between Sections 8.3.1.1.5.1 and 8.3.1.4.2.2.3 as to required separation between redundant circuits within a cabinet will be corrected by a revision to the SSAR as was reflected in the draft submittal dated September 4, 1991. This item is considered resolved contingent on clarification of the SSAR as to separation in panels throughout the plant, documentation in the SSAR of the information included in the draft submittal discussed above, and the clarification of the following listed items:

1. Criteria for separation between safety (or associated) and non safety cables and between divisional cables within cabinets or any other type of enclosure located inside and outside the main control room.

2. Marking of cables inside of cabinets and/or panels (Reference: Section 8.3.1.3.2.1(3)).

In addition, Section 8.3.1.4.2.2.3 of Amendment 10 to the SSAR included the statement that the purpose of criteria for physical separation of cables in panels is to preclude the possibility of fire propagating between redundant circuits and preventing safe shutdown of the plant. The staff felt that this statement of purpose may be misleading in that it does not fully delineate the requirements of GDCs 2, 4, and 17. The purpose for physical separation is to preclude failure of non-safety circuits from causing failure of any safety circuit and to preclude failure of one safety circuit from causing failure of any other redundant safety circuit (i.e., to preclude common cause failure of safety circuits). Draft information included in the submittal dated September 4, 1991, revised the SSAR to be consistent with the above purpose for separation criteria. This item is considered resolved contingent on clarification of the text in the SSAR.

8.3.2.5 Associated circuits

Section 8.3.1.1.5.1, Physical Separation and Independence, of Amendment 10 to the SSAR stated, in part, that associated cables are treated as Class 1E circuits. The staff interprets this statement to mean that associated cables or circuits will meet all requirements placed on Class 1E circuits. All components in the associated circuit's current loop (loads, cables, connectors, switches, relays, protective devices, etc.) will meet Class 1E requirements.

In the draft submittal dated September 4, 1991, GE indicated that the ABWR design does not currently have any known associated circuits but that the criteria for associated circuits is in place in the ABWR SSAR. The criteria in the ABWR SSAR implies (by reference to RG 1.75) but does not explicitly state in the SSAR that associated circuits will meet all requirements placed on Class 1E circuits. Therefore, it was the staff concern that the criteria for associated circuits may be misinterpreted.

GE indicated in discussions with the staff that they would revise the criteria to specifically address the guidelines of Position 4 of RG 1.75 in the appropriate sections of the SSAR. This item is considered resolved contingent on an acceptable revision of the SSAR.

8.3.2.6 Cable/Raceway Identification

In regard to marking of cables and raceways, response to Question 435.29 of Amendment 10 to the SSAR indicated that the identification criteria specified in Sections 8.3.1.3.1 and 8.3.2.3.2.1 of Amendment 10 to the SSAR fully complies with the requirements of RG 1.75 (revision 2) and IEEE 384-1974. The staff reviewed this criteria with respect to the guidelines of Positions 10 and 11 of RG 1.75 (revision 2) and Section 5.1.2 of IEEE 384-1974 and as a result identified a number of concerns. To resolve these concerns, additional information is required for the following items.

1. The method for color coding power, instrumentation, and control cables and raceways.

2. The method for distinguishing between Non-Class 1E circuits associated with different redundant divisions.

3. The staff concern relating to the durability of markings is considered resolved contingent on documentation of draft information provided on September 4, 1991.

8.3.2.7 Cables Approaching and/or Exiting Cabinets/Panels

The response to Question 435.30 of Amendment 10 of the SSAR stated that cable spreading areas are not applicable to the ABWR and are not in the plant layout because the majority of the signals will be multiplexed to the control room. Thus, it was implied that the 1 foot 3 foot separation guidelines allowed by Section 5.1.3 of IEEE Standard 384-1974 will not be applicable to ABWR nor will the guidelines of Position C12 of RG 1.75. The criteria for the separation and protection of cables approaching and/or exiting cabinets/panels was therefore not clearly addressed in the ABWR SSAR. To complete the staff's review in this area, additional information is required for the following items.

1. Routing criteria and protection to be provided electrical and/or optical cables used to carry multiplexed or other type of signals approaching and/or exiting cabinets/panels.

2. Criteria for routing of safety or non-safety power cables in any room with instrumentation and control cables.

3. The inconsistency between Item (5) of Section 8.3.1.4.2.2.2 and response to Question 435.30 of Amendment 10 to the SSAR was corrected in the draft information dated September 4, 1991. This item is considered resolved contingent on documentation of the draft information in the SSAR.

4. Cable separation in cable tunnels.

8.3.2.8 Compliance with IEEE Standard 384, "Criteria for Independence of Class 1E Equipment and Circuits"

In the draft submittal dated September 4, 1991, GE indicated that there will be no limitation on the use of IEEE 384-1981 in the ABWR design for separation. The NRC staff has not reviewed the changes between the 1974 and 1981 versions of this standard. Thus, to complete our review each difference between the 1974 and 1981 standards needs to be identified, justified, and approved for use on the ABWR.

Also, IEEE 384 is not the only standard in this classification for which an NRC evaluation (i.e., RG endorsement) has not been completed for the latest version of the standard. For any standard where the staff has not completed its evaluation of the version of the IEEE standard being referenced by GE in the SSAR, the differences will also have to be identified and justified by GE, and approved by the staff for use on the ABWR to ensure that positions taken are equally conservative as those included in standards currently approved by the staff.

In the draft submittal dated September 4, 1991, GE indicated that the ABWR design meets the requirements of SECY-89-013, "Design Requirements Related to the Evolutionary Advanced Light Water Reactors (ALWRs)," dated January 19, 1989. Specifically, this document requires that an analysis demonstrate that following loss of all equipment within a fire area, safe shutdown can be achieved. This requirement thus necessitated designing for the ABWR a 3-hour fire barrier around fire areas. Based on this stated requirement, it appears that the response to Question 435.23 (i.e., "the ABWR will provide separation by fire barriers sufficient to meet the requirements of letter SECY-89-013") means that two of the three divisions would be permitted to be located in the fire area provided safe shutdown can be achieved, not necessarily within accident analysis time restraints, with the remaining one of three divisions. To resolve this issue, the design basis for separation needs to be further clarified in the ABWR SSAR. This remains an open item pending adequate SSAR revision.

8.3.3 Protection

8.3.3.1 Electric Penetrations

Item 7 of Section 8.3.1.4.1.2 of Amendment 10 to the SSAR indicated that power circuits going through electric penetration assemblies are protected against over current by redundant interrupting devices. In addition, response to Question 435.31(b) of Amendment 10 to the SSAR indicated that it is an ABWR design requirement that redundant interrupting devices be provided for electrical circuits going through containment penetrations, if the maximum available fault current (including failure of upstream devices) is greater than the continuous current rating of the penetration. Based on the above design requirements, it appears that the proposed design will include redundant interrupting devices on all instrumentation and control circuits as well as power circuits that pass through containment. In addition, when calculating maximum available fault current at the penetration, current limiting devices will not be used in the calculation (i.e., worst case failure or shorting of the upstream or current limiting devices will be assumed as a given in the calculation). Based on the above interpretation, the staff concluded that the proposed design meets RG 1.63 (revision 3) and is acceptable contingent on resolution of the following items, and revision of the SSAR to reflect information included in the draft submittal dated September 4, 1991.

1. Clarification of interface requirements presented in Section 8.3.4.4 to clearly state the criteria or design requirements that must be demonstrated by (a) fault current clearing-time curves for protective devices, (b) thermal capability curves of the penetration, (c) location of protective devices, and (d) power supplies for protective devices.

2. Clarification of what is meant by current limiting devices versus protective devices.

8.3.3.2 Safety Bus Grounding

On every bus shown in Figures 8.3-1, 8.3-2, and 8.3-3 of Amendment 10 to the SSAR, there is one circuit shown connected to ground through a circuit breaker. The circuit breaker or bus grounding device is used to provide a safety ground on buses during maintenance operations. Interlocks for the bus grounding device include:

1. Under voltage relays must be actuated;

2. Related breakers must be in the disconnect position; and

3. Voltage for bus instrumentation available.

The staff feels that the proposed grounding device may be an important personnel protection enhancement for performing maintenance on safety buses and should be included in the design;

however, the staff is concerned that the above proposed interlocks may not be sufficient in and of themselves to prevent inadvertent closing of the device during non-maintenance operation. This item is considered resolved contingent on documentation of draft information provided on September 4, 1991, in the SSAR.

8.3.3.3 Qualification

Section 8.1.3.1.2.2 of Amendment 10 to the SSAR indicated, by reference to compliance with RG 1.32, that each type of Class 1E equipment will be qualified by analysis, successful use under similar conditions, or by actual test to demonstrate its ability to perform its function under normal and design basis events. Sections 8.3.1.2.4 and 8.3.3.1 of Amendment 10 to the SSAR included the following items in support of compliance with this RG 1.32 requirement.

1. Class 1E equipment essential to limiting the consequences of a LOCA are designed to operate in normal service and post accident environments;

2. Electric equipment is seismically qualified;

3. All Class 1E cables are moisture and radiation resistant and highly flame resistant;

4. Separate certification proof tests are performed to demonstrate 60 year life, radiation resistance, environmental capability, flame resistance, and gas evolution of cables;

5. Each power cable has a radiation resistant covering;

6. Conductors are specified to continue to operate at 100 percent relative humidity with a life expectancy of 60 years; and

7. Class 1E cables are designed to survive the LOCA ambient conditions at the end of a 60 year life span.

Each of the above items meets, in part, the guidelines of RG 1.32; however, based on the information presented, it was not clear that all cables, for example, are designed and qualified to survive the combined effects of temperature, humidity, radiation, etc. associated with a LOCA environment or other design basis event environments at the end of their qualified and/or design life.

Clarification of the design and qualification requirements for cables as well as other Class 1E equipment to survive normal and accident environments (including identification with justification of exceptions to the design and qualification requirements) should be provided in the SSAR.

In addition, Section 8.3.1.2.4 of Amendment 10 to the SSAR indicated that all Class 1E equipment which is essential to limiting the consequences of a LOCA is designed for operation in normal service environment and to operate in the post accident environment expected in the area in which it is located. Also, this section indicates that electric equipment is qualified to IEEE 344-1987, "Recommended Practices for Seismic Qualifications of Class 1E Equipment for Nuclear Power Generating Stations," i.e., electric equipment will be demonstrated to meet its performance requirements during and following the design basis seismic event by test and/or analysis).

Based on information presented, the design and qualification commitment for electric equipment in the proposed ABWR design was not clear with respect to the capability of equipment to survive the combined effects of a LOCA environment.

GE indicated in draft information provided on September 4, 1991, and in discussions with the staff, that all Class 1E equipment essential for LOCA or for any design basis event will be designed and qualified without exception to operate in a normal, accident, and post accident environments for any design basis event. In addition, it was indicated that SSAR would be revised to reflect the full compliance of the ABWR design to 308-1980 guidelines as it relates to design and qualification of equipment (i.e., Sections 5.3, 5.4, and 5.7 of 308-1980).

In regard to the design and qualification of equipment to operate within allowable design basis limits such as for 5 minutes when subject to voltage below 90 percent and to operate for a predetermined time when voltage is below 70 percent, GE indicated in discussions with the staff that the qualification of equipment would be included in the equipment specification and that the SSAR would be revised to indicate this qualification.

This item is considered resolved contingent on documentation of the above information in the SSAR.

8.3.3.4 Submergence

Item (6) of Section 8.3.1.4.2.3.2 of Amendment 10 to the SSAR stated that any electrical equipment and/or raceway for RPS or ESF located in the suppression pool level swell zone will be designed to satisfactorily complete their function before being rendered inoperable due to exposure to the environment created by the level phenomena. In response to staff Question 435.36 of Amendment 10 to the SSAR, the licensee identified electrical equipment that may be submerged as a result of suppression pool level swell phenomena or as a result of a LOCA. The licensee further indicated that the design specifications associated with this electric equipment would require that terminations be sealed such that equipment operation would not be impaired by submersion. The qualification of this equipment in accordance with the guidelines of Section 4.7 of IEEE Standard 308-1974, "Criteria for Class 1E Power Systems for Nuclear Power Generating Stations," was, however, not specifically addressed.

Based on information presented, it appeared that electrical equipment subject to submergence was not qualified and only partially designed for submergence. This conclusion contradicted Section 8.3.1.2.1 of Amendment 10 to the SSAR which stated that all Class 1E equipment is qualified.

It was the staff concern that equipment failure due to submergence may adversely affect the safe operation of the plant and may adversely affect Class 1E power sources serving this equipment.

The draft information provided by GE on September 4, 1991, indicated that all equipment is designed for the submergence environment. In addition, it was indicated in discussions that the SSAR would be revised to provide an explicit commitment to qualification for submergence in conformance to Section 4.7 of IEEE Standard 308-1974. This item is considered resolved contingent on documentation in the SSAR of the draft information and the commitment noted above.

8.3.3.5 Impingement of Fire Suppressant

Section 8.3.3.1 of Amendment 10 of the SSAR stated that the cable installation is such that direct impingement of fire suppressant will not prevent safe reactor shutdown. Based on this statement it was not clear whether impingement of fire suppressant will or will not cause failure of cable systems. The staff was concerned that cables and other electric equipment may not be designed and qualified to perform their safety function while being subjected to the direct impingement of fire suppressant.

The draft information provided on September 4, 1991, by GE indicates that cables and other electric equipment are not designed and qualified to perform their safety function while being subjected to the direct impingement of fire suppressant; however, the justification for this lack of design and qualification indicated that redundant divisions are provided such that with failure of the cable system or equipment in one division due to fire suppressant impingement and single failure of a second division, safe shutdown of the plant can be achieved with the third division. This item is considered resolved contingent on confirmation that fire suppressant can only cause impingement on one division and that, the plant can be safely shutdown with the one division.

8.3.3.6 Isolation Between Safety Buses and Non-Safety Loads

Section 8.3.1.1.2.1 of Amendment 10 to the SSAR indicated that isolation breakers are provided between the Class 1E and non-Class 1E buses. In addition to normal over current tripping of the isolation breaker, zone selective interlocking is provided between each isolation breaker and its upstream Class 1E bus feeder breaker. Section 8.3.1.2.1 of Amendment 10 to the SSAR indicated that even though the isolation breaker is fault-current actuated in non-compliance with the guidelines of Position 1 of RG 1.75, the intent of this Guide is met through the zone selective interlocking technique; thus, the design meets the recommendations of this and other guides.

With respect to protecting Class 1E systems from failure of non-Class 1E systems and components, the staff agrees with GE that coordinated breakers with zone selective interlocking meets the intent of Position 1 of RG 1.75, and meets the protection requirements of Criteria 2 and 4; however, with respect to meeting the sufficient independence requirement of Criterion 17, the staff disagrees with the licensees assessment. Non safety computers and transient recorder loads shown on Figure 8.3-5 of Amendment 10 to the SSAR have provisions included in their power supply design for automatically transferring these loads from Class 1E division 1 to 3 and from Class 1E division 2 to 3. In addition, it appears that the power supply may also include provision for automatic transfer of these loads between Divisions 1 and 2. The design does not meet the guidelines of RG 1.6 nor the intent of Position 1 of RG 1.75. The proposed design thus may not meet the independence requirements of Criterion 17 of Appendix A to 10 CFR Part 50. To resolve this and other issues, additional information is required for the following items and/or staff positions.

1. Reliability, testability, test frequency, functional test, and calibration of the isolation breaker coordination and zone selective interlocking.

2. All non safety circuits connected to the Class 1E system through the isolation breaker with zone selective interlocking shall be treated as associated circuits.

3. Interconnection between redundant divisions (through safety or non safety buses) shall be maintained with two normally open and interlocked devices that are separate and independent such that single failure or single operator action cannot cause the interconnection of or challenge to redundant divisions.

Administrative interface criteria and/or alarms for maintaining and assuring interconnections open.

5. Identification of all safety and non safety loads that can be powered from more than one Class 1E division power supply. Appendix 20B should be modified to clearly indicate loads powered from more than one safety division.

6. A description and analysis of the use of fault actuated isolation devices in the Class 1E constant voltage constant frequency power system.

7. The use of uninterruptable power supplies as isolation devices (Reference response to Question 435.34c).

8. The contradiction between Figure 8.3-3 and response to Question 435.49c. Response to Question 435.49c states that T/B motor control center (MCC) is non-Class 1E and is powered from non-Class 1E power centers. Figure 8.3-3 in contradiction shows T/B MCC to be powered from Class 1E power sources. The draft information provided by GE on

September 4, 1991, indicated that the subject MCC has been moved from the safety to the non-safety bus. This item is considered resolved contingent upon revision of Figure 8.3-3 and documentation of the draft information in the SSAR.

9. The contradiction between response to Question 435.18e and other Amendment 10 SSAR sections as to tripping of non-safety loads on a LOCA signal. The draft information provided on September 4, 1991, revised the response to Question 435.18e to indicate that loads are only tripped on a loss of offsite power (LOOP) signal. The revision resolves this item contingent upon documentation of the draft information in the SSAR.

10. Identification of all non-safety loads and their KW ratings that can be powered from safety related diesel generators and identification of the extra KW capacity available to supply non-safety loads during the various modes of plant operation.

11. The capacity, margin, and other provisions that will be included in the sizing criteria for electric systems and components (i.e., diesel generators, batteries, distribution systems, etc.) which will allow them to perform their safety function reliably while supplying non-safety loads.

12. Inconsistency between Responses c and d to Question 435.18 as to loads that are disconnected for a LOCA occurring after loads have been sequenced following a loss of offsite power. Response c indicates loads not required for LOCA are tripped while Response d implies that LOOP loads remain connected.

8.3.3.7 Diesel Generator Protective Relaying

Section 8.3.1.1.6.4, Protection Requirements, of Amendment 10 to the SSAR indicated that the following protective relaying will trip the diesel generator and will be retained under accident conditions: Generator differential, bus differential, engine over speed, low diesel cooling water pressure (two out of two sensors), and low differential pressure of secondary cooling water (two out of two sensors). Other protective trips will be bypassed during LOCA conditions. This protective relaying (except for bus differential) meets Position 7 of RG 1.9 (revision 2) and is acceptable.

In regard to bus differential; GE indicated in discussions with the staff that the bus differential would be removed from the list and that the SSAR would be revised to indicate the removal. This item is considered resolved contingent on revision of the SSAR and clarification of (1) diesel generator trip alarms, (2) extent of compliance of trip bypass circuitry with IEEE 279 requirements, and (3) independence of trip sensors.

8.3.3.8 Thermal Overloads

Response to Question 435.60 of Amendment 10 to the SSAR indicated that thermal overload protection for Class 1E motor operated valves (MOV's) is in effect only when the MOV's are in test and are bypassed at all other times by means of closed contacts in parallel with the thermal overload contacts. A visual indication is provided in the main control room when the MOV is in the test mode. The proposed design for bypass can assure that the thermal overload protection will not be in effect during accident conditions to prevent operation of valves. The design thus meets the intent of RG 1.106 and is acceptable with the possible exception of testability.

Sufficient information relating to the capability for periodically testing the contacts that are in parallel with the thermal overload contacts to assure they are closed during normal operation has not been presented. To resolve this concern, additional information is required concerning testing of the thermal overload bypass device.

8.3.3.9 Breaker Coordination

Section 8.3.1.1.2.1 of Amendment 10 of the SSAR stated that tripping of the Class 1E bus feeder breaker is normal for faults which occur on its Class 1E loads. The staff disagrees with this statement. Class 1E load breakers should be coordinated with the Class 1E bus feeder breaker so that faults which occur on its Class 1E loads will, to the extent possible, not cause trip of the bus feeder breaker. This is to minimize the potential for loss of safety equipment.

The draft information provided on September 4, 1991, revised the SSAR to remove the statement that trip of the bus supply breaker is normal for faults that occur on its Class 1E loads. Also in discussions with the staff, GE indicated that the SSAR would be clarified to state that the Class 1E load and bus supply breakers are coordinated. This item is considered resolved contingent on documentation of the above information in the SSAR.

8.3.3.10 Protective Relaying

Experience with protective relay applications has established that relay trip set point will drift with conventional types of relays. Set point drift at nuclear power plants has resulted in premature trip of redundant safety related pump motors when they were required to be operative. While the basic need for proper fault protection for feeders/equipment is recognized (and may be a requirement for the design basis event fire), the total non-availability of redundant safety systems due to spurious trips of protective relays is not acceptable. The primary safety function of the electrical distribution system is to reliably provide power to safety related equipment. GE in response to this concern (Question 435.58 of Amendment 10 to the SSAR) indicated that loads, such as motors, will be designed with sufficient current carrying capability or overload margins so that set points of protective devices can be set sufficiently above the operating current point of loads to allow for set point drift. The use of loads, such as motors, with sufficient overload margins resolves the above concern if one assumes the following:

1. Specific design parameters and/or interface requirements clearly define (in the ABWR SSAR) the overload margin requirements with respect to protective device trip set point, the margin between the trip set point and operating current point of loads, set point drift, and the margin between the trip set point and overload rating of loads.

2. The protective device trip set point is periodically verified and calibrated.

3. The protective device is subjected periodically to a functional test to demonstrate (a) its capability to not trip at its design rating i.e., the normal operating current of load plus margin and (b) its capability to trip when subjected to a fault current.

The staff is concerned that the ABWR design may not meet the above assumptions. In order to resolve this concern regarding the acceptable testing of systems, additional information is required describing how the ABWR design will meet each of these assumptions.

8.3.3.11 Fault Interrupting Capacity

Design criterion (4) in Section 8.3.1.1.5.2 of Amendment 10 to the SSAR stated that the interrupting capacity of switchgear, load centers, motor control centers, and distribution panels is compatible with the short circuit current available at the Class 1E buses. Based on this statement, it is not clear that the interrupting capacity of this equipment will be equal to or greater than the maximum available fault current to which it would be exposed. To clarify the criteria for the interrupting capacity of equipment and to resolve other related concerns, additional information is required for the following items.

1. Clarification of the criteria for interrupting capacity, and

2. Compliance of both Class 1E and non-Class 1E switchgear, load center, motor control centers, and distribution panels to applicable industry standards.

8.3.3.12 Control of Design Parameters

Valve problems such as excess friction, packing too tight, etc. can result in an operational condition where the current drawn will exceed the design rating or capability of the insulation system used in the valve motor winding. Operating experience has shown that excessive current, if undetected during operation, can cause premature or unexpected failure when the valve is next operated. Methods, design provisions, alarms, or procedures for assuring the valve motor will not be operated with excessive currents without operator knowledge (or will always be operated within their design limits) was not presented in the ABWR SSAR.

The draft information provided on September 4, 1991, indicated that thermal overloads will provide protection at all times for non-Class 1E valves and will provide protection during manual testing or maintenance for Class 1E valves. At all other times, the Class 1E valve will not be protected. Protection to assure that the valve motor windings will not be overloaded and damaged during operation of the valve for normal plant operation remains a concern. In order to resolve this issue, additional information is required as to the protection and/or monitoring that will be implemented to assure that the valve motor windings will not be overloaded and damaged during operation of the valve for normal plant operation.

8.3.3.13 Fire Protection of Cable Systems

Section 8.3.3.2 of Amendment 10 of the SSAR indicated that spatial separation is used as a method of preventing the spread of fire between adjacent cable trays of different divisions (e.g., inside primary containment). The objective is always to separate cable trays of different divisions with structural fire barriers such as floors, ceilings, and walls. Where a floor, ceiling, or wall is not possible, divisional trays are separated spatially by 3 ft. horizontally and 5 ft. vertically. Where this 3 ft.-5 ft. spatial separation is not possible, fire rated barriers are used to separate divisional cable trays. For a fire initiated by a cable fault within one division, the above defined separation meets the guidelines of RG 1.75, will provide reasonable assurance that a fire in one division will not propagate to a redundant division, and is acceptable.

For the design basis event fire, the adequacy of spatial separation and fire rated barriers to prevent the spread of fire between adjacent cable trays will be contained in the Section 9.5.1 of this report which addresses fire protection.

8.3.3.14 Electrical Protection Assemblies (EPAs)

Two independent electrical protection assemblies (EPAs) were required (by a September 24, 1980, generic letter to all operating BWRs) on the output of RPS power supplies in order to satisfy the single failure criterion for non-fail-safe type failures which may be caused by under voltage, over voltage, and under frequency conditions.

Response to Question 435.7 of Amendment 10 to the SSAR indicates that EPAs will not be used in the ABWR design because of special design features. These special features include monitoring of voltage and frequency, automatic transfer of power supply input sources when voltage and/or frequency exceed preestablished limits, control room alarm for abnormal conditions, operator action in response to alarm of abnormality, and design and qualification of equipment to not fail after operation for a period of time under the extremes of voltage and frequency.

Based on a review of these special features, it appears that they may provide reasonable assurance that any abnormality in voltage and frequency (which can cause failure of fail-safe-type equipment) will be promptly disconnected by alarms and operator action. The special features, however do not meet the single failure criterion. Failure of the special features to alarm or failure of the operator to take prompt appropriate action are single failures which may cause a non-fail-safe type failure. The capability to scram the reactor may thus be compromised.

An explicit statement of compliance with the staff position that two EPAs will be provided on the output of the RPS power supplies with justification for areas of non compliance should be included in the ABWR SSAR.

8.3.4 Electrical Independence

8.3.4.1 Interconnections

Figure 8.3-8 of Amendment 10 to the SSAR shows two interconnections between redundant divisions:

1. Division III 480 volt bus 6E-1 is connected to Division I 480 volt bus 6C-1 through circuit breakers and a mechanical interlock. Section 8.3.2.1 of Amendment 10 to the SSAR indicated that this interconnection is used to transfer the 250 V DC normal battery charger between Divisions I and III load centers.

2. Division III 480 V MCC is connected to Division I 480 V 6C-1 through battery chargers, breakers, and key interlocked breakers. Section 8.3.2.1 of Amendment 10 to the SSAR indicated that this interconnection is used for selection of the normal or the standby battery charger.

Criterion 17 of Appendix A to 10 CFR Part 50 requires independence between redundant divisions such that failure of one will not challenge or cause failure of the remaining redundant divisions. Sufficient information describing these and other interconnections as to their compliance with the independence requirement of GDC 17 was not provided in the ABWR SSAR. It is the staff position that two independent open disconnect links, locked open breakers, or other equivalent open devices be maintained between redundant divisions if redundant divisions are to be electrically interconnected. Additional information as to the extent of compliance with the above staff position with justification of areas of non compliance is required in the ABWR SSAR for resolution of this issue.

8.3.4.2 Constant Voltage Constant Frequency Power Supplies

Section 8.3.1.1.4.2 of Amendment 10 to the SSAR indicated that each of the four independent trip systems of the reactor protection logic and control system are powered by four constant voltage constant frequency control power buses (Divisions I, II, III, and IV). This section also states that each of these buses is supplied independently from an inverter which, in turn is supplied from one of four independent and redundant AC and DC power supplies. Subsequent sections and Figure 8.3-6 of Amendment 10 to the SSAR, however, indicated that the AC supply for Divisions I and IV originates from a single 480 volt motor control center (C14). A single 480 volt motor control center is not independent and redundant as stated in Section 8.3.1.1.4.2. The draft information provided on September 4, 1991, revised the SSAR to indicate that there are four independent and redundant DC systems and three (versus four) independent and redundant AC systems. This item is considered resolved contingent on documentation of the draft information in the SSAR.

8.3.4.3 Power Supply Circuits for Safety/Relief Valves (SRVs)

Section 19E.2.1.2.2.2 of Amendment 10 to the SSAR indicates that portions of each safety relief valve (SRV) control circuit utilize non-safety grade power and that this non-safety grade power is taken from the Class 1E DC system through DC/DC converters or isolation devices connected to each of the four redundant and independent Class 1E DC system buses. Section 19E.2.1.2.2.2 implies that control power for each SRV comes from a minimum of two different Class 1E power source divisions. One source directly from the Class 1E DC bus with the other from a different Class 1E DC bus through the DC/DC converter. The staff was concerned that the proposed design for powering the SRV's may not provide sufficient independence between the redundant DC power sources in accordance with the requirements of GDC 17.

Draft information provided by GE on September 4, 1991, modified Section 19E.2.1.2.2.2 of the SSAR to delete reference to the use of non-safety grade power taken from safety grade batteries for a portion of each SRV control circuit. In addition, the information indicated that non-divisional power is not utilized in either the SRV or automatic depressurization system (ADS) functions. This item is considered resolved contingent on documentation of the draft information in the SSAR.

8.3.5 Lighting Systems

Section 9.5.3 .2 f Amendment 10 to the SSAR indicates that adequate li... for any safety related area, such as areas used during emergencies or shutdowns, including those along the appropriate access or exit routes, are provided from 3 different lighting circuits: (a) Normal, (b) Standby, and (c) Emergency DC and/or self-contained battery fixtures.

In order to complete our review of lighting systems in accordance with SRP Section 9.5.3, additional information is required for the following items.

1. Criteria for what constitutes an adequate level of lighting for various areas of the plant and for the various modes of plant operation.

2. Identification with justification for specific plant areas and modes of plant operation that do not meet criteria for what constitutes adequate lighting.

3. Source of power for normal lighting.

4. Frequency of inspection for normal lighting.

5. Plant areas where 50 percent lighting shall be secured with one standby lighting power supply.

6. Method of distinguishing between norm..., standby, and emergency DC circuits to assure that they will be routed separately.

7. Source of power for standby lighting.

8. Separation between the two standby power source circuits.

9. Seismic design of standby lighting.

10. Compliance of standby lighting with Class 1E circuit requirements.

11. The redundancy of the emergency DC lighting circuits.

12. The level of illumination of emergency lighting.

13. Periodic inspection and testing of lighting.

14. Justification for not having self contained battery fixtures seismically qualified.

15. The illumination levels with justification of the self contained battery fixtures.

16. Justification for having self contained battery fixture lighting turn off with restoration of power versus restoration of adequate light.

17. Justification for not having any seismically qualified lighting.

8.3.6 Design Control

8.3.6.1 Control of the Design Process

Recently, there have been a number of problems identified with the electrical system design at nuclear power plants. Although the majority of these problems arose as a result of modifications performed after plant licensing some were (and all could have been) the result of poor original design. Generic letter 88-15 addresses a number of these problems that have occurred primarily as a result of inadequate control over the design process. These problems have occurred in areas of electrical system design which have historically well established and comprehensive design criteria and guidelines available for the design engineer such as circuit breaker coordination and fault current interruption capability. The staff does not normally undertake a detailed review of these areas. The staff instead relies on the designers proper exercise of the well established design criteria and guidelines. To ensure that the criteria and guidelines are followed, design control is required.

Draft information provided on September 4, 1991, by GE indicated that the required design control being implemented for the ABWR electrical systems and for any subsequent modifications thereto is based on the document, "Nuclear Energy Business Operations Quality Assurance Program Description," (May 1987), NEDO-11209-04a, the "Green Book," Revision 7 which is referenced in SSAR Section 17.1.3. The acceptability of the GE design control process has been included as part of the staff's review of Chapter 17 of the ABWR SSAR.

C.3.6.2 Control of the Design Bases

The bases for the design described and presented in the ABWR SSAR should be, for the most part, used as the basis by which the NRC could issue a plant operating license. Based on a review of the bases presented in Chapter 8 and other related chapters, numerous inconsistencies have been identified. These inconsistencies are identified in other sections of this report. Given these inconsistencies, it appears that the process for controlling the design bases being presented in the ABWR SSAR may be deficient relative to the requirements of Appendix B of 10 CFR Part 50.

Draft information provided by GE on September 4, 1991, indicated that the formal engineering review and update of the SSAR will not occur until after completion of the NRC's SER for Chapter 8. The formal engineering review was not completed by GE so that resolutions to SER issues could be incorporated into the ABWR SSAR along with the general update of Chapter 8.0.

This item will remain open until receipt of the general update of Chapter 8.0 with the resolution to SER issues and completion of the staff review of the updated Chapter 8.0 for clarity and consistency of the ABWR design bases for the plant systems and their electrical support systems.

8.3.7 Testing

Section 8.3.1.1.5.3, Testing, of Amendment 10 of the SSAR indicates that the design of Class 1E equipment provides for periodically testing the chain of system elements from sensing devices through driven equipment to assure that Class 1E equipment is functioning in accordance with design requirements. This section also implies that the requirements of the single failure criterion described in IEEE Standard 379-1977, "Standard Application of the Single Failure Criterion to Nuclear Power Generating Station Safety Systems," are met with respect to testing of Class 1E equipment.

The staff interprets this section to mean that one complete electrical system division may be de-energized and taken out of service for maintenance and/or repair during any mode of plant operation and still have the remaining electrical systems in compliance with the single failure criterion. The staff concluded that this design provision for testability of electrical systems as interpreted meets the sufficient testability requirement of Criterion 17 and is acceptable.

In order to confirm and clarify this interpretation in the ABWR SSAR and address other related issues, additional information is required for the following items.

1. Explicit statement for testability during normal plant operation while meeting single failure requirements with remaining systems for any design basis event.

2. Proposed allowed outage times for one division to be out of service to perform preplanned and unplanned maintenance.

3. Frequency for periodically testing each system element to assure its availability to mitigate design basis events.

4. Basis for establishment of test frequency for each system element.

5. Identification (with justification for their use) of any divisional cross connection which must be used to meet the above design provision for testability.

6. Clarification of the version of IEEE Standard 379 being referenced in Section 8.3.1.1.5.3.

7. Identification with justification for any areas of non-compliance with the above design provision and GDC 18 requirement for testability.

8. Inconsistency between Sections 8.3.1.1.5.3 and 8.3.1.2.2 with respect to meeting the single failure criteria while testing one division of the CVCF power supply system.

9. Periodic testing provisions to assure the capability of the diesel generator to accept loads in any loading order (Reference: 435.18).

10. Periodic testing to demonstrate the diesel generator's capability of being started in 20 seconds and fully loaded within 30 seconds.

11. Testing and calibration of the diesel generator over current relay.

12. Testing and/or analysis to be performed periodically to demonstrate the capability of the diesel generator to supply the actual full design basis load current for each sequenced load step.

13. Interface requirements for compliance with RG 1.47, Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems and Power Systems Branch Technical Position 2, BTP PSB-2, Criteria for Alarms and Indications Associated with Diesel-Generator Unit Bypassed and Inoperable Status.

14. Confirmation that the testability inherent in the design of protection systems is not so burdensome operationally that required testing at intervals of 1, 2, or 3 months cannot be included in the Technical Specifications if deemed necessary. The systems addressed should include but not be limited to the reactor protection system and the engineered safety features actuation system. Identify exceptions.

15. Testing to demonstrate the capability of the diesel generator to automatically revert to the emergency response mode while in the test mode if a design basis accident or loss of offsite power event were to occur.

8.3.8 Capacity and Capability

8.3.8.1 Shutdown Capability of Each Load Group

Section 8.3.1.2.1 of Amendment 10 to the SSAR stated that the standby power system redundancy is based on the capability of any one of the four divisions (one of three load groups) to provide the minimum safety functions necessary to shut down the unit from the control room in case of an accident and maintain it in the safe shutdown condition. However, in apparent contradiction, Section 1.2.1.2.5.2 stated that the Class 1E power systems are designed with three (3) divisions with any two divisions being adequate to safely place the unit in the hot shut down condition.

In discussions with the staff GE indicated that they would modify Section 1.2.1.2.5.2 of the SSAR to provide correction and clarification that one of three load groups and one of four divisions is required to maintain the plant in hot shutdown and to bring the plant to safe cold shutdown. This item is considered resolved contingent on revision of the SSAR.

8.3.8.2 Non-Safety DC Power Systems

Section 1.2.2.5.1.6 of Amendment 10 to the SSAR indicated that the ABWR design includes a unit auxiliary DC power system that supplies power to DC loads that are non-safety related. However, Section 8.3.2, which is suppose to address DC power systems included in the ABWR, omitted description and analysis of the unit auxiliary DC power system. Description and analysis of this system and the extent it will be used to supply DC control power to systems that are important to safety (such as offsite power circuits) should be defined in the ABWR SSAR.

8.3.8.3 Class 1E 125 volt DC Battery Capacity

Section 8.3.2.1.3.2 of Amendment 10 to the SSAR indicated that each of the four Class 1E 125 volt batteries have sufficient stored energy to operate connected essential loads continuously for at least two hours without recharging. During loss of AC power, Section 5.4.6.1 of Amendment 10 to the SSAR indicated that the battery capacity should allow over four hours of operation of the RCIC system. Item 3 of Section 19E.2.1.2.2.2 of Amendment 10 to the SSAR indicated that the DC batteries will be sized to be capable of operating the RCIC system for a minimum of 8 hours assuming load shedding and use of all four Class 1E batteries.

Item 2a of Section 19E.2.1.2.2.2 of Amendment 10 of the SSAR indicated that Division 1 battery by itself has sufficient capacity to operate the RCIC system for 8 hours. These inconsistencies should be clarified and the design basis load profile for each battery should be explicitly stated in the ABWR SSAR.

8.3.8.4 Use of Silicone Diode in the DC System

Figure 8.3-7 and response to Question 435.51 of Amendment 10 to the SSAR indicated that a silicone diode (SID) which has a voltage drop of 10 volts has been installed in series with the output of the battery and battery charger. During normal operation (i.e., battery charger output voltage is set at 140 volts for equalize charge) the switch in parallel with the silicon diode will be open so that the voltage from the battery charger to the DC bus will remain at 130 volts (140 volts minus the 10 volt drop across the silicone diode) while 140 volts is supplied to the battery for equalize charge. The staff feels that the proposed design has merit; however, sufficient descriptive information and analysis to reach a conclusion on acceptability for all modes of plant operation has not been presented in the ABWR SSAR. To resolve staff concerns, additional information is required for the following items.

1. Reliability of the proposed DC system. The addition of the silicon diode in the DC system circuit adds an additional level of unreliability to the system while at the same time may improve overall DC system reliability.

2. Capacity and capability of the DC system to supply design basis loads during loss of offsite power events.

3. Design provisions to assure the battery will never have to supply its design basis loads with the silicon diode connected in series with the battery and DC bus.

4. Monitoring for the switch installed in parallel with the diode.

8.3.8.5 Class 1E AC Standby Power System

As a result of our review of the standby power system proposed in the ABWR SSAR, the following areas of concern have been identified.

1. Inconsistency between Sections 8.3.1.1 8.2 and 8.3.1.1 8.3 of Amendment 10 to the SSAR as to the design capability of the diesel generator to start and attain rated voltage and frequency. Section 8.3.1.1.8.2 of Amendment 10 to the SSAR indicated a 13 second design capability while Section 8.3.1.1.8.3 indicated 20 second capability.

   Amendment 17 to the SSAR corrected the inconsistency by changing the 13 seconds start time for the diesel generator to 20 seconds with the sequence start times for loads changing accordingly. In discussions with the staff, GE indicated that delayed start times for the various safety loads are within the required accident analysis times with at least a 19 second margin. This item is considered resolved contingent on documentation of this margin.

2. Clarification of the diesel generator design details which are to be supplied by others (Reference: Question 435.21(b) of Amendment 10 to the SSAR) and the criteria the design must meet (i.e., interface requirements).

3. Clarification of the continuous and overload ratings of the diesel generator defined in Section 8.3.1.1.8.2.

8.3.9 Station Blackout

The ABWR coping analysis for Station Blackout is presented in Section 19E.2.1.2.2 of Amendment 10 to the SSAR. Also, Table 19E.2-2 of Amendment 10 to the SSAR presents design basis values for various plant parameters that will not be exceeded at the end of the 8 hour coping duration for a Station Blackout event. Based on a review of this coping analysis and design information presented in other sections of the ABWR SSAR, the staff has identified the following areas of concern.

1. Analysis results demonstrating safe plant shutdown can be accomplished starting with reestablishment of AC power to any one of the three AC divisions from either offsite, diesel generators, or combustion turbine generator at the end of the 8 hours of coping.

2. Design and qualification of equipment for the environments expected during and following the 8 hour coping time analyzed for station blackout events.

3. Extent to which the combustion turbine generator complies with Position 3.3.5 of RG 1.155, Station Blackout.

4. The inconsistency between response to Question 435.2 and the September 4, 1991, draft Section of 19E.2.1.2.2 of the SSAR with respect to the number of SRVs powered from Division 1.

9 AUXILIARY SYSTEMS

9.3 Process Auxiliaries

9.3.1 Compressed Air Systems

The designs of the compressed air (CA system) system were reviewed in accordance with SRP Section 9.3.1, "Compressed Air System." The review of the ABWR's CA system system involved a review of information in the SSAR and GE's responses to RAIs. The acceptance criteria for the safety related portions of the CA systems includes compliance with GDC 1 which is addressed in Section 3.2.

The ABWR CA system is comprised of four subsystems: 1. the atmospheric control system (ACS); 2. the high pressure nitrogen gas supply system (HPIN); 3. the Instrument air (IA) system; and, 4. the service air (SA) system. These systems provide compressed gas (either air or nitrogen) to operate safety-related equipment relied upon to mitigate the consequences of design basis events and plant equipment used for normal facility operation. Since the non-safety related portions of the system are interconnected with the safety-related portions of the system, the designs of the four subsystems are summarized below.

The AC system establishes and maintains an inert atmosphere within the primary containment during all plant operating modes except: 1) during shutdown for refueling or equipment maintenance; and, 2) during limited periods of time to permit access for inspection at low reactor power. The AC system is non-safety class except as necessary to assure primary containment integrity (e.g., penetrations and isolation valves). The AC system includes nitrogen storage tanks, vaporizers, valves and piping carrying nitrogen to the containment, valves and piping from the containment to the SGTS and HVAC exhaust line, non-safety oxygen monitoring, and all related instruments and controls. The AC system provides nitrogen from the nitrogen evaporator to the High Pressure Nitrogen Gas Supply System during normal operation.

The HPIN is comprised of both non-essential (i.e., non-safety related) and essential systems. A single non-essential system provides a continuous nitrogen supply to all pneumatically-operated components in the primary containment during normal operation. As noted above, during normal operation, the High Pressure Nitrogen Gas Supply System is supplied from the nitrogen gas evaporator/storage tank via the makeup line to the AC system.

The essential system is comprised of two independent divisions, with each division containing a safety-related emergency stored nitrogen supply capable of supplying 100 percent of the requirements of the division being serviced. Nitrogen gas for the essential system is supplied from high pressure nitrogen gas storage bottles. There are tielines between the non-essential and each division of the essential system. Each tieline has a motor operated shut-off valve.

The IA system provides dry, oil-free, compressed air for valve actuators and for non-safety related instrument control functions and for general instrumentation and valve services outside the containment. (All instrumentation and control systems located inside the containment are supplied with nitrogen gas during normal plant operation.) The primary containment penetrations of the IA system are of seismic Category I design and are equipped with sufficient isolation valves to satisfy single failure criteria. (In GE's response to Question 430.215, the staff noted that the reference to "...containment penetrations and drywell penetrations of the instrument air system..." in SSAR Section 9.3.6.1.1 should be revised to reference primary containment penetrations only. The staff considers the correction of the SSAR text to be a confirmatory issue).

The SA system is designed to provide compressed air of suitable quality for non-safety related functions. The SA system provides compressed air for services requiring air of lower quality than that provided by the IA system. The containment penetrations and drywell penetrations of the SA system are of seismic Category I design and are equipped with sufficient isolation valves to satisfy single failure criteria. Since the SA system does not directly interface with the HPIN system and does not provide any safety related function, this system will not be addressed further in this section.

As noted above, only the HPIN system provides compressed gas to safety related components. However, both the Atmospheric Control System and the Instrument Air System directly interface with the HPIN system and could affect the reliability of safety related components relied upon to mitigate the consequences of design basis events. Therefore, in assessing the adequacy of the CA systems, in accordance with acceptance criteria in Section 9.3.1 of the SRP, these three systems were addressed.

The SSAR was reviewed to identify the safety related portions of the CA system. The text and figures in SSAR Section 6.7 do not clearly identify which portions of the HPIN system are safety related; however, the response to an RAI and valve and instrument numbers reflected on a draft revision to SSAR Figure 6.7-4 provide information which identify which portions of the system are safety-related. These portions include: nitrogen storage bottles and their headers up to and including F002A through F002D; piping and valves from F002 (A through D) to the accumulators for the ADS valves; and piping from the crosstie valves F012A & B to the piping identified previously. Additional safety related piping in the IA and AC systems include piping and valves from F200 to F209, inclusive. (The staff noted an inconsistency in the designation of the inboard isolation valve, in that, it is referred to as F209 in all drawings and as F208 in the response to RAIs. In Table 9.3.1-1 of the SER, additional numbering inconsistencies, including valve numbers and valve operator type, are identified. Resolution of these inconsistencies is an open item. Also, the information related to the safety classification of components has not been fully incorporated into the SSAR text and figures. While the information provided in response to requests for information is acceptable, this information should be fully incorporated into the SSAR. This is an open item.

Contrary to the guidelines of the SRP, the SSAR does not indicate the failure mode for the valves in the HPIN system. Except as noted below and assuming that the MOVs fail "as-is," the system configuration is acceptable. Information related to the failure mode of components should be incorporated into the SSAR text and drawings. Identification of the failure modes of th valves in the CA systems, i.e., confirmation that they do fail 'as-is,' is an open item.

Valves AO F018A & B in SSAR Figure 6.7-1 are indicated as "NC. FO." (i.e., normally closed and fail open). This is inconsistent with the text of Section 19E.2.1.2.2.2(2)(b) which indicates that upon a loss of power, the operator will have to manually open these valves. This is also inconsistent with a design which should protect the storage bottles from inadvertent depressurization due to a postulated line break. Confirmation that these valves do not fail in the open position remains an open issue.

The vessels, piping and fittings of the safety related portions of the HPIN system (except penetrations) are designed to seismic Category I, ASME Code III, Class 3, Quality Group C and Quality Assurance B requirements. In addition, the SSAR indicates that the crosstie valves (i.e., F012A & B) which connect the safety related portions of the HPIN system to the non-safety related portions are safety related (and by implication are seismically qualified). While two isolation valves are not provided on the non-essential/essential interface within the HPIN system, the system is deemed adequate in accordance with information provided in the response to Question 430.211. One isolation valve is provided between each division of safety related HPIN system and the non-essential portion of the system. Additionally, check valves are provided to prevent backflow of nitrogen from the accumulators through any possible break in the non-essential portion of the system. The inboard isolation valves are check valves and each safety related accumulator is downstream of a separate check valve. The accumulators are sized to perform their function several times without requiring recharging. The combination of isolation valves, check valves, and the sizing of the accumulators ensures that the system would fulfill its function in the event of a rupture in the non-essential portion of the piping.

The piping and valves for the containment and drywell penetrations for the HPIN and IA systems are designed to seismic Category I, ASME Code III, Class 2, Quality Group B and Quality Assurance B requirements. The isolation provisions for the AC system primary containment penetrations include two isolation valves that are both located outside the primary containment, not strictly in conformance with GDC 56. However, the penetrations do not extend inside the containment and (as described in the response to Question 430.209) an inboard isolation valve would not be practical. An inboard isolation valve for the AC system would be exposed to a more severe environment and would not be easily accessible for inspection, surveillance, and maintenance. A similar design was approved in the staff review of the GESSAR II BWR/6 SER. Therefore, it has been determined that the design is acceptable.

The accumulator design is considered adequate to provide the necessary time for operator action to manually actuate any valve(s) required to replenish the accumulators. Therefore, the CA system complies with Positions C.1 and C.2 of RG 1.29.

The safety related portions of the HPIN system are located within the reactor building. The reactor building is designed to withstand and protect equipment from tornadoes, externally generated missiles, floods and other natural phenomena. In addition, it is indicated that the safety related portions of the HPIN system will retain their function during LOCA and seismic events in which non-safety related parts may be damaged. Section 6.7.3 of the SSAR further indicates that the pipe routing of Divisions 1 and 2 of nitrogen gas is kept separated by enough space so that a strike by a single high energy whipping pipe, the jet force from a single broken pipe, or an internally generated missile cannot prevent the other division from accomplishing its safety function. Thus, the system satisfies GDC 2.

The safety related operation of the ADS valves relies on a compressed gas supply that will be provided by the HPIN under most conditions. This nitrogen is supplied during normal operation from the AC system Nitrogen Storage Tank. During design basis events, nitrogen is supplied by accumulators charged by either the AC system during normal operation or by nitrogen bottles during periods when the AC system is unavailable. In addition, stored nitrogen can be used to replenish the accumulators or to supplement their operation. In these system line-ups, unfiltered nitrogen is supplied to the accumulators and ADS/SRV valve components. This does not comply with the guidance of ANSI MC 11.1-1976 which requires clean, dry, oil-free air (or nitrogen) to safety related components. Section 6 of the SSAR does not identify any filters or driers which are installed in the nitrogen gas system. (A draft version of a system diagram includes single filters of unknown integrity on the supply line from the AC system, however, these filters are not reflected in other SSAR figures nor are they discussed in the SSAR text.) Section 6 should address the nitrogen quality provided to safety related components by the High Pressure Nitrogen Gas Supply System. Compliance of the nitrogen supply system with the requirements of ANSI MC 11.1-1976 is an open item.

Section 9.3.6.1.2 indicates that the non-safety related Instrument Air System is also used as a backup to the nitrogen system when, during normal operation, the nitrogen gas supply pressure drops below a specified setpoint. This conflicts with GE's response to Question 430.218 which states: "Instrument air system does not serve as a backup to HPIN system during normal operation...." The resolution of this discrepancy is an open item.

However, in evaluating the IA system as a potential backup to the HPIN system, the staff found that the system complies with all aspects of the ANSI MC 11.1-1976 criteria with the exception of particulate size. The ABWR design proposes a 5 micron criteria for particulate size which is contrary to the 3 micron criterion of the ANSI standard. GE has not justified this aspect of the CA system design, therefore the IA system's compliance with the requirements of GDC 1 is an open item.

The review of the pre-operational testing of the CA systems and compliance with RG 1.68.3 is addressed in Section 14 of this SER.

Several open issues and inconsistencies have been identified as a result of the review of the information provided. The SSAR does not clearly identify which portions of the HPIN system are safety related (although this information has been provided in response to RAIs, it has not been completely incorporated into the SSAR). The SSAR does not discuss the failure mode of motor and air/nitrogen operated valves and conflicting information has been supplied regarding the failure position of valves F018A and B. Inconsistencies exist between drawings of the HPIN system in the SSAR and responses to RAIs - these inconsistencies include valve numbers (see Table 9.3.1-1), class break designations, and valve operator type (i.e., air or motor operated). An additional inconsistency involves whether or not the IA system is available as a backup to the HPIN system. Finally, the ability of the safety related air supply systems to meet the air quality requirements of ANSI MC 11.1-1976 is an open issue.

Based on the above review and contingent upon resolution of the identified open issues, the CA systems comply with the requirements of GDC 2 and applicable portions of RGs 1.29 and 1.68.3.

TABLE 9.3.1-1
EXAMPLES OF VALVE NUMBERING INCONSISTENCIES

Valve Numbers In Figure 6.7-1	Valve Numbers In Figure 20.3-55
AO F018A & B	MO F003A &B
MO F024A & B	MO F007A &B
F021A & B	F008A &B
MO F027A &B	MO F012A &B
MO F005	MO F203
MO F007	MO F200
F008	F209
F010 or F012	F216
No Number	Penetration X-71
No Number	Penetration X-71
No Number	Penetration X-72

9.3.2 Process and Post-Accident Sampling Systems

9.3.2.1 Process Sampling System

The process sampling system (PSS) is designed to collect water and gaseous samples contained in the reactor coolant system and associated auxiliary system process streams during all normal modes of operation. Provisions are made to ensure that representative samples (except from gaseous streams) are obtained from well-mixed streams or volumes of effluent by the proper selection of sampling equipment, sampling points, and sampling procedures. Additionally, grab samples are obtained for confirmatory analyses and to test for other chemicals. The reactor coolant sample lines penetrating the containment are each equipped with two normally closed, isolation valves which, if open, automatically close on a containment isolation actuation signal.

Section 9.3.2.1 of the ABWR Standard Safety Analysis Report (SSAR) contains insufficient information needed by the staff to evaluate conformance with SRP 9.3.2 in the following areas:

1. Under SRP 9.3.2, the PSS should include the capability of obtaining samples from at least the following points; main condenser evacuation system off gas, standby liquid control system tank, sumps inside containment and other locations given in SRP 11.5 in addition to those specified in the applicant's SSAR.

2. The guidelines of regulatory Position C.2 in RG 1.21, "Measuring Evaluating, and Reporting Radioactivity in Solid Wastes and Release of Radioactive Materials in

Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants," and positions in RG 1.56, "Maintenance of Water Purity in Boiling Water Reactors," should be used to meet the requirements of General Design Criteria 13, 14, 26, 63 and 64, or otherwise acceptable alternatives are proposed. Conformance with the above are not indicated in the SSAR.

3. In accordance with ANSI N13.1-1969, "Guide to Sampling Radioactive Materials in Nuclear Facilities," American National Standards Institute (1969), provisions should be made to assure representatives samples from gaseous process streams and tanks. This needs to be addressed in the design.

4. To meet 10 CFR 20.1(c) in keeping radiation exposures as low as reasonably achievable and the requirement of GDC 60 to control the release of radioactive materials to the environment, passive flow restrictions to limit reactor coolant loss from a rupture of the sample line should be provided.

5. To meet the requirements of GDCs 1 and 2, the seismic design and quality group classification of sampling lines for the PSS should conform to the classification of the system to which each sampling line and component is connected.

Until the above items are adequately addressed by GE, the staff concludes that the adequacy of the design of the process sampling system is an open item.

9.3.2.2 Post-Accident Sampling System

After the accident at Three Mile Island Unit 2, the staff recognized the need for an improved post-accident sampling system (PASS) to determine the extent of core degradation following a severe reactor accident. Criteria for an acceptable sampling and analysis system are specified in NUREG-0737, "Clarification of TMI Action Plan Requirements," (November 1980), Item II.B.3. According to this document the PASS should have the capability to obtain and quantitatively analyze reactor coolant and containment atmosphere samples without exposing any individual to radiation exceeding 5 rem to the whole body or 75 rem to the extremities (GDC 19, "Control Room") during and following an accident in which there is core degradation. Materials to be analyzed and quantified include certain radionuclides that are indicators of severity of core damage (e.g., noble gases, isotopes of iodine and cesium, and nonvolatile isotopes), hydrogen in the containment atmosphere, and total dissolved gases or hydrogen, boron, and chloride in reactor coolant samples.

The PASS design as described in SSAR Section 9.3.2.1.1 is not adequate. The NUREG-0737 Item II.B.3 criteria should be addressed to indicate the PASS provisions to satisfy each of the eleven specified criteria. The upper limit for activity levels in liquid samples of 1 Ci/cm^3 in GE's PASS design (Section 9.3.2.3.1) is not justified. NUREG-0737, Item II.B.3, Criterion 9 and RG 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," specify reactor coolant and sump gross activity sampling capability in the 1 Ci/ml to 10 Ci/ml range. Delaying sampling till the PASS sample radioactivity decays to 1 Ci/ml is unacceptable because of the inordinate and unjustified delay in obtaining sample radioactivity results. All PASS systems in operating reactor plants are presently designed with the capability of sampling liquids with up to 10 ci/ml radioactivity.

Based on the above, the staff concludes that the adequacy of the design of the ABWR PASS is an open item.

9.3.3 Equipment and Floor Drainage System

The equipment and floor drainage systems are designated as being non-safety related and are designed to transfer both effluents that are radioactive or potentially radioactive and those that are not radioactive or potentially radioactive. The radioactive drain transfer systems normally addressed as part of Section 9.3.3 are discussed in SSAR Section 9.3.8. Non-radioactive drainage systems are not considered to be part of the ABWR design. An interface requirement has been established requiring that there be no interconnections between the radioactive drain transfer system and the non-radioactive drain transfer system. This interface requirement provides sufficient guidance to a plant specific applicant to allow for the design of a non-radioactive drain transfer system that will meet the applicable requirements of SRP Section 9.3.3. The review of the radioactive drain transfer system portion of the equipment and floor drain systems is presented in Section 9.3.8 of this report.

9.3.6 Instrument Air (IA) System

The ABWR design includes four CA systems: the IA system, the SA system, the High Pressure Nitrogen Gas Supply System, and the Atmosphere Control System. This section of the SSAR provided information on the design of the IA system and was reviewed in accordance with SRP Section 9.3.1, "Compressed Air System."

The IA system provides dry, oil-free, compressed air for valve actuators and for non-safety related instrument control functions and for general instrumentation and valve services outside the containment. (All instrumentation and control systems located inside the containment are supplied with nitrogen gas during normal plant operation.)

Because the IA system is one of the four systems that perform functions addressed in SRP Section 9.3.1, the review of this system was performed as part of an integrated review of the ABWR CA systems. The results of this review are presented in Section 9.3.1 of this SER.

9.3.7 Service Air (SA) System

The ABWR design includes four CA systems: the IA system, the SA system, the HIPN system, and the AC system. This section of the ABWR SSAR provided information on the design of the SA System and was reviewed in accordance with SRP Section 9.3.1, "Compressed Air System."

The SA system is designed to provide compressed air of suitable quality for non-safety related functions. The SA system provides compressed air for services requiring air of lower quality than that provided by the IA system.

Because the SA system is one of the four systems that perform functions addressed in SRP Section 9.3.1, the review of this system was performed as part of an integrated review of the ABWR CA systems. The results of this review are presented in Section 9.3.1 of this SER.

9.3.8 Radioactive Drain Transfer System

The radioactive drain transfer system was reviewed in accordance with SRP Section 9.3.3, "Equipment and Floor Drainage System."

The radioactive drain transfer system is designated as non-safety related and is designed to collect radioactive or potentially radioactive effluents in equipment or floor sumps and then transfer the effluents to the liquid radwaste system for processing. The radioactive drain transfer system is identified as being partially within the scope of the ABWR design and partially within the scope of the plant specific design. The portions of the system within the ABWR design include the sumps, sump pumps (two per sump, each 100 percent capacity), sump instrumentation, and piping and valves from the sumps to the radwaste system. Floor drains and drain lines from the equipment to the equipment sumps are designated to be part of the plant specific design and outside of the scope of the ABWR design.

The piping, pumps, instrumentation, and valves of the radioactive drain transfer system are classified as non-nuclear safety-related with the exception of the containment (drywell) penetrations and containment isolation valves which are safety Class 2, designed in accordance with seismic Category I and Quality Group B criteria and the reactor building penetrations which meet ASME Code III Section 3 requirements. There is a discrepancy in the identification of these containment isolation valves between Figure 11-2 and Table 6.2-7. In Figure 11-2 these valves are identified as being air operated valves. Table 6.2-7 shows these containment isolation valves as being motor operated valves. Resolution of this discrepancy is an open item. All system piping designed to remain intact following a seismic event. The drain system is not the only method of leak detection available for any of the areas served by the system and is not taken credit for in the facility flood analysis. The designation of most of the system as non-nuclear safety-related is appropriate. However, the check valves which provide backflow protection for sumps in the ECCS equipment rooms should be classified as safety Class 3 and designed to seismic Category I and Quality Group C criteria. The classification of these valves is an open issue.

Two additional items were identified in the review of the systems and components shown in Figures 11.2-1 and 11.2-2. First, in Figure 11-2.1a the shower facility is shown discharging into the HCW Collector Tank. In all other figures and text the shower facility discharges to the HSD Receiver Tank. Second, the points at which changes in component qualification requirements occur were not identified, for example for the containment isolation valves described in the preceding paragraph. Resolution of the discrepancy in Figure 11.2-1a and the addition of component qualification requirements to the figures is an open item. An interface requirement preventing connections between radioactive and non-radioactive systems has been included in the ABWR SSAR. Additionally, the effluent from non-radioactive systems is to be monitored prior to discharge to ensure that there are no unacceptable (radioactive) discharges from the non-radioactive drain systems. This interface requirement and the monitoring of non-radioactive effluents will allow the applicant to design an equipment and floor drainage system that will be in compliance with the requirements of GDC 60.

The staff noted that SSAR Section 9.3.8.2 inaccurately refers to SSAR Section 9.3.9.1, when the appropriate reference is the interface criteria discussed in Section 9.3.12. Also, GE should revise the first design basis in Section 9.3.8.1 to clearly indicate, consistent with the staff's dialogue with GE, that only portions of the drain system are considered safety related. This is an open item.

Based on the above review and contingent upon resolution of the identified open issues, the radioactive drain transfer system meets GDCs 2, 4, and 60, with regard to protection against natural phenomena, environmental conditions, missiles, and the release of radioactivity to the environment, and to RG 1.29, Positions C.1 and C.2, concerning seismic classification of the system. Except as noted, therefore, the system would meet the criteria of SRP 9.3.3.

9.3.9 Hydrogen Water Chemistry System

Hydrogen water chemistry (HWC) reduces intergranular stress corrosion cracking (IGSCC) by using feedwater additions of hydrogen to decrease the oxidizing power of reactor water and reduce its aggressiveness toward plant material. To suppress IGSCC, reactor coolant conductivity must be maintained below 0.3 micro-Simon per centimeter and sufficient hydrogen must be added to the feedwater to reduce the electro chemical potential (ECP) below -0.23 volts (Standard Hydrogen Electrode) (See EPRI NP-4947-SR, "BWR Hydrogen Water Chemistry Guidelines: 1987 Revision," October 1988).

The SSAR in Section 9.3.9 indicates that the HWC system utilizes the guidelines given in EPRI NP-5283-SR-A, "Guidelines for Permanent BWR Hydrogen Water Chemistry Installations-1987 Revision," September 1987. The NRC SER which is included in this topical report (EPRI NP-5283-SR-A) specifies that an applicant referencing this document should indicate any exceptions or deviations from EPRI NP-5283-SR-A and justify that all identified exceptions or deviations will not affect the safety of the plant or the public.

Based on the staff's review of SSAR Section 9.3.9, it has determined that GE should reference EPRI NP-4947-SR for HWC guidelines or provide the exception/deviation information in the SSAR for the staff to review. Based on the above the staff concludes that the adequacy of the design of the hydrogen water chemistry system is an open item.

9.3.10 Oxygen Injection System

The oxygen injection system is designed to add sufficient oxygen (20 to 50 ppb) to suppress erosion/corrosion, general corrosion and corrosion product release in the condensate and feedwater systems. The oxygen injection system and oxygen storage facility should meet EPRI NP-5283-SR-A for design, operation, maintenance, surveillance, and testing to provide safe system and plant operation.

SSAR Section 9.3.10 does not indicate that the design of the system is based on the requirements included in the EPRI document noted above and therefore its adequacy is considered to be an open item.

9.3.11 Zinc Injection System

The control of build-up of radiation in reactor systems has been of concern in BWR plants. GE's review of this concern indicates that operating plants having 5 to 15 ppb of soluble zinc in the reactor water had lower piping dose rates than plants that had only trace amounts of zinc.

Laboratory tests confirmed that Co-60 deposition is greatly reduced in both normal and hydrogen water chemistry. Zinc injection into the feedwater system to provide reactor water concentrations of 10 to 15 ppb zinc during initial conditioning and 5 to 10 ppb over the fuel cycle will help keep radiation levels as low as possible; thereby, reducing personal exposure especially during outages.

Based on the foregoing, the staff concludes that Section 9.3.11 of the ABWR SSAR is acceptable.

9.5 Other Auxiliary Systems

9.5.1 Fire Protection System

Fire Protection requirements for nuclear power plants are provided for in 10 CFR 50, Appendix A, Criterion 3, and 10 CFR 50.48 Criterion 3 of Appendix A to 10 CFR Part 50 governed fire protection for nuclear power plants and was considered adequate until the Browns Ferry fire of March 22, 1975. This remains the most serious fire to date at a commercial domestic (U.S.) nuclear power plant. A committee was formed to investigate the fire and make recommendations based on their findings. Among the recommendations made by the investigation committee was that specific fire protection guidance should be developed that would supplement the general requirements contained in Criterion 3.

That specific guidance was published in Branch Technical Position (BTP) APCSB 9.5-1, "Guidelines for Fire Protection for Nuclear Power Plants" for new plants docketed after July 1, 1976, dated

May 1976 (revision of Section 9.5.1 of NUREG-75/087, dated May 1, 1976). Following publication of that detailed fire protection guidance, the staff developed Appendix A to BTP APCSB 9.5-1 (published in August 1976) to provide specific fire protection guidance for those plants docketed for construction permit before July 1, 1976. All licensees of operating plants and applicants of plants in various stages of design and construction were requested to review their plants against the guidance contained in Appendix A to BTP APCSB 9.5-1 and identify areas of compliance and noncompliance.

For those identified items of noncompliance, each licensee and applicant was asked to propose modifications to achieve compliance or show why compliance was not required.

By mid-1979, most plants had complied with most of the provisions of Appendix A to BTP APCSB 9.5-1. However, 18 open issues existed in various combinations at a total of 33 operating plants. The staff then developed § 50.48 and Appendix R to 10 CFR Part 50 (published on November 19, 1980 and effective February 17, 1981) as a means of resolving the remaining 15 open issues (reduced from the original 18 open issues) at plants licensed to operate before January 1, 1979. In addition, three sections of Appendix R were considered by the Commission to be so important that those provisions were required for all plants even if the staff had previously approved the design in those areas. The three sections of Appendix R that applied to all plants were III.G (Fire protection of safe shutdown capability), III.J. (Emergency lighting), and III.O (Oil collection system for reactor coolant pump).

Following publication of § 50.48 and Appendix R to 10 CFR Part 50, BTP APCSB 9.5-1 was revised (July 1981 as part of NUREG-0800) to include provisions of Appendix R so as to give additional guidance to those applicants that had docketed their application for a construction permit before July 1, 1976, and that were still being completed and preparing for operating licenses.

It is important to note that this subsequent fire protection guidance for operating plants, as well as for plants still being constructed, is derived from and represents deviations from the original guidance (BTP APCSB 9.5-1, May 1, 1976) developed for new plants, the category of plants that the GE ABWR design represents. The intention has always been that when any advanced reactor design was proposed, fire protection would be provided on the basis of the best technology available, not on the basis of methods allowed for plants already operating or in advanced stages of design and construction. On this basis, the fire protection system of GE ABWR has been evaluated against the criteria of SRP Section 9.5.1 (BTP APCSB 9.5-1, May 1, 1976). Fire protection guidance applicable to advanced reactor design also is contained in supplemental guidance documents that have been issued from time to time. Two examples of such supplemental guidance are the information pertaining to safe shutdown methodology contained in Generic Letter 81-12, dated February 20, 1981, and some important technical information, such as conformance with National Fire Protection Association codes and standards contained in Generic Letter 86-10, dated April 24, 1986.

The Commission has concluded that fire protection issues that have been raised through operating experience and through the External Events Program must be resolved for the ABWR. To minimize fire as a significant contributor to the likelihood of severe accidents for the ABWR, the staff concludes that current NRC guidance must be enhanced. As indicated in SECY-90-016, "Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements," dated January 12, 1990, the ABWR design must ensure that safe shutdown can be achieved, assuming that all equipment in any one fire area will be rendered inoperable by fire and that re-entry into the fire area for repairs and operator actions is not possible. Because of its physical configuration, the control room is excluded from this approach, provided an independent alternative shutdown capability that is physically and electrically independent of the control room is included in the design. The ABWR must provide fire protection for redundant shutdown systems in the reactor containment building that will ensure, to the extent practicable, that one shutdown division will be free of fire damage. Additionally, the ABWR designers must ensure that smoke, hot gases, or the fire suppressant will not migrate into other fire areas to the extent that they could adversely affect safe-shutdown capabilities, including operator actions. Because the layout of a nuclear plant is unit-specific, final plant specific design details will be reviewed by the staff on an individual basis based on the utility applicant submittals. The staff will require a description of safety-grade provisions for the fire-protection systems to ensure that the remaining shutdown capabilities are protected, as well as demonstration that the design complies with the migration criteria discussed above.

9.5.1.1 General Evaluation Fire Protection Program

GE has generally followed the NRC's concept of defense in depth with regard to fire protection as described in the Standard Safety Analysis Report (SSAR). The three steps of defense in depth and GE's implementation of these steps follow.

(1) To reduce the possibility of fire starting in the plant, GE used fire resistant and fire retardant materials in its design of GE ABWR to minimize and isolate fire hazards. GE used low-voltage multiplexed circuits and/or fiber-optic circuits in its design to eliminate the need for cable spreading rooms and substantially reduce the amount of combustible cable insulation and higher voltage ignition sources in the control room.

(2) To promptly detect and suppress fire, GE has provided adequate automatic detection and a suitable mix of automatic and manual fire suppression capability in the ABWR design.

(3) To ensure that any fire that might occur will not prevent safe shutdown of the plant even if fire detection and suppression efforts should fail, GE provided a fire protection program in its application.

The fire protection program described by GE in the SSAR is intended to protect safe shutdown capability, prevent release of radioactive materials, minimize property damage, and protect personnel from injury as a result of fire.

In addition to the three aspects of defense in depth outlined above, GE also considered such features of general plant arrangement as:

- access and egress routes
- equipment locations
- structural design features separating or isolatin redundant safety-related systems

- floor drains
- ventilation
- construction materials

The SSAR reflected the use of applicable National Fire Protection Association codes and standards in its design and layout of the ABWR. However, GE must identify any deviations from these codes and standards and describe in the ABWR fire hazards analysis the deviations and measures taken to ensure that equivalent protection is provided. The staff is currently reviewing the fire hazards analysis included in the appendix to SSAR Chapter 9, and will include its evaluation in the final safety evaluation. Therefore, this remains an open item.

9.5.1.2 Specific Features of Protection

9.5.1.2.1 Protection of Safe Shutdown Equipment

The ABWR design relies on 3-hour-rated fire barriers to separate safe shutdown equipment from the remainder of the plant and from redundant systems and components outside of primary containment. Inside containment GE proposed using (1) a combination of structural components that do not fully enclose equipment and (2) separation by horizontal distance of more than 20 feet. In Question 430.325 the staff stated that it does not accept methods that rely only on spatial separation for advanced reactors.

GE has modified Subsection 9A.3.1(8) of the ABWR application in response to Question 430.325. The clarification concerning use of 3-hour-rated fire barriers exclusively for separation of safety-related equipment in the Nuclear Power Block (NPB) areas outside containment is in accordance with the review criteria and is acceptable. The staff also recognizes the need for open communication between compartments inside containment in order to be able to relieve and equalize pressure following a high-energy line break. Therefore, the use of structural walls inside containment as fire barriers to separate safety-related systems (cabling, components and equipment), even though such walls may not fully enclose the equipment requiring separation, is acceptable in intent. GE has stated that all four safety divisions will be widely separated around containment so that a single fire will not be able to fail any combination of active components which could prevent safe shutdown. In addition, the ABWR containment will be inerted with nitrogen during power operation. This will prevent propagation of any potential fire inside containment.

GE has mentioned two areas outside containment and one inside containment that will not conform to the 3-hour-rated fire barrier separation criteria listed above. The three exceptions discussed below were well justified by GE and are acceptable to the staff.

(1) The main steam tunnel (MST) was called out as an exception to separation of redundant safety-related components outside containment.

In Question 430.335, the staff requested clarification of Subsection 9A.4.1.4.26(9) which stated that all safety-related valves in the MST would fail closed upon loss of actuation power. GE responded that the original submission was misleading and corrected the text. In addition, GE indicated that while not all valves fail closed, they are designed to fail with acceptable consequences. For example, power operated valves are backed up with air operated valves not subject to damage from the same fire, or redundant valves are located in another fire area.

(2) The main control room (MCR) was called out as an exception to separation of redundant safety-related components outside containment.

GE has described alternative shutdown capability that is spatially remote from, and electrically isolated from, the MCR. In the event of fire in the MCR, Control Room Operators will have full safe shutdown capability available at this alternative shutdown station.

(3) Inside containment was called out as an exception to separation of redundant safety-related components.

The entire containment is one fire area. As described above, the four shutdown trains enter containment widely spaced around the perimeter. This spacing assures that no single fire will be able to damage any combination of active components that would prevent safe shutdown. In addition, the ABWR containment will be inerted with nitrogen during power operation. This assures that any potential ignition/fire hazards inside containment will not propagate.

9.5.1.2.2 Passive Fire Protection Features

Passive fire protection features for the GE ABWR design consist of building assemblies (such as walls, partitions, floor-ceiling assemblies, columns, beams, and doors) and insulating materials (such as cable wraps and heat resistant coatings). Penetrations through the building assemblies such as doorways, hoistways, stairways, and cable trays and conduits are protected by appropriate fire rated doors, dampers, plugs, and seals. GE has stated its intention to select passive fire protection components of proven design, which have previously been tested and are listed by nationally recognized testing laboratories.

GE has paid particular attention to ventilation paths in the ABWR design. Each fire area has its own heating, ventilating and air conditioning (HVAC) system. This means that ventilation air supply, return and exhaust for any fire area is independent of all other fire areas, and that HVAC ducting does not penetrate 3-hour rated fire barriers separating fire areas. Therefore, fire dampers are eliminated from the GE ABWR design. This simplifies not only design of the ABWR HVAC systems, but also installation and maintenance of the system throughout the life of the plant.

In addition, GE intends to utilize the HVAC system for smoke removal in the event of fire. The staff asked for clarification of this smoke removal capability of the HVAC system in Question 430.320. The GE response to this question is not clear in a few particulars. While the intent of this design is acceptable, the staff will require more detailed information to complete its review. Specifically, GE should provide a system operational description as well as design information for components used in the smoke removal mode of operation. This will remain an open issue.

9.5.1.3 Fire Protection System Description

9.5.1.3.1 Fire Detection

The ABWR automatic fire detection systems, designed and installed in accordance with National Fire Protection Association Standards 72-D and 72-E, will be provided for all significant hazards and safe-shutdown components. (Note: National Fire Protection Association (NFPA) 72A, 72B, 72D, and 72E have been incorporated into NFPA 72, "Installation, Maintenance and Use of Protective Signaling Systems," and no longer exist as separate standards. Therefore, the reference to NFPA standards 72D and 72E should be changed to NFPA 72.) Detection capability will be provided for major cable concentrations, safe-shutdown-related major pumps, switchgear, motor control centers, battery and inverter areas. relay rooms, fuel areas, and all other areas containing appreciable in situ or potentially transient combustibles. Detector devices will be selected on the basis of type of anticipated fire and located on the basis of ventilation, ceiling height, ambient conditions, and burning characteristics of the involved materials. Detection systems will alarm and be annunciated in the control room and will give a distinctive audible and, if necessary (to facilitate fire brigade identification of fire location), visual local alarm.

The staff concludes that the automatic fire detection capability to be provided for the GE ABWR meets the guidelines of Section IV.C.1 of BTP 9.5-1 and is acceptable.

9.5.1.3.2 Fire Protection Water Supply System

A dedicated fire protection water supply and distribution system will be designed and installed in accordance with National Fire Protection Association Standards 11, 13, 14, 15, 20, and 24 to meet the anticipated needs for fixed water suppression systems and manual hose stations.

The sprinkler systems in the reactor building and the wet standpipe systems in the reactor and control buildings are designed in compliance with ANSI B31.1 and analyzed to remain functional following a safe shutdown earthquake. A portion of the water supply system including a tank, a pump and part of the yard supply main are designed to these requirements also. The remainder of the water systems are designed to the appropriate fire protection standards. During normal operation the seismically designed and non-seismically designed systems are separated by normally closed valves and a check valve such that a break in the non-seismically analyzed portion of the system cannot impair the operation of the seismically designed portion of the system.

The water supply system is required to be a fresh water system, filtered if necessary to remove silt and debris. Two sources with a minimum capacity of 300,000 gallons for each source is provided. If the primary source is a volume limited supply such as a tank, a minimum of 120,000 gallons must be passively reserved for use by the seismically designed portion of the suppression system. This reserve will supply two manual hose reels for 2 hours. A motor driven pump is in the train designed to remain functional following the safe shutdown earthquake. Its power is supplied from a non-Class 1E bus which is fed by one of the diesel generators. A jockey pump to keep the system pressurized is provided.

The turbine building is provided with modified Class III standpipes, hose reels and ABC portable extinguishers throughout the building. In addition, the following fire suppression systems provide primary fire suppression capability to the following areas:

(1) Automatic closed head sprinkler systems are provided in the open grating area of the three floors under the turbine.

(2) Deluge foam-water sprinkler systems are provided in the lube oil conditioning area and the lube oil reservoir area.

(3) A deluge sprinkler system is provided in the hydrogen seal oil unit area.

(4) A preaction sprinkler system is provided in the auxiliary boiler area.

The turbine building fire suppression systems receive water from the portion of the supply system which is not required to be seismically analyzed for safe shutdown earthquake.

The main power, unit auxiliary and reserve transformers are provided with deluge water spray suppression systems. The systems are automatically actuated by flame or temperature detectors. An oil and water collection pit is provided beneath each transformer. Drains away from buildings and transformers are provided for each pit. Shadow type fire barrier walls are provided between adjacent transformers.

Alarm systems, both manual and automatic, are provided in all areas of the plant as passive systems. They alarm without controlling an extinguishing function.

The two fire pumps will be located in separate fire areas cut off from each other and from the rest of the plant by 3-hour-rated fire barriers.

The fire-main loop in the yard will be designed and installed with sectional control valves that will deliver total fire flow to all automatic and manual fire suppression systems and manual hose stations even if the shortest portion of the water distribution piping is out of service.

On the basis of GE's commitment that the fire water supply and distribution system will conform to the applicable National Fire Protection Association Standards mentioned above, the staff concludes that the system meets the guidelines of Section IV.C.2 of BTP APCSB 9.5-1 and is acceptable.

9.5.1.3.3 Water Fire Suppression Systems

Automatic water and foam fire suppression systems will be installed over major fire hazards that will be identified by the fire hazards analysis as described above. The system will be designed and installed in accordance with National Fire Protection Association Standards 11, 13, and 15.

Standpipe and hose stations will be installed throughout the plant on the basis of needs identified in the fire hazards analysis. The standpipe systems will be designed and installed in accordance with National Fire Protection Association Standard 14. Each hose station will be equipped with a maximum of 100 feet of 1½-inch hose and an adjustable on/off spray nozzle that are listed or approved by a nationally recognized testing laboratory.

Pressure-reducing orifices will be installed at each hose station as required, to ensure that excessive pressures are not delivered to the each nozzle.

Exterior hydrants and hose houses will be provided according to needs identified in the fire hazards analysis. They will be designed and equipped in accordance with National Fire Protection Association Standard 24.

Control and sectionalizing valves in the fire water system will be electrically supervised and will be indicated in the main control room.

GE ABWR will not include floor penetrations that are susceptible to the potential of channeling water from fire extinguishing operations in one redundant fire area to an adjacent fire area. Floor penetrations will only be used for interconnections within one train of safe-shutdown equipment.

The above details concerning the fire protection water distribution and extinguishing systems conform to the guidelines contained in Sections IV.C.2 and 3 of BTP APCSB 9.5-1 and are acceptable, pending staff approval of an acceptable fire hazards analysis. This is considered an open item.

9.5.1.3.4 Gaseous Fire Suppression Systems

GE originally described an Automatic Carbon Dioxide (CO_2) fire suppression system for protection of the Emergency Diesel Generators (EDGs). The staff asked for clarification of this protection in Question 430.318. In GE's response to Question 430.318, the protection for the EDGs was changed from CO_2 to an automatic foam sprinkler protection. The automatic foam system, in addition to providing appropriate fire protection for the EDGs, will permit continued operation of the diesels in the event of operation of the foam sprinkler system.

Details of the automatic foam sprinkler system are consistent with the guidelines provided in Section IV.D.9 of BTP APCSB 9.5-1 and are acceptable.

9.5.1.3.5 Fire Extinguishers

Portable fire extinguishers will be provided in areas with in situ or potentially transient combustibles. Extinguishers will be chosen on the basis of the anticipated type of fire in the area and the effect of the extinguishing agent on equipment in the area. Selection, installation, and maintenance of the portable extinguishers will be done in accordance with provisions of National Fire Protection Association Standard 10. This conforms to the guidelines of Section IV.C.6 of BTP APCSB 9.5-1 and is acceptable.

9.5.1.4 Fire Protection Support Systems

9.5.1.4.1 Emergency Communication and Lighting

Portable radio communications will be provided for fire brigade and plant operations personnel during a fire incident. This communication system will have a distinct and separate frequency so that plant security force communications and actuation of protection relays will not be affected. The portable radio communication system will use fixed repeaters, as necessary, to ensure communications capability with any location in the station from the control room. The fixed repeaters will be arranged and protected so that exposure to fire damage will not disable the entire system.

Sealed-beam emergency lights with individual 8-hour battery supplies will be provided in areas that must be occupied for safe shutdown and in routes used for access and egress to these locations. The lighted areas will include areas where operator actions occur if the control room is evacuated. In addition to the sealed-beam 8-hour emergency lights, portable sealed-beam battery-powered hand-held lights will be provided for use by fire brigade and plant operations personnel during a fire incident.

In Question 430.321, the staff requested additional information regarding battery powered emergency lights located in harsh (extreme high or low temperature) environments. In its response to Question 430.321, GE revised Section 9.5.3.1.1(5) as follows:

(f) Non-essential battery pack lamps shall be self-contained units suitable for the environment in which they are located.

(g) The light fixtures for essential battery packs may be located remote from the battery if the environment at the lamp is not within the qualified range of the battery. Alternatively, lamps powered from the station batteries may be provided.

These details conform to the guidelines of Section IV.B.5 of BTP APCSB 9.5-1 and are acceptable.

9.5.1.4.2. Emergency Breathing Air

Emergency breathing air will be provided for fire brigade and control room personnel. The breathing air will be delivered by a self-contained apparatus or a storage reservoir. Full-face positive-pressure masks approved by the National Institute for Occupational Safety and Health will be used by all personnel required to use emergency breathing air.

A minimum of 10 self-contained breathing units will be provided for fire brigade use. Each unit will be provided with two extra air bottles located on site. Rated service life for the self-contained units will be a minimum of 1/2 hour. In addition to the two extra bottles for each self-contained unit, compressors will be provided so that exhausted air bottles may be quickly replenished. The compressors will operate in areas free of dust and contaminants and will be powered from a vital power bus so that breathing air is available if offsite power is lost. These provisions for emergency breathing air conform to the guidelines contained in Section IV.B.4 of BTP APCSB 9.5-1 and are acceptable.

9.5.1.4.3 Curbs and Drains

Floor drains and curbs that are sized to remove expected fire fighting water flow will be provided in areas protected by fixed water fire suppression systems or hand-held hose lines if water accumulation will cause unacceptable damage to safety-related equipment. Water drained from areas that may contain radioactivity will be properly collected, analyzed, and treated before being discharged to the environment.

Floor drains located in areas containing combustible liquids will be designed so that these liquids cannot flow back into safety-related areas through the drainage system.

These provisions for curbs and drains conform to the guidelines contained in Section IV.B.1 of BTP APCSB 9.5-1 and are acceptable.

9.5.1.4.4 Smoke Control

Smoke will be removed from each area by the normal ventilation systems. Release of smoke that may contain radioactive materials will be monitored to ensure compliance with applicable guidelines and regulations.

The general arrangement of the GE ABWR design of safe-shutdown trains features a high degree of separation with no piping and minimal cabling interconnections. With such a physical arrangement, the ventilation system can become the most likely pathway for fire propagation and smoke dispersal. The GE ABWR will employ separate, dedicated ventilation systems for each fire area containing redundant trains of safe-shutdown equipment. This arrangement of the ventilation systems serving the areas containing safe-shutdown equipment will facilitate the venting of smoke originating in one area containing safe shutdown equipment and preclude spreading of this smoke to the redundant area containing safe-shutdown equipment and is generally acceptable.

As discussed in Section 9.5.1.2.2 above, GE has designed the normal HVAC system to serve as a smoke removal system for each fire area in the event of a fire. However, the staff still requires some clarification on details of operations of the HVAC systems in the smoke removal mode of operation.

9.5.1.4.5 Access/Egress Routes

Clearly marked fire exit routes will be provided for each fire area. These routes will be designed to comply with applicable life safety codes and standards. These provisions for access and egress routes conform to the guidelines contained in Section IV.B.4.(f) of BTP APCSB 9.5-1 and Section III.G of Appendix R to 10 CFR Part 50 and are acceptable.

9.5.1.4.6 Construction Materials and Combustible Contents

Noncombustible materials having radiant energy heat flux equal to or less than 50 kW/cm2 will be used for interior wall and structural components, thermal insulation, radiation shielding, soundproofing, interior finishes, and suspended ceilings.

Transformers located inside fire areas containing safety-related equipment will be of the dry type, insulated with noncombustible liquid or separated from safety-related equipment by 3-hour-rated fire construction.

These provisions comply with the intent of the guidelines contained in Sections IV.B.1.(d) and (g) of BTP APCSB 9.5-1 to use only noncombustible materials for interior finish and are acceptable.

Interface Requirement:

With regard to noncombustible liquid insulated transformers, care must be taken to ensure that the insulating liquid does not present any unacceptable health hazards to employees in the event of release of the material to the building environment. Consideration of this hazard should be included in each plant specific application referencing the GE ABWR design.

9.5.1.4.7 Interaction with Other Systems

With three trains of safe shutdown capability provided, and given the ABWR design assumption of separate fire areas for each shutdown train which can survive total loss of all equipment in any fire area, the question of vulnerability of safe shutdown equipment to fire protection water is not applicable. Safe shutdown equipment in the ABWR design requires no special protection from the effects of fire protection water suppression systems failures.

Pipe rupture criteria will be used to ensure that the flood inventory in fire protection piping will not cause damage to safety-related equipment. Drains and sumps in the NPB will be sized to control maximum flood inventory of fire protection piping.

These provisions comply with the guidelines of Section IV.C.3 and IV.B.1(i) of BTP APCSB 9.5-1 and are acceptable.

9.5.1.4.8 Preoperational Testing

All of the active components of the entire plant fire protection system(s) will pass a preoperational acceptance test in accordance with the appropriate National Fire Protection Association Standard governing design and installation of the system. Components and systems subject to passing the preoperational testing before being placed in service include:

- fire pumps - controls, flow volume and pressure
- water distribution - flush and hydrostatic

- control valves

- fire detection and alarm systems including electronic supervision for other fire detection and fire suppression systems

- water fire suppression systems

- emergency radio communication systems

- emergency lights

- emergency breathing air systems and components

The above preoperational testing requirements are consistent with the guidance contained in BTP APCSB 9.5-1 and are acceptable.

9.5.1.5 Administrative Controls

The description of administrative controls that will be established to govern various details of operations of the plant conform to guidelines of BTP APCSB 9.5-1 and are acceptable. However, a detailed review and acceptance of the administrative controls will be performed during the plant-specific licensing process of an application referencing the GE ABWR design. Items of interest under the administrative controls review will include:

- control of combustible materials such as combustible/flammable liquids and gases, fire retardant treated wood, plastic materials, and dry ion exchange resins

* transient combustible materials and general housekeeping, including health physics materials

* open-flame and hot-work permits and cutting and welding operations

* quality assurance with respect to fire protection system(s) components, installation, maintenance, and operation

* qualification of fire protection engineering personnel, fire brigade members, and fire protection system(s) maintenance and testing personnel

* instruction, training, and drills provided to fire brigade members

9.5.1.6 Summary

The staff has found that fire protection to be provided for the GE ABWR is acceptable. Most of the fire protection features described in Section 9.5.1 of the ABWR application conform to the applicable sections of BTP 9.5.1. However, three exceptions to such conformance exist in (1) the main steam tunnel, (2) the main control room, and (3) inside containment. In each of these three instances, the exceptions were fully justified by GE and found acceptable by the staff.

The issue of smoke control is open because the staff requires additional information on the details of operation of the HVAC systems in the smoke removal mode of operation. The intent to

1

IMAGE EVALUATION
TEST TARGET (MT-3)

150mm

6"

1

IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6"

1

IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6"

provide independent HVAC systems for each train of safe shutdown equipment, which is located in a single fire area, is consistent with staff guidance and is acceptable.

The fire hazards analysis is currently under staff review and the evaluation will be included in the final safety evaluation for the ABWR. The closure of the issue related to the fire protection water distribution and extinguishing systems will be based on the approval of the fire hazards analysis.

Finally, the administrative controls to be provided for a new plant will be the responsibility of that particular applicant. Therefore, those administrative controls will be subject to the final staff review for each new plant.

9.5.2 Communication Systems

See the discussion of this topic in Section 7.7.1.15 of this report.

9.5.3 Lighting Systems

See the discussion of this topic in Section 8.3.5 of this report.

9.5.4 Diesel Generator Fuel Oil Storage and Transfer System

9.5.4.1 Diesel Generator Auxiliary Support Systems (General)

There are three standby (emergency) diesel generators in the ABWR design. Each diesel engine has the following auxiliary systems which are addressed in detail in the systems indicated: 1) fuel oil storage and transfer (Section 9.5.4.1); 2) cooling water

(Section 9.5.5); 3) starting air (Section 9.5.6); 4) lubrication (Section 9.5.7); and 5) combustion air intake and exhaust (Section 9.5.8). This section applies to all of these systems.

The diesel engine vendor has not been selected. The interface between the diesel engine and the support (auxiliary) systems, therefore, cannot be fully defined. Many component parameters (tank size, pump capacity, etc.) are dependent on the specific needs of the diesel engine selected at a particular site. For this reason, selection of the diesel engine support systems interface must be evaluated on a plant specific basis.

Most components of the diesel generators and their auxiliary support systems are located in the seismic Category I reactor building structure that provides protection from the effects of tornados, missiles, and floods. The diesel generator exhaust silencer is located on top of the reactor building, well above the PMF level and designed to be able to function during design basis events such as seismic vibrations, wind, hail, tornados, rain and snow storms. Fuel oil storage tanks, pump motors, valves and piping are located underground and are of seismic Category I construction. The only portions of the fuel oil storage and transfer system located above ground are the fill, sample and vent lines. Regarding these areas, the SSAR states that "in the unlikely event of a missile hitting and knocking off the storage tank vent no adverse effects will occur." The staff concluded that GE should provide additional information which describes provisions to minimize the effect of tornado missiles for these exposed components (see discussion of design basis tornado in Section 3.5.1.4 of the SER for information on the staff's assessment of designed protection from tornado missiles). This is an open item. With the exception of this item the system designs meet the requirement of GDCs 2 and 4 and RGs 1.115, "Protection Against Low-Trajectory Turbine Missiles," and 1.117, "Tornado Design Classification." Since the design provides that each diesel generator would have separate and independent auxiliary systems, GDC 5 is met.

The evaluation of the design of the diesel generator support (auxiliary) systems with respect to the effects of postulated pipe failures is addressed in Section 3.6 of this report. The adequacy of the fire protection for diesel generators and the associated auxiliary support systems is addressed in Section 9.5.1 of this report.

The designs of the diesel generator auxiliary systems have also been reviewed with respect to the recommendations of NUREG/CR-0660, "Enhancement of Onsite Emergency Diesel Generator Reliability." This report made specific recommendations on increasing the reliability of nuclear plant emergency diesel generators. Table 9.1 summarizes compliance of the ABWR auxiliary systems with the recommendations of NUREG/CR-0660.

NUREG/CR-0660 recommends that, as a minimum, operations and maintenance personnel receive training that is provided by the vendor, or is equivalent to vendor training since a lack of proper training has been a contributor to degradation of diesel generator reliability. Increased attention to investigative testing, replacement and adjustments as part of preventive maintenance and increased root cause analysis of system (component) failures are also recommended to improve diesel generator reliability. Programs to achieve these goals are site specific and will need to be addressed as part of the site specific application.

Criteria for the test loading of the diesel generators, loading the generator to 40 percent for 1 hour after up to 8 hours of no load or light load operation, have been included as an interface requirement. Diesel generator testing is to be performed in accordance with RG 1.108 requirements and other support system tests are to be performed periodically. No specific test intervals for these other system tests are provided. For all diesel generator auxiliary systems, the identification of test and calibration frequencies remain an open issue.

Provisions to reduce the effect of dust and dirt on diesel generator operation and reliability have been included as interface requirements. Instrumentation is to be located in dust tight steel cabinets with gasketed doors/openings and filtered louvers where ventilation is required. Ventilation is to be taken from a location high in the reactor building, approximately 30 feet above grade. Construction-related activity will be required to use appropriate dust control techniques. Concrete flooring is to be painted with concrete or masonry paint.

The vibration concerns expressed in NUREG/CR-0660 are addressed by locating free standing control panels on the floor above the diesel generator. All instruments and sensors that must be mounted on process equipment is to be protected against the effects of vibration.

Based on the above, assurance of diesel generator reliability will require the resolution of several issues. Some of these issues must be treated on a plant specific basis and are therefore best described as interface requirements.

Personnel training; test, test load, and maintenance procedures; and root cause analysis requirements need to be established. These personnel issues are to be addressed on a plant specific basis.

Table 9.1 Conformance to NUREG/CR-0660 Recommendations to the Diesel Generator Auxiliary Support Systems

Recommendation	Conformance	DSER Section
1. Moisture in air starting system	Yes[1]	9.5.6
2. Dust and dirt in diesel generator room	Yes	9.5.4.1
3. N/A to support systems		
4. Personnel Training	Site specific	9.5.4.1
5. Automatic prelube	Yes	9.5.7
6. Testing, test loading, and preventive maintenance	Site specific	9.5.4.1
7. Improve the identification of root cause of failures	Site specific	9.5.4.1
8. Diesel generator ventilation and combustion air systems	Yes	9.5.8
9. Fuel storage and handling	Yes	9.5.4.2
10. High temperature insulation		9.5.4.1
11. Engine cooling water	Yes	9.5.5
12. Concrete dust control	Yes	9.5.4.1
13. Vibration of instruments	Yes	9.5.4.1

[1] The air starting system includes provisions for air dryers of the capacity called for by NUREG/CR-0660. However, system design characteristics that will insure operation of the air dryer to design specifications were not provided and remain an open issue.

[2] Explicit conformance is considered unnecessary by the staff in view of the equivalent reliability provided by the design, margin, and qualification testing requirements that are normally applied to emergency standby diesel generators.

9.5.4.2 Diesel Generator Fuel Oil Storage and Transfer System

The design function of the fuel oil storage and transfer system is to provide a separate and independent fuel oil supply train for each diesel generator. Minimum storage capability for full load operation of each diesel generator for 7 days with replenishment of fuel is provided. GE has identified the required capacities of diesel generator support systems such as fuel tank capacity and system ratings, such as cooling system heat removal ratings, as interface requirements. The acceptability of the specified capacities selected will be verified for plant specific applications.

This review is based on the descriptive information and system diagrams supplied in the ABWR SSAR (primarily Chapters 8 and 9) and the information supplied by GE in response to RAIs in a GE letter dated November 15, 1990. This review was performed in accordance with SRP Section 9.5.4, Emergency Diesel Engine Fuel Oil Storage and Transfer System.

There are three standby diesel generators in the ABWR design. Each diesel engine fuel oil and transfer system consists of a day tank with sufficient capacity to supply fuel oil to power the diesel for eight hours; a fuel oil storage tank with a capacity sufficient to power the diesel 7 days; two fuel oil transfer pumps, an engine driven and a redundant DC driven fuel oil pump, both gravity fed, supplying fuel from the day tank to the engine fuel manifold; and associated piping, valves, instrumentation and controls. Each fuel oil storage and transfer system is independent and physically separated from the other two systems, each system is located in a separate quadrant of the reactor building. Thus, a single failure within any one of the systems will affect only the associated diesel generator.

GE identified in the SSAR that selection of the fuel oil transfer pump is an interface requirement. The staff noted that the type of motive power (required to be available during a loss of offsite power) should also be part of this interface criteria. GE should include this interface requirement in the SSAR. This is an open item.

All fuel oil storage and transfer system piping and components up to the diesel engine interface are designed to seismic Category I requirements. All piping and components, including engine mounted, meet RG 1.29 and will be designed, fabricated and installed in accordance with ASME Code Section III Class 3 requirements.

Instrumentation provided for the fuel oil storage and transfer system includes level indication for the day tank, temperature sensors at the intake and discharge (this second temperature sensor does not appear on the system P&ID for the fuel oil system) of the day tank and pressure indication for the suction of the engine mounted and DC driven fuel oil pumps. Level sensors provide signals to start the fuel oil transfer pumps, one starts on low level, a second on low-low level. At the low level a 60 minute supply (at full diesel generator load) of fuel oil is available for diesel generator operation. An interface requirement for provision of a stick gauge for measuring tank level is provided. From the information provided it is unclear whether storage tank level instrumentation is provided. In describing the commitment for a stick gauge provision, GE stated that level switches are provided to monitor tank level. However, a review of the diesel generator trouble alarms listed in Section 8.3.1.1.8.5 does not list low storage tank level as an annunciated condition and such level switches are not identified in the instrumentation section of

9.5.4. GE should verify that the stick gauge interface criteria is for the storage tanks and should resolve the apparent level instrumentation discrepancies. This is an open item.

The fuel oil storage tanks are located in three separate areas adjacent to the reactor building. The interior and exterior of both tanks and buried piping would have a protective waterproof coating. Also, an impressed current type cathodic protection to control corrosion of underground piping will be used.

SSAR Figure 9.5-6 depicts the standby diesel generator fuel oil system. Based on its review of this figure, the staff concluded that the fuel storage tanks and associated instrumentation should be added to the figure. The staff noted discrepancies between the text and Figure 9.5-6, regarding the optional characterization of the electric fuel oil pump. Also, GE's response to RAI Question 430.273 stated, in part, that "Two local fuel oil temperature indicators are provided (one in the suction line and one in the discharge line) from the day tank." Figure 9.5-6, however, identifies only one temperature sensor. GE should resolve these discrepancies to clearly identify whether the fuel oil pump is optional or not optional and to confirm the number of temperature sensors. The aforementioned discrepancies related to SSAR Figure 9.5-6 are an open issue.

Section III.5 of SRP Section 9.5.4 addresses the need to minimize the creation of turbulence of the sediment at the bottom of the fuel oil storage tank during refueling. To ensure continuous operation of the diesel generator while refueling, the ABWR design relies on duplex filters, strainers between the storage tank and the day tank and at the fuel oil pump suction to remove any sediment. The SSAR suggests that refueling would probably occur while the day tank is full which would allow time for sediment to settle before fuel is transferred from the storage tank to the day tank. The staff noted that refueling procedures should be established as an interface requirement to verify that the day tank is full prior to refilling, thereby minimizing the likelihood of sediment obstruction of fuel lines and any deleterious impacts on diesel generator operation. This is an open item.

GE has described a program to ensure that the diesel fuel oil is tested and maintained according to the appropriate ASTM and ANSI requirements. Fuel oil is to be sampled and tested monthly for quality and contaminants. New fuel will be visually inspected before addition to the storage tank and analyzed within two weeks for other required properties. Fuel oil not meeting all requirements will be replaced within a week. The system will be tested as part of the required diesel generator tests and hydrostatically tested prior to startup. Each fuel storage tank will be tested against ASME requirements every 10 years. The system design fuel oil quality and tests meet the requirements of RG 1.137.

Based upon the staff's review and contingent upon resolution of the identified open items, the design of the fuel oil storage and transfer system, meets the requirements of GDC 17 as related to the capability of the fuel oil system to meet independence and redundancy requirements, and RGs 1.9 and 1.137. The design also incorporates the recommendations of NUREG/CR-0660, and the appropriate industry standards (ANSI-N195-1976 and IEEE-Standard 387).

9.5.5 Diesel Generator Cooling Water System

The function of the diesel generator cooling water system is to maintain the temperature of the diesel engine within a safe operating range under all load conditions and to maintain the engine coolant preheated during standby conditions. The system should be designed to meet the requirements of GDCs 2, 4, 5, 17, 44, 45, and 46. The ability of the ABWR diesel generator cooling water system to meet GDCs 2, 4, and 5 is discussed in Section 9.5.4.1.

This review is based on the descriptive information and system diagrams supplied in the ABWR SSAR (primarily Chapters 8 and 9) and the information supplied in the GE response RAI's dated November 15, 1990.

The diesel generator cooling water system is a closed loop system that cools the engine jacket water, lube oil, and combustion air. The major components of this system include a jacket water heat exchanger, lube oil heat exchanger, combustion air heat exchangers (air intercooler and exhaust manifold), an expansion tank, two jacket water circulating pumps, an electric immersion heater, a jacket water keep warm system, a three way temperature control valve, and the required controls, alarms, instrumentation, piping, and valves. Heat generated during diesel generator operation is rejected to the reactor cooling water system through the jacket water heat exchanger. All system piping and components are designated ASME Code III Class 3, designed to seismic Category I requirements, and would be procured according to the requirements of 10 CFR Part 50 Appendix B.

There are 3 standby diesel generators in the ABWR design. Each diesel generator has a physically separate and independent engine cooling water system, as described in the proceeding paragraph. Each cooling water system is powered from the respective diesel generator's safety-related Class IE MCC. Therefore, the redundancy and single failure criteria of requirements of GDC 17 are met.

During operation of the diesel generators the temperature of the diesel engine cooling water is regulated through the action of three-way temperature control valves. When the standby diesel generators are idle, the cooling water is heated by an electric heater and maintained at 120°F (assuming ambient temperature of 60°F). GE indicated that the specific information regarding the design and capability of the cooling water system and the keep warm system are interface requirements. These requirements include the following:

1) jacket water circulating pump characteristics (NPSH and motive power source, i.e., shaft, engine, etc.)

2) the keep warm system description (design may or may not include provision for a keep warm pump)

3) temperature sensor selection (Amot type or equivalent)

4) heat removal capability of system (to be based on maximum permissible diesel engine overload output)

5) expansion tank size

6) expected water loss over 7 day period and system volume capacity (needed to ensure adequate volume is available to maintain system water level and pump NPSH without refill).

The staff noted a discrepancy between the text of Section 9.5.5 and Figure 9.5-7 regarding the circulating water pump. Section 9.5.5 identifies the jacket circulating water pumps as engine and motor driven while Figure 9.5-7 identifies both as being motor driven. This discrepancy is considered an open item. Additionally, the interface criteria of Section 9.5.13.6 states that the selection of the motive power for these pumps is an interface requirement. This implication disagrees with the text of Section 9.5.5 and Figure 9.5-7, wherein motive power for the pumps, although inconsistent, are clearly specified by GE. This is an open item.

Also for item 3 above, the selection of an Amot type or equivalent valve was not specifically identified as part of the interface criteria for the selection of this valve. The staff concluded that GE should include this selection information in the interface criteria for temperature sensor selection. This is an open item.

The diesel engine is also identified as having the capability to operate at full load for 2 minutes without secondary cooling. This interface requirement will ensure that the diesel engine can operate at full load in excess of the time required to restore cooling water (reactor service water and reactor cooling water) which are sequenced onto the emergency power supply within one minute following a loss of offsite power.

The diesel generator cooling water system conforms with RG 1.9 Position C.7 as it relates to engine cooling water protective interlocks. All trips are bypassed during LOCA conditions except low cooling water pressure and low differential pressure of secondary cooling water. Both of these trips are 2 of 2 logic trips (the diesel generator system protective interlocks are discussed in Section 8.3). The cooling water system is provided with an expansion tank and expansion tank vent line, both of which are to be located above the system piping and pump location. A static head will ensure that the pumps and piping are filled with water.

The applicant has stated that the operating procedures for the diesel generator will require the loading of the engine up to a minimum of 40 percent of full load (or lower per manufacturers recommendation) for 1 hour after up to 8 hours of continuous no-load or light load operation. Such no-load or light load conditions would exist for a LOCA with offsite power available. Procedures utilizing this interface criteria will meet the requirements of SRP 9.5.5 Item III.7.

The components of the diesel engine cooling water system can be periodically inspected through surveillance testing and monitoring instrumentation for pressure, temperature and level. The water system cooling water would be analyzed periodically to ensure that adequate quality is maintained. In addition, the diesel generator would be tested in accordance with RG 1.108 requirements. These characteristics meet the inspection and functional testing requirements of GDCs 45 and 46.

Based upon the staff's review, the design criteria and bases for the diesel generator cooling water system conform to GDCs 17 and 44, regarding redundancy and physical independence, and GDCs 45 and 46 regarding inspection and testability of the system. The design also meets the cited RGs and industry codes and standards, accommodates the recommendations of NUREG/CR-0660 and includes the capability to maintain stable diesel engine cooling water temperature under all load conditions. The design meets the requirements of GDCs 2, 4, 5, regarding the protection of equipment from environmental effects and sharing of system components. The system also meets the requirements of GDCs 45 and 46, regarding inspection and testability. Contingent upon resolution of the identified open items, the design of the diesel generator cooling water system meets the requirements of GDC 44 with regard to system operability. Interface requirements detailing the system heat removal capacity and keep warm system must be revised to include the information discussed above. This is an open item.

9.5.6 Diesel Generator Starting Air System

The design function of the diesel generator starting air system is to provide a supply of compressed air for starting the emergency diesel generator engines without external power. The air storage system is to perform its function in a manner that ensures that the time interval between a diesel engine start signal and a "ready to load" status is less than 13 seconds. (The first load block for the diesel generators are sequenced onto the diesel generators at 13 seconds). The system is to be designed to meet the requirements of GDCs 2, 4, 5 and 17. The meeting of the requirements of GDCs 2, 4, and 5 is discussed in Section 9.5.4.1.

This review is based on the descriptive information and system diagrams supplied in the ABWR SSAR (primarily Chapters 8 and 9) and the information supplied by GE in response to RAIs in a GE letter dated November 15, 1990.

There are three emergency diesel generators in the ABWR design. Each diesel generator has its own starting air system, separate and independent of the starting air systems for the other two diesel generators. Each starting air system consists of two 100 percent capacity sections, i.e., each section is capable of supplying sufficient air for five automatic or manual start attempts without recharging the air receiver tanks. The starting air system consists of two air compressors, two air receivers, four air admission valves (two redundant air admission valves on each of two engine starting air manifolds) and associated piping and valves to connect system components.

One compressor and one air receiver comprise one section of a diesel generator starting air system. Controls are provided to automatically start and stop each air compressor to maintain the required pressure in each air receiver. Each compressor can be manually started to restore pressure in the air receivers if needed. Each starting air system is equipped with an air receiver low pressure alarm which is indicated locally and displayed in the control room as part of a diesel generator alarm/annunciation refresh unit. Each starting air system is equipped with a blowdown connection at the bottom of the receiver which would be used periodically to manually blowdown the receiver to remove any accumulated water from the tank.

In response to requests for information the applicant indicated that each air dryer system is to be provided with an air dryer equipped with pre-filters and after-filters. Figure 9.5-8 does not specifically identify pre-filters and after-filters for the air dryers. Addition of these filters to the P&ID (or a statement specifically identifying the filters as an integral part of the air dryer component) is an open issue. Each diesel generator is completely separate and independent of the others so that a malfunction or failure in one starting air system does not impair the starting capability of the other diesel generators. Therefore, the independence and redundancy requirements of GDC 17 are met.

Several design parameters have been identified as future interface requirements to be determined once a diesel generator vendor has been selected. Interface requirements to be specified include the devices to crank the engine, air start requirements with regard to the duration of the cranking cycle, and the number of engine revolutions per start attempt. These interface requirements should dictate design parameters such as the volume and design pressure of the air receivers (sufficient for 5 start cycles per receiver) and compressor size (sufficient discharge flow to recharge the system in under 30 minutes). The staff believes that once established, these interface criteria should provide an adequate basis for the selection of component capacities. This is an open item.

The air compressors air storage tanks and valves and piping (up to the first connection on the engine skid) are designed in accordance with the requirements of ASME Code Section III Class 3 requirements and are seismic Category I. No other components of the starting air system are classified.

The starting air system description does not include a reference to coolers at the discharge of the air compressors, although Figure 9.5-8 includes after coolers located downstream of the starting air compressors. This discrepancy is an open issue. The ABWR SSAR also states that the starting air quality would comply with the diesel engine manufacturer s recommendation regarding dew point as opposed to the requirements stated in SRP Section 9.5.6 part II.4.j. The staff determined that this requirement should provide an equivalent level of protection as the criteria in the SRP.

On the basis of this review and pending resolution of the issues identified above and in Section 9.5.4.1, the diesel generator starting air system meets the requirements of GDCs 2, 4, 5, and 17, the guidance of the cited (in SRP Section 9.5.6) RGs, the recommendations of NUREG/CR-0660. The starting air system design is, therefore, acceptable except as noted.

9.5.7 Diesel Generator Lubrication System

The design safety function of the diesel generator lubrication system is to provide a supply of filtered lubrication oil to the various moving parts of the diesel engine (including pistons and bearings) during engine operation and during periods of standby to enhance first-try-start reliability. The ability of the system design to meet the requirements of GDCs 2, 4, and 5 is discussed in Section 9.5.4.1 of this report.

This review consisted of the descriptive information and system diagrams supplied in the ABWR SSAR (primarily Chapters 8 and 9) and the information supplied by GE's response to RAIs dated November 15, 1990. The basis for acceptance in the review was conformance of the design to GDC 17, regarding redundancy and physical independence, the guidance and additional acceptance criteria of SRP 9.5.7, and the recommendations of NUREG/CR-0660.

The major components of the lubrication system include the engine lube oil pump (within the engine frame), an engine driven pump, an oil cooler, a generator shaft lube oil cooler, an electric lube oil heater, a keep warm circulating pump, oil filter and strainer. Local alarms signal low oil pressure, high oil temperature, and low oil level. These signals are part of a general diesel generator trouble alarm located in the control room. Low oil level alarms are described in Section 8.3.1.1.8.5 and refill is described in Section 9.5.7 as being performed on the indication of low level (a lube oil supply pump actuated by a low level indication in the engine sump). The staff noted that Figure 9.5-9 does not show any level indication for the lube oil system. This is an open item.

Each of the diesel generator lubrication systems is completely independent of the other two systems and is dedicated to the support of a single diesel generator. A malfunction in one lubrication system will not impair the operational capability of the remaining lubrication systems or diesel generators. This meets the requirements of GDC 17 regarding system independence and the single failure criteria.

The staff requested GE to provide the following specific design criteria; pump flows, operating pressure, temperature differentials, cooling system heat removal capabilities and electric heater characteristics for the diesel generator lubrication system. In response to this request, GE stated that the "lubrication system design criteria will be furnished as an interface criteria after selection of the diesel vendor is finalized." GE also stated that the protective features to prevent crankase explosions and features to mitigate the consequences of such an event (such as relief ports) are vendor specific and would also be included as interface criteria after selection of a diesel engine vendor. Identification of these design criteria as interface criteria is an open item.

The diesel generators have provisions for maintaining lubrication oil temperature and circulating heated lubrication oil under pressure to the moving parts of the diesel engine while the engine is in the standby mode. A lube oil priming pump will operate intermittently to keep the lube oil piping pressurized. This same pump is used in conjunction with the lube oil heater to maintain system temperature. On low lube oil temperature both the heater and priming pump will automatically start thereby circulating heated oil throughout the system. The priming pump discharge pressure switch is Class 1E. The diesel generator lubrication system conforms to the recommendations of NUREG/CR-0660 with regard to enhancing diesel engine starting reliability.

All diesel generator lubrication system piping and components are to be designed in accordance with ASME Code, Section III, Class 3 requirements or ANSI B31.1 requirements and are to be seismic Category I. The diesel engine interface for the lubrication system has not been identified (a diesel engine vendor specific definition) and therefore the components to be designated to meet the ASME requirements have not been separately identified from those required to meet the ANSI standard. To meet NRC requirements all components up to the diesel engine interface must meet the ASME requirements. The NRC staff has, in the past, accepted the ANSI classification of engine mounted components provided they are pressure tested to 1.5 times design pressure and information to that effect documented. Recognizing that the keep warm heater and the priming pump do not have to be nuclear safety grade the classification of components in the lubrication system is an open item. Components that are to meet the ASME requirements must be identified and those that are to meet the ANSI requirements with the pressure testing provision must also be identified. This is an open item.

The diesel generator lubrication system conforms to RG 1.9, Position C.7 as it relates to diesel engine lubrication system protective interlocks. All trips associated with the lubrication system are bypassed during LOCA conditions. (The diesel generator system protective interlocks are discussed in Section 8.3.)

The quality of the lubrication oil is maintained through periodic sampling and analysis of the lubrication oil. Access to the lubrication system is controlled. The system is located in locked diesel generator rooms, thereby limiting the possibility of contamination.

Based on their review, the staff concluded that the design of the diesel generator lubrication system meets GDCs 2, 4, 5 and 17, the cited RGs, SRP 9.5.7 and the recommendations of NUREG/CR-0660 with the exceptions of the open items identified above and in Section 9.5.4.1 of this report. The system design is therefore acceptable, except as noted.

9.5.8 Diesel Generator Combustion Air Intake and Exhaust System

The design function of the diesel generator combustion air intake and exhaust system is to supply filtered air for combustion to the engine and to dispose of the engine exhaust to the atmosphere. The compliance of the system design with the requirements of GDCs 2, 4, and 5 is discussed in Section 9.5.4.1 of this report.

This review consisted of the descriptive information and system diagrams supplied in the ABWR SSAR (primarily Chapters 8 and 9) and the information in GE's response to RAI's dated November 15, 1990. The bases for the acceptance in the review were conformance of the design to GDC 17, regarding redundancy and physical independence, the cited RGs, SRP 9.5.8, NUREG/CR -0660, and industry codes and standards. The system was also assessed regarding its ability to provide sufficient combustion air and release of exhaust gases to enable the emergency diesel generator to perform on demand.

Combustion air for each diesel generator is taken from the associated inlet air cubicle above the diesel generator room and passed through floor grates into the combustion air inlet plenum, duct work, intake silencer, turbocharger and air intercooler. The exhaust gas passes through the turbocharger and the exhaust ducting to the exhaust silencer located on the roof of the reactor building. Each of the three diesel generators is provided with a separate and independent combustion air intake and exhaust system. There are no active components (such as louvers) that can fail and obstruct the inlet or outlet air flow paths. Thus, the system independence, redundancy, and single failure criteria requirements of GDC 17 are met. System design air flow capacity has not been specified. As with the other diesel generator auxiliary systems this design characteristic will be dependent on selection of a diesel generator vendor. Selection of a combustion air flow capacity sufficient to ensure complete combustion is an interface requirement. This is an open item.

The diesel generator combustion air intake and exhaust system conforms to RG 1.9 Position C.7 as it relates to system protective interlocks. All diesel engine combustion air intake and exhaust system piping and components are designed to seismic Category I and ASME Code Section III Class 3 requirements. Engine mounted piping and components beyond the engine interface are considered part of the engine assembly and are seismic Category I as part of the diesel engine package.

The combustion air intakes are located on the side of the reactor building and are protected (by vertical grills) from tornado missiles. The intakes are located 11.5 meters (37.7 feet) above grade and are designed to minimize any effects from dust and debris through the use of vertical grills set into the reactor building wall with filters located behind the grills. The intakes are protected from flooding by their location. The diesel generator exhausts are partly housed within the reactor building with the exhaust silencer located on the roof of the reactor building. The design basis for the silencer requires that it be seismically qualified and able to withstand the effects of tornadoes. However, the means of protection from tornado missiles has not been adequately discussed. This is an open item. All other portions of the system meet the requirements of GDC 4, RG 1.115 and RG 1.117, and the recommendations of NUREG/CR-0660.

Combustion air is not taken from the diesel generator room. Combustion air and ventilation air enter the reactor building through common filters into the air inlet cubicle. Prior to entering the diesel generator room the air is directed into separate inlet plenums for ventilation and combustion air. This design meets the intent of the recommendations of NUREG/CR-0660 regarding combustion air and dust and dirt control in the portion of the reactor building housing the diesel generators.

The SSAR provides no information on the design of the Diesel Generator Combustion Air Intake and Exhaust System which extends from the crankcase vacuum blowers to the outside environment. Information regarding this area, identified in SSAR Figure 9.5-6, should be provided. This is an open item.

On the basis of this review, the emergency diesel engine air intake and exhaust system meets GDCs 2, 4, 5, and 17; NUREG/CR-0660; the cited RGs; SRP 9.5.8; and industry codes and standards. It is, therefore, acceptable, except as noted above and in Section 9.5.4.1.

9.5.9 Suppression Pool Cleanup System

(See Section 9.1.3, Spent Fuel Pool Cooling and Cleanup System, for a discussion of the spent fuel pool cleanup system)

The Suppression Pool Cleanup System (SPCU) purifies water in the suppression pool, fills the upper pools from the suppression pool before a refueling outage, and serves as a backup source of makeup water to both the fuel pool and the Reactor Cooling Water (RCW) system surge tanks when normal makeup from the condensate system is not available.

Suppression pool water is pumped to the fuel pool cooling and cleanup system where the water is cleaned and purified using a filter-demineralizer. It is then returned to the SPCU system. The system is isolated from containment on a LOCA signal (low reactor water level and high drywell pressure). The power for the systems containment isolation valves is supplied by Class 1E power buses.

GE has committed to testing the system in accordance with ASME Section III, Class 2 and 3 requirements. The containment isolation valves will be manually tested periodically to assure operability. The SPCU system serves no safety-related function. Failure of the system does not compromise any safety-related system nor does it prevent safe reactor shutdown.

STEAM AND POWER CONVERSION SYSTEM

10.2 Turbine Generator

10.2.3 Turbine Integrity

GE-ABWR turbine rotors and parts are made from vacuum melted or vacuum degassed Ni-Cv-Mo-U allcy steel by processes which minimize flaw occurrence and provide adequate fracture toughness.

The fracture appearance transition temperature (50 percent FATT) as obtained from Charpy tests will be no higher than 0 degree for low-pressure turbine disks. The Charpy V-notch energy at the minimum operating temperature of low pressure disks in the tangential direction will be at least 60 ft-lbs.

The ratio of fracture toughness (K_{IC}) of the disk material to the maximum tangential stress at speeds from normal to 115 percent of rated speed is at least 2 square inches. However, K_{IC} will be used only on materials that exhibit a welldefined Charpy energy and FATT curve and are strain-rate insensitive. The applicant referencing the GE-ABWR design will submit the test data and the calculated toughness curve to the NRC staff for review. Sufficient warmup time and metal temperature will be specified in the turbine operating instruction to assure that toughness will be adequate to prevent brittle fracture during startup.

The turbine rotor design will be a solid forged monoblock rotor rather than shrink-on disks. The forged rotor will not be as susceptible to stress corrosion cracking as experienced in the shrink-on disks. The combined stresses of low-pressure turbine disk at design overspeed due to centrifugal forces, interference fit, and thermal gradients will not exceed 0.75 of the minimum specified yield strength of the material.

The design overspeed of the turbine will be 5 percent above the highest anticipated speed resulting from a loss of load. The applicant referencing the GE-ABWR design will provide the basis for the turbine design overspeed.

The inservice inspection (ISI) for the turbine assembly includes high and low-pressure turbine rotor, low pressure turbine buckets, turbine shafts, couplings, and couplings bolts. During plant shutdown coinciding with the ISI schedule for ASME Section III components, turbine inspection will be performed at least once every 10 years. The low pressure turbines in operating nuclear plants are inspected, on average, once every 5 operating years. Most of the current turbines, however, are of shrunk-on design which is more susceptible to stress corrosion cracking than the forged monoblock rotor in the GE-ABWR turbine. However, the applicant reference the ABWR design should submit actual turbine inspection schedule after third refueling outage. The actual turbine inspection schedule should be based on probability caluclation of turbine missile generation. The probability should be 1.0 E-4 for favorably oriented turbine and 1.0 E-5 for unfavorably oriented turbine. The NRC approved methodology is discussed in NUREG-1048, "Safety Evaluation Report related to the Operation of Hope Creek Generating Station," Supplement 6, July 1986.

The ISI schedule and program for the overspeed protection valves (e.g., main stop valves, control valves, extraction non retur valves, and combined intermediate valves) are adequate as comparing to the current standard Technical Specification (NUREG-0123, Rev. 3).

The staff concludes that the integrity of the turbine disk is acceptable and meets the relevant requirements of GDCs 4 of 10 CFR Part 50. This conclusion is based upon the following:

The applicant has met the requirements of GDC 4 of 10 CFR Part 50 with respect to the use of material with acceptable fracture toughness and elevated temperature properties, adequate design, and the requirements for preservice and inservice inspections. The applicant has described his program for assuring the integrity of low-pressure turbine disks by the use of suitable materials of adequate fracture toughness, conservative design practices, and provide reasonable assurance that the probability of failure with missile generation is low during normal operation, including transients up to design overspeed.

10.3 Main Steam Supply System

10.3.6 Steam and Feedwater System Materials

The Class 2 materials specified for the main steam and feedwater system satisfy Appendix I to Section III of the ASME Code, and to parts A, B, and C of Section II of the code. The fracture toughness properties of the ferritic materials of these components meet the requirements of NC-2300, "Fracture Toughness Requirements for Materials," of ASME Code, Section III and RG 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containg Components of Nuclear Power Plants."

The materials selection and fabrication follow RG 1.71, "Welder Qualification for Areas of Limited Accessibility," and RG 1.85, "Code Case Acceptability ASME Section III Materials," RG 1.37, "Quality Assurance Requirements for Cleaning of fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants," and ANSI Standard N45.2.1, "Cleaning of fluid Systems and, Associated Components During Construction Phase of Nuclear Power Plants." The non-destructive examination of the steam and feedwater piping meets the acceptance criteria in NC-2550 through 2570 of the ASME Code, Section III.

The staff concludes that the main steam and feedwater system materials are acceptable and meet the relevant requirements of 10 CFR 50.55a GDCs 1 and 35, and Appendix B to 10 CFR Part50.

10.4 Other Features

10.4.6 Condensate Cleanup System

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

The staff has reviewed the sampling equipment, sampling locations, and instrumentation to monitor and control and the CCS parameters. On the basis of this review, the staff finds that the instrumentation and sampling equipment provided is adequate to monitor and control parameters. Based on its review of the applicant's criteria and design bases for the CCS and the requirements of the system, the staff concludes that the design of the CCS and its supporting systems conforms to staff guidelines (RG 1.56, "Maintenance of Water Purity in Boiling Water Reactors"), and is, therefore, acceptable.

However, Section 10.4.6 of the ABWR Standard Safety Analysis Report (SSAR) contains insufficient information needed by the staff to evaluate conformance with SRP 10.4.6 in the following areas:

1. Under SRP 3.6.1, the effects of high and moderate energy piping failures to assure that other safety-related systems are not rendered inoperable must be evaluated.

2. Under SRP 12.2, the adequacy of the shielding design of the CCS polisher vessels must be evaluated.

   Although the ABWR SSAR indicates conformance with RG 1.56, "Maintenance of Water Purity in Boiling Water Reactors," in order to meet the requirements of General Design Criteria 14 and to mitigate the potential of intergranular stress corrosion cracking the applicant should indicate conformance with EPRI NP-4947-SR, "BWR Hydrogen Water Chemistry Guidelines: 1987 Revision, October 1988."

4. In Section 10.4.6.3 of the ABWR, the applicant should indicate that the CCS removes condensate system corrosion products and impurities from condenser leakage in addition to radioactive material, activated corrosion products and fission products that are carried-over from the reactor.

Based on the foregoing, the staff concludes that the adequacy of the condensate cleanup system based on SRP requirements remains an open item.

12 RADIATION PROTECTION

Chapter 12 of the Standard Safety Analysis Report (SSAR) submitted by General Electric Company (GE) provides information on the radiation protection features and estimated occupation exposure associated with the ABWR design. The radiation protection measures incorporated in the ABWR are intended to ensure that internal and external occupational radiation exposures to plant personnel, contractors and the general population, as a result of plant operations, including shutdown periods and anticipated operational occurrences (AOOs), will be within applicable limits of regulatory criteria and will be as low as is reasonably achievable (ALARA).

The staff reviewed the SSAR to determine whether the design of the ABWR is sufficient to permit plant operations while maintaining radiation doses to personnel within the limits of 10 CFR 20 and that ABWR design features are consistent with the guidelines of Regulatory Guide (RG) 8.8 "Information Relevant to Ensuring That Occupational Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable" (Rev. 3). (The draft safety evaluation review was based on the requirements included in the original version of 10CFR Part 20, however, the final safety evaluation review and documentation will be based on the revised Part 20 issued on May 21, 1991. The staff expects that GE will modify the SSAR as necessary to meet the new requirements.)

On the basis of this review, the staff has determined that GE has not provided sufficient information to conclude that the radiation protection measures incorporated in the design will provide a reasonable assurance that occupational doses will be maintained ALARA and below the limits of 10 CFR Part 20 during plant operations.

The following sections provide the bases for the staff conclusions.

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

The staff has audited the information in the SSAR for completeness against the guidelines in RG 1.70 "Standard Format and Content of Safety Analysis Reports For Nuclear Power Plants," and against the criteria set forth in NUREG-0800 "Standard Review Plan," Section 12.1, regarding the radiation protection consideration of the ABWR design. The staff review consisted of ensuring that GE had either committed to following the criteria of the RGs and staff positions referenced in NUREG-0800 (SRP) Section 12.1 or provided acceptable alternatives. In addition, the staff selectively reviewed GE's SSAR against acceptance criteria of the SRP using the review procedures in the SRP. Details of the review follow.

12.1.1 Policy Considerations

Section 12.1.1 of the SSAR describes the policies put into place by GE to ensure that ALARA considerations were factored into each stage of the ABWR design process. GE provides a management commitment to ensure that the ABWR will be designed and constructed in a manner consistent with RG 8.8.

The ALARA philosophy was applied during initial design of the plant. Therefore, the policy considerations are acceptable. The policy considerations regarding plant operations contained in

RG 8.8, RG 1.8 "Qualification and Training of Personnel For Nuclear Power Plants" (Rev. 2) and RG 8.10 "Operating Philosophy For Maintaining Occupational Radiation Exposures As Low As Is Reasonably Achievable" are outside the scope of this review. Applicants seeking an operating license by referencing the ABWR certified design will be required to address these operational policy considerations to ensure radiation doses are ALARA.

12.1.2 Design Considerations

The objective of the ABWR design is to minimize the costs, both in terms of maintenance time and radiation exposure, associated with plant operation. The ABWR design employes features designed to 1) eliminate required maintenance; 2) facilitate plant maintenance and operations; and 3) minimize the sources of radiation exposure in the plant.

Several design features consistent with the guidelines in RG 8.8 are provided for in the ABWR design. The plant layout provides shielded rooms or cubicles for components that are the source of high radiation levels. Components in redundant systems are located in separate shielded rooms or cubicles such that radiation levels associated with an operating train of equipment will not effect maintenance on the redundant train. Shielded rooms or cubicles are provided with labyrinth design access to reduce scattered radiation in areas outside the cubicle. Removable shielded walls or hatches have been provided where space limitations in a roor or cubicle does not allow adequate laydown area for maintenance. The need to enter shielded rooms or cubicles has been minimized with the appropriate use of remote operators and instrumentation. The remote backflushing capability for plant filter/demineralizers employs gravity drains and piping that slope toward the backwash tank to minimize traps that would become radiation hot spots. The use of grafoil valve stem packing to reduce leakage of contaminated water from reactor systems and to minimize the maintenance requirements of these valves is also an ABWR design feature consistent with the guidance in RG 8.8.

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

Additional examples of design features that should reduce radiation exposure include: control rod drive (CRD) mechanism design, plant lay out of lower drywell, and steam relief value (SRV) design and layout. CRD mechanisms in the ABWR have been redesigned to include an internal restraint system. Current BWRs have external restraints on CRDs to prevent a rod ejection in the event of a CRD housing failure, which have to be cleared out of the way during CRD maintenance. The internal CRD restraint feature on ABWR will allow for easier CRD removal and reduced radiation exposure associated with CRD maintenance.

The arrangement of the lower drywell should also contribute to lower radiation exposures during CRD maintenance. The layout of the ABWR lower drywell allows easy access to the lower reactor vessel head for CRD and primary internal pump (PIP) removal. A transport system is also provided to remove CRDs and PIPs from the drywell so that maintenance can be performed in a lower radiation area.

The ABWR design uses direct action SRV's which require less maintenance than current pilot-operated valves. These SRVs are placed circumferentially around the reactor vessel with a dedicated hoist to facilitate maintenance.

Two important areas where current operational BWR experience has not been adequately addressed in the ABWR design, are the dose rates in the upper drywell during the transfer of irradiated (spent) fuel assemblies (SFA), and exposures resulting from a complete withdrawal of the traversing incore probe (TIP).

The anticipated operational occurrence (AOO) of dropping a SFA onto the reactor vessel flange during transfer to the fuel storage pool has the potential for creating extremely high dose rates in the upper drywell. Individuals in the upper drywell during this AOO could receive potentially lethal radiation doses before they could evacuate the area. GE has acknowledged that current BWR designs are inadequate to ensure radiation protection during this AOO, and recommended the use of a shielded bridge arrangement as a fix. GE issued two generic information letters on this subject in 1973 and 1980.

Due to concerns over inadequate implementation of GE's recommended fix by operating BWRs, the NRC augmented its inspection program in 1987 to direct inspectors attention to this area. Backfitting a shielded bridge onto existing BWR was accepted by the staff as a solution to this AOO. However, this solution only reduces the probability of high dose rates during this AOO, it does not completely eliminate the possibility. In response to the staff's question regarding how the ABWR design ensures protection of personnel in the drywell from the intense radiation resulting from a dropped SFA, GE responded that access to the upper drywell would be precluded during SFA transfer by procedural controls. GE's position on this issue is that it is an operational consideration. As noted in Section 12.1.3 of the SSAR, operational considerations are outside the scope of this review. Furthermore, it is an operational decision that is inconsistent with BWR operational experience to date. Therefore, this response is not acceptable to the staff.

By memorandum dated July 29, 1991, GE provided some detailed information on a proposed upgrade to the shielding in the upper drywell. This remains an open item pending further review and evaluation by the staff. The staff concludes that the ABWR design as described in the SSER is inadequate to ensure radiation protection during this event. It is the staff's position that GE must show why the ABWR design cannot be modified, such as providing additional shielding at the top of the drywell, to ensure that radiation doses are ALARA during this AOO.

Several uncontrolled and/or overexposures of operating personnel resulting from the complete withdrawal of the highly activated TIP and drive cable have been experienced at BWRs. The ABWR SSAR does not indicate that any improvement to the TIP system has been made (or evaluated) to minimize the possibility of this AOO for this design. GE's response to the staff's question regarding this system (Q471.26) focused on adequate egress from the room. GE has not addressed preventative measures or measures that would ensure exposures are ALARA during the recovery from this AOO. Based on the above, the staff considers this an open item.

12.1.3 Operational Consideration

Operational considerations regarding the implementation of a radiation protection program are outside the scope of this design certification review. Applicants seeking a operating license by referencing the ABWR certified design must address these operational considerations to the level of detail provided in RG 1.70.

12.2 Radiation Sources

The staff has audited the contained sources and airborne radioactive material source terms provided in Section 12.2 and Chapter 11 of the ABWR SSAR for completeness against the guidelines in RG 1.70, and against the criteria set forth in Section 12.2 of NUREG-0800. The contained source terms are used as the basis for designing radiation protection features (including dose assessment) and for radiation shielding calculations. Airborne radioactive source terms are used in the design of ventilation systems and personnel dose assessment. The staff review consisted of ensuring that GE had either committed to following the criteria of RGs and staff positions contained in Section 12.2 of NUREG-0800 or provided acceptable alternatives. In addition, the staff selectively compared source terms for specific systems against those used for plants of similar design.

The results of this review indicate that GE has not provided sufficient information for the staff to conclude that ABWR meets the requirements of 10 CFR Part 20 and General Design Criterion (GDC) 61 of 10 CFR Part 50. Details of the review follow.

12.2.1 Contained Sources and Airborne Radioactive Ma' ~ial Sources

GE's description of radioactive sources in the ABWR are provided in Chapters 11 and 12 of the SSAR. Section 11.1 provides information on the radioactive source terms in reactor water and steam. Section 12.2 provides descriptions of plant components that become significant sources of radiation during plant operations, including shutdown. Sources of airborne radioactive material are discussed in Section 12.2.2 of the SSAR.
GE's description of radioactive sources within the ABWR design in the SSAR is not acceptable to the staff for the following reasons:

1) The SSAR provided insufficient characterization of the source (i.e., source location, geometry, etc.) for the description to be useful for input to shielding calculations. Less than one half of the sources identified in Section 12.2 could be located on plant layouts provided.

2) The description of sources provided in Section 12.2 is incomplete. Obvious omissions include the TIP system (both the cable and detector) following TIP withdrawal, and turbine building sources such as reheaters, moisture separators and condensate filter/demineralizers.

3) Section 12.2.2 of the SSAR contains an inadequate description of in-plant airborne radioactive sources. As provided in RG 1.70 this description should include a tabulation of the calculated concentrations of airborne radioactive material, by nuclides, expected during normal operation and AOOs, for equipment cubicles, corridors, and operating areas normally occupied by operating personnel. No such description has been provided. Therefore, the staff was unable to determine if the ABWR design could be operated within the limits of 10 CFR 20.103.

4) There is no discussion of radiation sources in the ABWR design during accident conditions. Although Section 1AA.2 in SSAR Chapter 1 (Appendix A) discusses the plant areas requiring access during an accident, no description of sources associated with these areas and tasks required under accident conditions is provided.

5) Sources inside the drywell, such as the reactor internal pumps and their respective heat exchangers, that become a radiation protection concern during shutdown/maintenance periods, also have not been included.

On July 29th GE provided fuel bundle source term information which the staff is currently evaluating. GE still needs to revise the SSAR to provide the information listed above to fully resolve this open issue related to source descriptions.

12.3 Radiation Protection Design

The staff has audited the facility design features, shielding, ventilation, and radiation and airborne monitoring instrumentation contained in the ABWR SSAR for completeness against the guidelines in RG 1.70 and against the criteria set forth in NUREG-0800, Section 12.3. The staff review consisted of ensuring that GE had either committed to following the criteria of RGs and staff positions referenced in Section 12.3 of NUREG-0800, or provided acceptable alternatives. In addition, the staff selectively reviewed GE's SSAR against the acceptance criteria of the SRP using the review procedures in NUREG-0800. This review found several deficiencies in the ABWR SSAR. Details of the review follow.

12.3.1 Facility Design Features

GE has provided some evidence that radiation dose accumulating tasks (maintenance, refueling, radioactive material handling, in-service inspection, decommissioning, and accident recovery) have been considered in the plant design. Several features, as discussed above (see Section 12.1.2), have been included in the design to help maintain doses ALARA. These features will facilitate access to work areas, reduce or allow the reduction of source intensity, reduce the occupancy requirements in high radiation fields, and provide for portable shielding and remote-operation and instrumentation of radioactive systems. These ABWR features are consistent with the guidance of RG 8.8 (Rev. 3) and NUREG-0800 and are acceptable to the staff.

GE has provided drawings of the plant layout which indicate radiation zones used in the plant design. The six radiation zones provide a basis for classifying occupancy and access restrictions for various areas within the plant during normal operations and accident conditions. On this basis, maximum design dose rates are established for each zone and used as input for shielding of the respective zones. This method of plant zoning is consistent with the guidance in RG 1.70 and NUREG-0800 and is generally acceptable to the staff. However, since GE has not provided sufficient information regarding shielding design (see Section 12.3.2), the staff is unable to verify the zone designations provided in the SSAR.

In addition, the staff has identified the following deficiencies in the plant layout drawings:

1) Plant layouts do not clearly identify all sources of radiation identified in Section 12.2 of the SSAR.

2) Figures 12.3-37 through 12.3-48 (showing the radwaste and control buildings) are illegible and do not clearly identify the radioactive systems and processes for each room and cubicle indicated.

3) Figures 12.3-49 through 12.3-53 depicting the turbine building during normal operation are inconsistent with Figures 12.3-69 through 12.3-73 provided to indicate the locations of radiation monitoring equipment in the turbine building. These two sets of figures appear to be mirror images of each other (respective to the indicated plant north). In addition, several design features of the turbine building, such as the function and use of the areas adjacent to the offgas charcoal beds, are inconsistent between these two sets of figures. Thus, the staff was not able to determine the radiation hazard to personnel in the turbine building, nor whether radiation monitors were properly located.

Several features are included in the ABWR design to minimize the buildup of activated corrosion and wear products, a major contribution to occupational doses. These features include a reduction in cobalt bearing components used in reactor systems (activated cobalt is a major contributor to plant radiation levels) and pre-filming of reactor systems prior to plant operation, to minimize activated material deposition on system interior surfaces. Main condenser tubes and tube-sheets will be made of titanium alloys to minimize the introduction of foreign material into the reactor system (which become activated and/or promote corrosion) resulting from condenser tube leakage. Other features such as the use of seamless piping, the use of straight through valve design wherever possible, the use of butt-welded piping connections, and the use of backflushing connections on instrument lines, minimize build-up of radioactivity in plant piping systems.

The SSAR indicates that for "highly radioactive systems" connections will be provided to facilitate chemical decontamination of heat exchangers. The staff requested that GE identify these "highly radioactive systems." However, to date GE has not responded to this request. This remains an open item.

GE's corrosion product control features are consistent with the guidance in RG 8.8 (Rev. 3) and NUREG-0800 and, with the exception of the lack of specifics regarding the capability to decontaminate heat exchangers, are acceptable to the staff. The need to provide this decontamination information is considered an open item.

The ABWR is designed such that operation will not require an application for alternate high radiation area controls (per 10 CFR 20.203(c)(5)), as experienced with current operating BWRs.

The design provides that all high radiation areas (greater than 100 mR/hr) are maintained locked to control unauthorized access and no credit is taken for the relief provided in Section 12.6 of the BWR Standard Technical Specifications (i.e., locked area at 1000 mR/hr). This design position is acceptable to the staff.

12.3.2 Shielding

The objective of the plant's radiation shielding is to provide protection against radiation exposure for operating personnel, both inside and outside the plant, during normal operation, including AOOs, and during reactor accidents. The ABWR shielding design is based primarily on existing practice with concrete wall thickness of 4 feet chosen for large radiation sources decreasing to 1 feet for low radiation sources. Concrete used for radiation shielding meets the NRC guidance provided in RG 1.69. GE's validation of the ABWR shielding design was performed primarily with the QAD-F computer code. Shielding calculations were also performed by GE with GGG and DOT.4. These are commonly acceptable shielding calculational codes and are therefore acceptable to the staff.

The physical dimensions (thickness, etc.) of specific radiation shields have not been provided by GE, in accordance with the guidance of RG 1.70 and the acceptance criteria of NUREG-0800. This plus the lack of specifics of radiation sources in the plant, as discussed above, does not allow the staff to conduct confirmatory calculations of shielding effectiveness. Therefore, the staff cannot conclude that the ABWR design can meet the radiation dose requirements of 10 CFR Part 20 or Item II.B.2 of NUREG-0737. This item remains open.

Two specific areas of concern identified by the staffs review are the adequacy of drywell and reactor vessel shielding during fuel handling operations and the adequacy of the shielding surrounding the personnel access hatch to the lower drywell during TIP removal from the core. The reactor vessel shield depicted in Figures 12.3-23 and 24 does not cover a significant portion of the top of the reactor vessel. As noted in Section 12.1, a fuel handling mishap resulting in dropping a spent fuel bundle across the reactor flange is a significant radiological hazard in BWRs. In addition to the radiological hazard presented by this AOO, it appears that raising an irradiated fuel bundle in proximity of the vessel wall could result in significant radiation dose rates in the upper drywell. On July 29, 1991, GE provided upper drywell design information relative to shielding dimensions which is currently being evaluated by the staff. It is the staff's position at this time that GE has not adequately addressed this concern, and it remains an open item as discussed in Section 12.1.2 above.

In addition, the plant drawings provided in the SSAR appear to indicate that high dose rates will exist in the lower drywell access air lock during TIP withdrawal from the core. This remains an open item as discussed in Section 12.1.2 above.

12.3.3 Ventilation

The ABWR ventilation systems are designed to protect personnel and equipment from extreme environmental conditions and ensure that plant personnel are not inadvertently exposed to airborne contaminants exceeding the concentration limits given in 10 CFR 20.103. Design features intended to maintain personnel exposures ALARA include:

1) Airflow between areas potentially having airborne contamination is always from the area of lower potential contamination to the area of higher potential contamination;

2) The appropriate use of negative or positive pressure in areas to prevent exfiltration or infiltration of possible airborne contamination respectively;

3) A dual fresh air intake system for the control room ventilation designed such that at least one intake is free of contamination following a LOCA accident.

These design criteria are in accordance with the guidelines of RG 8.8 (Rev. 3) and are acceptable to the staff. However, as noted in Section 12.2 the concentrations of airborne contamination in cubicles, rooms, and corridors has not been provided by GE. Therefore, the staff cannot conclude that the ABWR ventilation system design meets the acceptance criteria of NUREG-0800, and is adequate to maintain personnel exposures within the limits of 10 CFR Part 20. This remains an open item.

12.3.4 Area Radiation and Airborne Radioactive Monitoring Instrumentation

In Chapter 12 of the SSAR GE has provided tables and figures identifying the location of area radiation monitors. However, contrary to the guidelines in RG 1.70, GE has not provided information on these monitors regarding auxiliary and/or emergency power supplies, detector range, sensitivity, accuracy or precision, alarm capability or alarm setpoints, read out locations, details of airborne sampling lines and pump locations.

A description of the radiation instrumentation that will meet the criticality accident monitoring requirements of 10 CFR 70.24 for storage of new fuel has not been provided. Also GE has not addressed whether the guidance provided by RGs 1.21, 8.2, 8.8, 8.12 and ANSI N 13.1-1969, as they apply to area and airborne radiation monitoring, or alternative methods were applied. No information is provided in Chapter 12 regarding airborne radiation monitoring within the plant. This remains an open item.

To meet the criteria of TMI Action Plan Item II F.1.3, the ABWR design provides two high-range gamma monitors that measure up to 10^7 R/hr consistent with the criteria of Table II.F.1-3 of NUREG-0737. However, GE has not indicated the location of these high-range monitors on plant layout drawings. Therefore, the staff cannot conclude that the ABWR design meets the acceptance criteria of NUREG-0800, and this remains an open item.

12.4 Dose Assessment

The staff has audited GE's dose assessment for the ABWR design for completeness against the guidelines in RG 1.70, and against the criteria set forth in NUREG-0800, Section 12.3. This review consisted of ensuring that GE had either committed to following the criteria of RGs and staff positions in Section 12.3 at NUREG-0800, or provided acceptable alternatives. In addition, the staff selectively compared the dose assessment made by GE for specific functions against the experience of operating BWRs. Details of the review follow.

GE has provided an assessment of the radiation dose that would be received by operating a plant of the ABWR design in SSAR Section 12.4. Estimated person-rem doses are provided for major work within areas of the plant during maintenance/refueling periods and for power operations, based on a breakdown of estimated person hours per year and the average dose rate for each task. This assessment listed in Table 12.4-1, results in an estimated total annual dose of 95 person-rem.

The dose assessment presented in Section 12.4 is not acceptable to the staff. The staff review identified the following deficiencies:

1) The level of detail presented in this dose assessment is not consistent with the guideline of RG 8.19 or the acceptance criteria of NUREG-0800.

2) Numbers and descriptions in the text of Chapter 12 are difficult to correlate with numbers presented in Table 12.4-1. For example, the average dose rates for main steam isolation valve and SRV work are given as 4 mrem/hr and 5.5 mrem/hr in Section 12.4.1 of the text but listed as 13.5 mrem/hr and 20 mrem/hr in Table 12.4-1, respectively.

3) Summing the product of the hours/task and mrem/hr/task presented in Table 12.4-1 results in an annual dose of over 300 person-rem, not 95 person-rem.

4) No basis (hour-dose rate breakdown) is given for doses estimated for tasks in the turbine building or radwaste building.

5) Annual person-rem estimates in Table 12.4-1 for task performed in the turbine building are inconsistent with estimates provided as a response to Question 471.18 in Chapter 20 of the SSAR. No basis for the lower numbers in Chapter 12 is given.

6) No basis is provided for the estimate of 16 person-rem/year estimated in Table 12.4-1 for work at power. Experience with operating BWRs indicate the 'best performers' incur an average 150 to 200 person-mrem/day of non-outage operation. This implies an annual dose of at least 45 to 60 person-rem for work performed at power.

Based on this review the staff cannot conclude that the dose assessment provided by GE meets the acceptance criteria of RG 0800. This remains an open item.

12.5 Organization

The Organization required to implement an effective health physics program and assure that radiation exposures are within the limits of 10 CFR Part 20, and are ALARA, is outside the scope of this review. Applicants seeking an operating license by referencing the ABWR certified design will be required to address this concern to the level of detail discussed in RG 1.70.

13 CONDUCT OF OPERATIONS

13.3 Emergency Planning

In Section 13.3 of the SSAR, GE has indicated that emergency planning is not within the scope of the ABWR design. The staff agrees that this subject will be addressed by the utility applicant referencing the ABWR design and will significantly depend upon plant and site specific characteristics.

GE also provided in Table 13.3-1 a listing of design considerations pertaining to emergency planning. The staff has reviewed the information included in the table and has the following recommendations to add NRC requirements or guidance documents to the table for clarification.

NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants" (November 1980), and NUREG-0737, Supplement 1, "Clarification of the TMI Action Plan Requirements-Requirements for Emergency Response Capability" (January 1983) should be added as references to the ABWR Design Consideration column for the Technical Support Center, Operational Support Center, and the Emergency Operations Facility.

14 TEST PROGRAMS

14.2 Initial Plant Test Program

The staff has reviewed General Electric's (GE's) Chapter 14.2 submittal titled "Specific Information to be Included in Final Safety Analysis Reports" for the GE ABWR Standard Plant through Amendment 17, in accordance with Section 14.2 of NUREG-0800, "Standard Review Plan" (SRP) and in accordance with RG (RG) 1.68 Revision 2, "Initial Test Programs for Water-Cooled Nuclear Power Plants." The staff has concluded that, although the described program is generally comprehensive and covers most areas of the staff's concerns, the Initial Test Program described does not provide sufficient detail with respect to all systems and test descriptions to enable the staff to determine the adequacy of GE's commitments and tests described.

A request for additional information (RAI) consisting of a list of questions, comments, and errata information was forwarded to GE for its use in revising its standard safety analysis report (SSAR). A meeting was held with GE in order to discuss potential responses to these items on May 7, 1991 and subsequently a draft amendment was submitted via letter dated May 20, 1991 from R. C. Mitchell (GE) to C. L. Miller (NRC) in response to these items.

14.2.1 Evaluation

The following evaluation presents the staff's position based upon a review of the GE ABWR initial test program (ITP). This evaluation includes information included in the RAI with respect to the GE ABWR ITP prepared by the staff, GE's response to the RAI as contained in the May 20, 1991 letter, and the staff's findings regarding each response. The evaluation in this report addresses only those sections of the SSAR which the staff initially found to need modification or additional information. Sections of the SSAR not addressed were found to be acceptable and a discussion of each will be included in the final safety evaluation for the ABWR. Upon modification of individual test abstracts in response to the staff's findings, an evaluation of the test abstract coverage of system specific test requirements will be made and also reflected in the final safety evaluation report.

14.2.4 Conduct of Test Program

See next Section 14.2.5.

14.2.5 Review, Evaluation, and Approval of Test Results

Interface Requirement: In the review of the SSAR, the staff determined that Sections 14.2.4 and 14.2.5 should be modified to specify whose approval must be obtained before increasing power to the next higher test plateau. GE indicated in its response that such specifics will be a function of the plant owner/operator's unique organizational structure and detailed plant administrative procedures and are thus left to the applicant referencing the ABWR design. The staff finds this interface requirement acceptable. Modification of SSAR Section 14.2.13 should reflect this interface requirement.

14.2.7 Conformance of Test Program With RGs,

Based on its review of Section 14.2.7 of the SSAR, the staff determined that GE needed to add additional references to RGs. In the letter of May 20, 1991, GE agreed to amend the ABWR SSAR to include the following items:

a. Include RG 1.95, "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release," in accordance with SRP Section 14.2.

b. Include RG 1.139, "Guidance for Residual Heat Removal," in accordance with SRP Section 14.2.

c. Document the applicable revision number of each RG listed in Section 14.2.7 or reference Table 1.8-20 of the SSAR. The SSAR will be amended so that Section 14.2.7 will reference Table 1.8-20 for the applicable revision numbers of the listed RGs.

d. Correct the reference to RG 1.68.3, "Preoperational Testing of Instrument and Control Air Systems," contained in Table 1.8-20 of the SSAR or Section 14.2.7, as appropriate, to Revision 0, issue date of April 1982.

The staff finds the above acceptable.

14.2.10 Initial Fuel Loading and Initial Criticality

The staff determined based on its review of Section 14.2.10, that this section should be modified to state that completion of preoperational testing (including the review and approval of the test results) is required prior to fuel loading. If portions of any preoperational tests are intended to be conducted, or their results approved, after fuel loading GE should: (1) list each test; (2) state which portions of each test will be delayed until after fuel loading; (3) provide technical justification for delaying these portions; and (4) state when each test will be completed (key to test conditions defined in Chapter 14).

GE has stated its intent that all preoperational tests shall be completed, and the results obtained approved prior to commencement of fuel loading. However, there may be unforeseen circumstances that arise that would prevent this from occurring but that would not necessarily justify the delay of fuel loading. GE indicated that Section 14.2.10 of the SSAR will be revised accordingly to require that the above stated conditions be appropriately documented should the applicant referencing the ABWR design decide to request permission from the NRC to proceed with fuel loading under such circumstances. The staff finds that this is acceptable.

14.2.11 Test Program Schedule

Based on the staff's review of Section 14.2.11, the staff has determined that this section of the SSAR should be modified to include the following:

a. A figure which illustrates the power-flow operating map.

GE has indicated that SSAR Figure 4.4-1 illustrates the power flow operating map, however, the staff believes that this figure does not provide sufficient detail regarding test condition identification to determine that each startup test is conducted at appropriate power-flow conditions in accordance with RG 1.68, Appendix A.5. GE should provide or reference an appropriate power-flow operating map in Chapter 14. This is an open item.

b. A table which lists the startup tests and states at which test condition(s) each test is to be conducted.

Presently, the SSAR does not contain the table identified in b. above. The staff will use this table, together with the power-flow operating map requested above to determine that each of the startup tests are conducted at appropriate power-flow conditions in accordance with RG 1.68, Appendix A.5. This is an open item.

14.2.12 Individual Test Descriptions

The staff has reviewed Section 14.2.12 and has determined that Section 14.2.12.1 states that testing of systems outside the scope of the ABWR Standard Plant are discussed in Subsection 14.2.12.3. The information relative to testing of systems outside the scope of the ABWR can be found in SSAR Section 14.2.3 rather that in Section 14.2.12.3. The staff finds this information acceptable but the cross reference needs to be corrected.

The staff also determined that Section 14.2.12 test abstracts should be modified to address the following concerns:

a. Several preoperational and startup test prerequisites include the requirement that interfacing support systems shall be available.

Interface Requirement: The staff has asked GE to identify which support systems are required for each test and specify which individuals or groups are authorized to make this determination. GE has stated that the interfacing support system requirements will be specified in the detailed test procedures (and operating and maintenance procedures, if appropriate) which are required by RG 1.68 to be made available to NRC personnel at least 60 days prior to their intended use. Additionally, the startup manual and applicable plant administrative procedures shall delineate how such determinations of operability and availability will be authorized. Thus, these details are the responsibility of the applicant referencing the ABWR design.

The staff believes that the level of detail in the test abstracts is insufficient to determine conformance with RG 1.68, Position C.2. GE should 1) modify Section 14.2.12.1 to address generic interfacing support system availability and 2) modify individual test abstracts to address specific interfacing support system availability. This is an open item.

b. The use of the word "should" in most, if not all test abstracts, is not a commitment by GE to perform certain tasks. It should, therefore, be reevaluated and revised accordingly (i.e., "will," "must").

GE has agreed to reevaluate the commitments in the test abstracts. The staff will determine the acceptability of the commitments based upon the GE reevaluation in the final SER. This is an open item.

c. Several preoperational and startup test abstracts include imprecise acceptance criteria (e.g., applicable intervals, applicable design specifications, specified amounts, specified tolerances, perform as specified, function properly).

GE should modify the individual test abstracts to specify the bases for determining acceptable system and component performance. Acceptable criteria includes specific references to RGs, Technical Specifications, assumptions used in the safety analysis, other ABWR SSAR sections, and applicable codes and standards. GE has indicated that Chapter 14 of the SSAR was written primarily to document the appropriate testing commitments contained in RG 1.68. It was anticipated that precise acceptance criteria would be provided as part of the ITAAC effort. This is an open item.

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

The distinction between these types of acceptance criteria is unclear. GE should modify Section 14.2.12.2 and individual startup test abstracts to differentiate between various types of acceptance criteria and the resultant actions for each type if unsatisfactory test results are obtained. GE has stated in Section 14.2.12.2, "Specific actions for dealing with criteria failures and other testing exceptions or anomalies will be described in the startup administrative manual." (To be supplied by the applicant referencing the ABWR design)

The staff believes that this response is not acceptable. GE should modify Section 14.2.12.? and the individual test abstracts to address the subject acce_tance criteria on a test specific basis. This is an open item.

Startup tests listed in Section 14.2.12.2 that are not essential to the demonstration of conformance with design requirements for structures, systems, components, and design features which meet any of the following criteria should be identified:

a. Those that will be used for safe shutdown and cooldown of the reactor under normal plant conditions and for maintaining the reactor in a safe condition for an extended shutdown period; or

b. Those that will be used for safe shutdown and cooldown of the reactor under transient (infrequent or moderately frequent events) conditions and postulated accident condi[illegible] and for maintaining the reactor in a safe conditic [illegible] an extended shutdown period following such conditions or

c. Those that will be used for establishing conformance with safety limits or limiting conditions for operation that will be included in the facility technical specifications; or

d. Those that are classified as engineered safety features or will be used to support or ensure the operations of engineered safety features within design limits; or

e. Those that are assumed to function or for which credit is taken in the accident analysis for the facility, as described in the SSAR; or

f. Those that will be used to process, store, control, or limit the release of radioactive materials.

GE has stated that the tests abstracts contained in Section 14.2.12.2 of the ABWR SSAR are intended to meet the requirements of RG 1.68, updated and/or modified as necessary to reflect the actual ABWR design. A screening will be performed to identify and document any testing that is currently specified for systems that are not essential for demonstrating conformance with the aforementioned criteria.

The staff will determine the acceptability of this response upon completion of individual test abstract screening. Section 14.2.12.2 should be modified to document the results of this screening. This is an open item.

14.2.12.3 Conformance of the ABWR with RG 1.68 Revision 2

The staff's review of the preoperational and startup test phase descriptions disclosed that the operability of several of the systems and components listed in RG 1.68 may not be adequately demonstrated by the tests described in the SSAR.

GE should either expand the test descriptions to address the following items, insert cross-references in Section 14.2.12 if complete test descriptions for the following items are provided elsewhere in the ABWR SSAR, or modify Section 14.2.7 or

Table 1.8-20 of the SSAR, as appropriate, to provide technical justification for any exception to RG 1.68, Rev. 2. Thus the following items should be reflected in a subsequent amendment to the SSAR. (Note: each item is numbered in accordance with RG 1.68 Revision 2).

Preoperational Testing

1.a.(2)(d) Supports and restraints for discharge piping of SRVs.

GE has revised SSAR Section 14.2.12.1.1 indicating that testing of SRV discharge piping supports and restraints is specifically covered by that testing described in SSAR Section 14.2.12.1.51. SSAR Section 14.2.12.1.51 will be modified to specifically cross reference the applicable testing requirements given in SSAR Sections 3.9.2.1 and 5.4.14.4. The staff will determine the acceptability of this response upon completion of the evaluation of the individual test abstracts. This is an open item.

1.a.(4) Pressure boundary integrity tests.

GE stated that the integrity tests of the reactor coolant pressure boundary are specified in Section 5.2.4.6.2. Section 14.2.12.1.1 has been revised accordingly to cross-reference the applicable testing requirements. The acceptability of this response will be determined upon evaluation of the individual test abstracts. This is an open item.

1.c Protection of facility for anticipated transients without a scram (ATWS).

The ATWS protection functions are tested as part of the respective systems which perform such functions (i.e., standby liquid control system, rod control and information system (RCIS), fine motion cotrol rod system, recirculation flow control system). However, for the purpose of more explicitly demonstrating compliance with RG 1.68, the appropriate subsections of Section 14.2.12.1 will be revised to more specifically indicate where ATWS related testing requirements are being fulfilled, particularly those related to the alternate rod insertion (ARI) function.

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

The staff will determine the acceptability of this response upon evaluation of the individual test abstracts. This is an open issue.

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

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

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

The ABWR design does not utilize containment recirculation fans during normal operation or accident conditions. Therefore, the specific test described in RG 1.68 is not applicable. The staff finds this acceptable.

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

The ABWR containment structural integrity testing requirements are specified in Section 3.8.1.7.1. Accordingly, Section 14.2.12.1.40.2 has been added to cross-reference the applicable testing requirements. The staff will determine the acceptability of this response upon evaluation of the individual test abstracts. This is an open issue.

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

In the ABWR design the failed fuel detection function is performed by the leak detection and isolation system and the process radiation monitoring system. In particular, gross fuel failure would be detected first by the main steam line radiation monitors and secondarily by the offgas pre-treatment radiation monitors. In addition, the normal reactor water sampling system will allow for identification of trends indicative of possible fuel failure. Testing of the applicable features of the associated systems, as described in Subsections 14.2.12.1.13 and 14.2.12.1.23, will assure proper operation of the failed fuel detection function. The staff will determine the acceptability of this upon evaluation of the individual test abstracts. This is an open issue.

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

Automatic load following is performed by the Automatic Power Regulator whose testing is described in Section 14.2.12.1.17. This system will have the capability, if enabled, to accept external demand signals (e.g., from the load dispatcher). Should the applicant referencing the ABWR design decide to seek approval for utilization of this capability, designation of the appropriate testing will have to be included in the application for such. Section 14.2.13 will be revised accordingly to document this potential interface requirement. The staff will determine the acceptability of this interface requirement upon evaluation of the individual test abstracts. This is an open item.

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

Interface requirement: Traditional "Preoperational testing" of personnel monitors and radiation survey instruments is not appropriate as these instruments are subject to very specific calibration programs. It is the responsibility of the plant operator to verify and maintain the proper calibration and operation of such devices. Therefore, any required testing shall be the responsibility of the applicant referencing the ABWR design. Section 14.2.13 will be revised accordingly to document this as an interface requirement. The staff finds this acceptable.

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

The test description in Section 14.2.12.1.34 has been revised to indicate that the control room habitability function is to be included in the testing specified for the dedicated HVAC system of the main control room. Additionally, a specific requirement to demonstrate the system capability to detect smoke and/or toxic chemicals and to remove and/or prevent in-leakage of such has been added. The staff will determine the acceptability of this upon ev ation of the individual test abstracts. This is an open issue.

Initial Fuel Loading and Precritical Tests

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

Such testing will have been completed as part of the preoperational testing described in Subsection 14.2.12.1.14. Additionally, these tests are part of the plant Technical Specification surveillance program which is required to be instituted prior to commencement of fuel loading as specified in Section 14.2.10.1. However, Subsection 14.2.12.2.3 has been revised to specifically require that the demonstrations required by Position 2.c above be completed as prerequisites to fuel loading. The staff finds this acceptable.

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

Such testing will have been completed as part of the preoperational testing described in revised Subsection 14.2.12.1.1, which references the required reactor coolant leak rate tests specified in Subsection 5.2.4.6.1. However, Subsection 14.2.12.2.3 has been revised to specifically require that the demonstrations required by Position 2.d above be completed as prerequisites to fuel loading. The staff finds this acceptable.

<u>Low Power Testing</u>

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

The staff's review revealed that GE had not included this test in the SSAR. GE has indicated that Section 14.2.12.2 will be revised accordingly to address this test. The acceptability of this response will be determined when Section 14.2.12.2 is revised accordingly. This is an open item.

4.l Branch steamline valves and bypass valves used for protective isolation functions at rated temperature and pressure conditions.

For the ABWR design the only branch steamline valves used for protective isolation functions are those on the RCIC steamline and the common drainline from the main steamlines. Accordingly, the description of RCIC system testing in Subsection 14.2.12.2.22

has been revised to include specific testing of the RCIC steamline isolation valves and Subsection 14.2.12.2.26 has been revised to include specific testing of the main steamline branch drain line isolation valves in addition to the MSIV testing already specified. The staff will determine the acceptability of this upon evaluation of the individual test abstracts. This is an open item.

Power Ascension Tests

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

The ABWR design has no partial scram function. Rod-runback is accomplished by the select control rod run-in (SCRRI) function. Subsection 14.2.12.2.6 has been revised to assure that appropriate testing is performed to demonstrate proper functioning of SCRRI logic and hardware. Also, Subsection 14.2.12.2.30 has been revised to assure that proper plant response is demonstrated during the event that will result in initiation of SCRRI. The staff will determine the acceptability of this upon evaluation of the individual test abstracts.

5.n Reactor coolant system loose parts monitoring system.

An appropriate test description will be added to Section 14.2.12.2. The acceptability of the test description will be determined when Section 14.2.12.2 is revised. This is an open item.

5.o Reactor coolant leak detection systems.

It is expected that testing of reactor coolant leak detection systems will be completed during the preoperational stage. The staff finds this acceptable.

5.q Proper operation of failed fuel detection systems.

In the ABWR design the failed fuel detection function is performed by the process radiation monitoring system, the testing of which is described in Subsection 14.2.12.2.1. This test description has been revised to require the appropriate demonstration of the related failed fuel detection function. Also see response to item 1.j.(12) above. This staff finds this acceptable.

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

The staff's review determined that the test descriptions required additional information. The applicable test descriptions have been revised accordingly. See response to item 4.1 above. This is an open item.

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

An appropriate test description will be added to Section 14.2.12.2. The acceptability of this response will be determined when Section 14.2.12.2 is revised to provide the test descriptions. This is an open item.

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

The auxiliary systems required to support operation of engineered safety features include the cooling water and HVAC systems whose testing is described in Subsections 14.2.12.2.23 and 14.2.12.2.24, respectively. These subsections have been revised to assure that the testing performed, and results obtained, will ultimately demonstrate the adequacy of a particular auxiliary system's performance under limiting accident conditions. The acceptability of this response will be determined upon evaluation of the individual test abstracts. This is an open item.

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

This testing is part of that described in Section 14.2.12.2.1(3), which has been revised to specifically address Position 5.z above. The staff finds this acceptable.

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

An appropriate test description will be added to Section 14.2.12.2. The acceptability of this response will be determined when Section 14.2.12.2 has been revised to include the appropriate test description. This is an open item.

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

ATWS design features are comprised primarily of dedicated logic, and some hardware, which will be thoroughly checked out as part of the preoperational test program (see response to Item 1(c) of 2.1.8.1 above). Most hardware design features that perform ATWS related functions do so in their normal mode, only initiated by dedicated ATWS logic. Therefore, the functioning of these features is adequately verified via the testing already conducted for such. Thus, no dedicated testing of ATWS related features is planned during the power ascension test phase. The acceptability of this response will be determined upon evaluation of the individual test abstracts in conjunction with Item 1(c) of 14.2.1.8.1. This is an open issue.

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

This testing is intended to be a part of that described in Section 14.2.12.2.16, which has been revised to specifically address Item 5.h.h above. The staff finds this acceptable.

14.2.12.3 TMI Items

Section 1A.2.4 of the SSAR states that testing described in Chapter 14 is consistent with the BWR Owner's Group response to Item I.G.1 of NUREG-0737 as documented in a letter dated February 4, 1981 from D. B. Waters to D. G. Eisenhut. Section 14.2.12 test abstracts that describe testing outlined in Appendix E of this letter should be identified or modified accordingly.

Testing outlined in Appendix E of the referenced document is specified in the following test abstracts: 14.2.12.1.1(3)(a), 14.2.12.1.9(3)(j) and 14.2.12.1.44(3)(a). A more detailed review and comparison will be performed of the requirements of Item I.G.1 of NUREG-0737 versus the response given in Section 1A 2.4 of the SSAR and the correspondence referenced therein, and the SSAR test abstracts listed above. The acceptability of this response will be determined when the SSAR is revised accordingly. This is an open item.

14.2.12.4 Conformance with Other RGs

Section 14.2.12.1.19, "Reactor Water Cleanup System Preoperational Test," Section 14.2.12.1.54, "Condensate Cleanup System Preoperational Test," and Section 14.2.12.2.21, "Reactor Water Cleanup System Performance," should be modified to address the concerns of RG 1.56, "Maintenance of Water Purity in Boiling Water Reactors."

RG 1.56 deals mainly with design related issues, specifically the equipment and instrumentation needed to assure proper PWR reactor water chemistry. Subsections 14.2.12.1.19, 14.2.12.1.54 and 14.2.12.2.21 describe preoperational and power ascension testing that is adequate to demonstrate proper performance of the reactor water clean-up system and the condensate filter/demineralizer system in assuring that acceptable reactor water chemistry is maintained. Subsection 14.2.12.1.22 describes the preoperational testing intended to demonstrate the proper functioning of the instrumentation required by RG 1.56. However, this section has been revised to more specifically address functioning of conductivity meters, which are a major focus of RG 1.56.

Likewise, Subsection 14.2.12.2.1 verifies that a proper reactor water chemistry monitoring program is in place. However, this subsection has also been revised to more specifically address the required demonstration of the proper functioning of related instrumentation (i.e., conductivity meters). The acceptability of this response will be determined upon evaluation of the individual test abstracts. This is an open item.

Section 14.2.12.2.14, "Feedwater Control," should be revised to address the following items in accordance with RG 1.68.1, "Preoperational and Initial Startup Testing of Feedwater and Condensate Systems for Boiling Water Reactor Power Plants:"

a. Modify the test description to provide for demonstration of the operability of the feedwater system at low reactor power (less than or equal to 15 percent reactor power) (R.G.1.68.1.C.2.a).

Such testing is already specified in the current description. A more specific commitment to the RG position will be evident in the test matrix to be supplied in response to comment 2.1.4.b. The staff will determine the acceptability of this issue when the test matrix is submitted. This is an open issue.

b. Modify or clarify the test acceptance criteria to provide assurance that vibration levels for system components and piping are within predetermined limits (R.G.1.68.1.C.2.f); piping movement during heatup and steady state and transient operation are within predetermined limits (R.G.1.68.1.C.2.g); and adequate margins exist between system variables and setpoints of instruments monitoring these variables to prevent spurious actuations or loss of system pumps and motor-operated valves (R.G.1.68.1.C.2.h).

The testing called for by Positions C.2.f and C.2.g is included in the test abstracts of subsections 14.2.12.1.51, 14.2.12.1.53(b) and (k), 14.2.12.2.10, 14.2.12.2.11, and 14.2.12.2.18. Subsection 14.2.12.2.18 has been revised to more specifically address Position C.2.h. The acceptability of this response will be determined upon evaluation of the individual test abstracts. This is an oper item.

Section 14.2.12.1.27, "Instrument Air and Station Service Air System Preoperational Test," should be revised to address the following items in accordance with RG 1.68.3, "Preoperational Testing of Instrument and Control Air Systems:"

a. Determination that the total air demand at normal steady state conditions, including leakage from the system, is in accordance with design (R.G.1.68.3.C.5).

This determination is part of the testing specified in 14.2.12.1.27(3)(f) which has been revised to more directly address the issue of total demand, including leakage. The acceptability of this response will be determined upon evaluation of the individual test abstracts. This is an open item.

b. Demonstration that the plant equipment designated by design to be supplied by the instrument and control air system is not being supplied by other compressed air supplies (such as service air (SA)) that may have less restrictive air quality requirements (R.G.1.68.3.C.9).

Although the SA air system acts as a back-up to instrument air, it does so upstream of the instrument air filters. Furthermore, although totally separate (except for the manual back-up cross-tie) the design of the two systems is essentially identical. Thus, the air supplied to the inlet of the instrument air filters is of the same quality, whether it is sourced from the instrument or SA system; therefore, the outlet air will be of the same quality. Since the design precludes occurrence of the conditions hypothesized, no specific test demonstration is needed beyond the construction verification and preoperational testing already planned. The staff finds this acceptable.

c. Demonstration that functional testing of instrument and control air systems important to safety is performed to ensure that credible failures resulting in an increase in the supply system pressure will not cause loss of operability (R.G.1.68.3.C.11).

The test description will be revised accordingly. The acceptability of this response will be determined when the SSAR is revised. This is an open item.

Section 14.2.12.1.34, "Heating, Ventilation, and Air Conditioning Systems Preoperational Test," should be revised to address the concerns of RG 1.95, "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release."

See response to comment on Position 1.n.(14)(f). The acceptability of this response will be determined upon evaluation of the individual test abstracts. This is an open item.

Section 14.2.12.1.34 or other appropriate preoperational tests, should be revised to address the concerns of Position C.5 of RG 1.140, "Design, Maintenance, and Testing Criteria for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants."

The testing requirements specified by RG 1.140, and by the industry standards referenced therein, will be reviewed for their potential applicability to potentially effected filtration and adsorption units. Section 14.2.12.1.34 will then be revised accordingly. The acceptability of this response will be determined when the SSAR is revised accordingly. This is an open item.

Section 14.2.12.1.8, "Residual Heat Removal System Preoperational Test," should be revised to address the following items in accordance with RG 1.139, "Guidance for Residual Heat Removal:"

a. RHR system isolation (RG 1.139.C.2).

The applicable demonstrations were intended to be a part of the testing described in Subsection 14.2.12.1.8(3)(i). However, the testing description has been revised to specifically address testing of features designed to assure isolation of low pressure portions of the RHR system from RCS at high pressure. Subsection 14.2.12.1.8(3)(i) has not been modified as stated. The acceptability of this response will be determined upon evaluation of the individual test abstracts when the SSAR has been revised accordingly. This is an open item.

b. RHR system pressure relief (R.G.1.139.C.3).

The design of the RHR system includes the relief capability and capacity required by the above referenced position, in accordance with the applicable ASME code. GE has indicated that the verification of the proper setting of relief valves is a vendor bench test required per the same ASME code, and thus no specific additional preoperational test is needed. The acceptability of this response will be determined upon evaluation of the individual test abstracts. This is an open item.

14.3 Conclusion

The staff performs reviews of plant initial test programs in accordance with Section 14.2, of NUREG-0800, the SRP. The staff reviews eight areas relating to initial plant test programs, described in Chapter 14 of the SSAR, submitted by GE as part of its operating license (OL) application. These areas of review are: (1) Summary of the Test Program and Objectives; (2) Test Procedures; (3) Test Programs' Conformance with RGs; (4) Utilization of Reactor Operating and Testing Experiences in the Development of the Test Program; (5) Trial Use of Plant Operating and Emergency Procedures; (6) Initial Fuel Loading and Initial Criticality; (7) Test Program Schedule and Sequence and; (8) Individual Test Descriptions.

Based on a review of the GE ABWR ITP and the response to the RAIs, the staff concludes that the ITP is generally comprehensive and covers most areas of the staff's concerns; however, the ITP description, as noted above, does not provide sufficient detail with respect to all test descriptions to enable the staff to determine the adequacy of GE's commitments and the tests described. Additionally, an evaluation of test abstract coverage of system specific test requirements will be made upon modification of individual test abstracts in response to the RAI, as noted above. Therefore, the GE ABWR ITP will be found acceptable when the identified open items have been resolved, and an acceptable evaluation of test abstract coverage of system specific test requirements has been determined.

ACCIDENT ANALYSIS

The accident analysis for ABWR has been reviewed in accordance with SRP Section 15. Conformance with the acceptance criteria, except as noted for each of the sections, formed the basis for deciding if the design of the facility for each of the areas reviewed is acceptable.

Two groups of design-basis events are evaluated in this section. These two groups are anticipated operational occurrences and accidents. For the analysis of events in either group to be acceptable, it is required that a conservative model of the reactor be used and that all appropriate systems whose operations (or postulated misoperations) would affect the event be included. Anticipated operational occurrences are expected to occur during the life of the plant and are analyzed to ensure that (1) they will not cause damage to either the fuel or to the reactor coolant pressure boundary, and that (2) the radiological dose is maintained within 10 CFR Part 20 guidelines. Design-basis accidents are not expected to occur but are postulated because their consequences would include the potential for the release of significant amounts of radioactive material. They are analyzed to determine the extent of fuel damage expected and to ensure that reactor coolant pressure boundary damage, beyond that assumed initially to be the design-basis accident, will not occur and that the radiological dose is maintained within 10 CFR Part 100 guidelines.

For loss-of-coolant accidents, the acceptance criteria for the emergency core cooling system specified in 10 CFR 50.46 are:

(1) The peak cladding temperature must remain below 2200°F.

(2) Maximum cladding oxidation must nowhere exceed 17 percent of the total cladding thickness before oxidation.

(3) Total hydrogen generation must not exceed 1 percent of the hypothetical amount that would be generated if all the metal in the cladding cylinders, excluding the cladding surrounding the plenum volume, were to react.

(4) The core must be maintained in a coolable geometry.

(5) Calculated core temperatures after successful initial operation of the emergency core cooling system shall be maintained acceptably low, and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

The staff evaluation of loss- of-coolant accident analysis is given in Section 6.3 of this report.

To demonstrate the adequacy of the plant's engineered safety features (ESFs), the applicant calculated the offsite consequences that could result from the occurrence of each of several other design-basis accidents and presented the results of these computations in the Standard Safety Analysis Report (SSAR). The staff's review of this area was performed in accordance with the review guidelines and acceptance criteria in SRP Section 15.0 (NUREG-0800).

15.1 Anticipated Operational Occurrences

Anticipated operational occurrences (AOOs) are those transients expected to occur during normal or planned modes of plant operation. The acceptance criteria for these transients are based on GDCs 10, 15, and 20. GDC 10 specifies that the reactor core and associated control and instrumentation systems be designed with appropriate margin to ensure that acceptable fuel-design limits are not exceeded during any condition of normal operation, including the effects of AOOs. GDC 15 specifies that sufficient margin shall be included to ensure that design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including AOOs. GDC 20 specifies that a protection system be provided that automatically initiates appropriate systems to ensure specified acceptable fuel design limits are not exceeded during any condition of normal operation including AOOs.

Specific acceptance criteria in the SRP for transients that occur with moderate frequency are:

(1) Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design values according to ASME Code, Section III, Article NB-7000. For the ABWR, which has a design pressure of 1250 psig, the pressure should not exceed 1375 psig during any AOO.

(2) Fuel-cladding integrity should be maintained by ensuring that the reactor core is designed with appropriate margin during any conditions of normal operation, including the effects of AOOs. For boiling water reactors, the minimum value of the critical power ratio reached during the transient should be such that 99.9 percent of the fuel rods in the core would not be expected to experience boiling transition during core-wide transients. This limiting value of the minimum critical power ratio, called the safety limit, for ABWR is 1.07.

(3) An incident that occurs with moderate frequency should not generate a more serious plant condition unless other faults occur independently.

(4) An incident that occurs with moderate frequency in combination with any single active component failure, or operator error, should not result in loss of function of any barrier other than the fuel cladding. A limited number of fuel-rod-cladding perforations is acceptable. (See II.K-3.44 of the TMI-2 Requirements.)

The analyses of the abnormal operational transients were performed using the computer simulation model REDYA, which is the ABWR version of REDY. REDY is described in the GE Topical Report NEDO-10802. The REDY code has been reviewed by the staff and found acceptable. The pressurization transients were performed using the computer simulation model ODYNA, which is the ABWR version of ODYN. ODYN is described in Topical Report NEDO-24154. The staff has reviewed the ODYN code and found it acceptable as documented in "Safety Evaluation for the General Electric Topical Report Qualification of the One Dimensional Core Transient Model for Boiling Water Reactors," NEDO-24154 and 24154-P Volumes I, II, and III, June 1980." The revisions to REDYA and ODYNA are currently under review by the staff. Any findings that the staff makes at this time concerning the analysis presented in Chapter 15 is contingent on acceptable review finding on these codes by the staff.

The applicants used conservative assumptions with respect to initial power, scram reactivity, reactivity coefficients, and power profiles in the analyses.

The transients analyzed involved the following reactor scrams required in accordance with Criterion 20 of the General Design Criteria:

(1) Reactor vessel high pressure,

(2) Reactor vessel low water level,

(3) Turbine stop valve closure,

(4) Turbine control valve fast closure,

(5) Main steam line isolation valve closure,

(6) Neutron monitoring system scram.

Appropriate time delays to trip for each scram signal were included in the analyses.

The transient events were categorized in terms of the following system parameter variations:

(1) Decrease in Core Coolant Temperature

Transients analyzed in this group included loss of feedwater heaters, feedwater control failure, runout of one feedwater pump, runout of two feedwater pumps, opening of turbine control and bypass valves, pressure regulator failure in the open direction, inadvertent opening of a safety relief valve and inadvertent residual heat removal shutdown cooling operation.

For transients categorized under Decrease in Core Coolant Temperature, the most severe transient is runout of two feedwater pumps. The resultant minimum critical power ratio reached is 1.07 and the peak vessel pressure is 11.8 kg/cm²g (168 psig) below the ASME code limit. For loss of feedwater heating transient, the applicants assumed a 131°F drop in feedwater temperature. However, a drop of 150°F has occurred at a domestic boiling water reactor as the result of an electrical component failure. We require GE to analyze the loss of feedwater heater event with a greater drop in feedwater temperature, or provide adequate justification for selection of 131°F temperature decrease.

Inadvertent safety relief valve opening causes a decrease in reactor coolant inventory and results in a mild depressurization event which has only a slight effect on fuel thermal margins. Changes in surface heat flux are calculated to be negligible indicating an insignificant change in minimum critical power ratio. Thus, the transient is found to be acceptable.

Inadvertent RHR shutdown cooling operation event is categorized as a limiting fault, rather than a moderate frequency event. This is a significant deviation from the SRP. We require GE to submit additional detailed sufficient justification for the recategorization, or to categorize the event as one of moderate frequency.

(2) Increase in Reactor Pressure

Transients in this group included generator load rejection and turbine trip with and without turbine bypass, inadvertent MSIV closure, loss of condenser vacuum, loss of auxiliary power transformer, loss of all grid connections, loss of all feedwater flow, and failure of residual heat removal shutdown cooling.

The transient resulting in the highest system pressure was a generator load rejection without turbine bypass which resulted in a peak system pressure about 10.6 kg/cm^2g (149 psig) below the allowable maximum pressure of 96.7 kg/cm^2g (1375 psig).

(3) Decrease in Reactor Coolant System Flow Rate

Transients in this group included trip of three reactor internal pumps, trip of all reactor internal pumps, runback of reactor internal pumps and recirculation flow control failure to decrease flow.

For transients categorized under Decrease in Reactor Coolant System Flow Rate, the most severe transient is that resulting from simultaneous trip of all reactor internal pumps. GE analyzed pressure regulator downscale failure and trip of all reactor internal pumps (RIPs) as limiting fault events rather than moderate frequency events as identified in the SRP. The ABWR feedwater control system and the steam bypass and pressure control system use a triplicated digital system. GE claims that the probability of failure of the pressure regulator is very low. GE also claims that the probability of simultaneous failure of the motor generator sets of the RIPs control systems is low due to the advanced instrumentation and control system. Hence, the above events are analyzed as limiting faults.

Since the staff has not completed the review of electrical, instrumentation and control portions of the ABWR, it is premature to arrive at conclusions for the reliability of these systems. The staff requires a detailed justification from GE supporting the deviation. We are continuing to assess submitted information and are working with GE to resolve the issue. If adequate justification is not provided on recirculation system reliability (including control and power functions) to support categorizing as a limiting fault event, they must be reanalyzed using moderate frequency criteria. This is an open issue.

(4) Reactivity and Power Distribution Anomalies

Transients in this group included rod withdrawal error, abnormal startup of one reactor internal pump, fast runout of reactor internal pumps and control rod misoperations. The startup of an idle reactor internal pump is categorized under reactivity anomalies. This event is not a limiting transient and neither primary pressure boundary nor fuel damage criteria are exceeded.

(a) Rod Withdrawal Error at Low Power

The applicants have examined the design of the rod control system to ascertain if a single failure can lead to the uncontrolled withdrawal of a control rod during refueling and during startup and low power operation. During refueling operations interlocks assure that all control rods are inserted while fuel is being handled over the core. When no fuel is being handled, a maximum of one rod may be withdrawn.

However, the control system is designed (see Section 4.3.2) so that the core is subcritical with the highest worth rod withdrawn. Finally, the removal (from the top) of a control rod is not physically possible without removing the four fuel assemblies which surround the rod. Therefore, GE has not provided an analysis of control rod removal error during refueling. This is in accord with approvals for current BWRs and is acceptable.

GE claims that the uncontrolled withdrawal of a rod during reactor startup is prevented by the Rod Block Control System function of the Rod Control and Information System. This system enforces the banked position withdrawal sequence. Thus rod withdrawals other than those permitted in normal operation will be precluded. SRP Guidance (15.4.1) states that this transient need not be considered if single failures cannot cause the sequence. The presence of single failures of the reactor control system which could result in the uncontrolled withdrawal of control rods during refueling conditions is under review. The associated instrumentation is also still under review. However, in accordance with staff requirements on current BWRs, GE has also analyzed the erroneous withdrawal of a high worth control rod and has found that the results fall well within MCPR and other fuel criteria limits. This provides an acceptable analysis for the postulated event.

(b) Rod Withdrawal Error at Power

The causes of a potential rod withdrawal error transient are either a procedural error by the operator such that a gang of control rods is withdrawn continuously, or a malfunction of the automated rod withdrawal control logic during automated operation in which a gang of control rods is withdrawn continuously.

In ABWR, the multi-channel rod block monitor (MRBM) subsystem logic issues a rod block signal that is used in the Rod Control and Information System (RC&IS) logic to enforce rod blocks that prevent fuel damage by assuring that the Minimum Critical Power Ratio (MCPR) and Maximum Linear Heat Generation Rate (MLHGR) do not violate the fuel thermal operating and safety limits. The operating thermal limits rod block function will block rod withdrawal when the operating thermal operating limit is reached.

The rod block algorithm and setpoint are based on on-line core information, e.g., core flow and Local Power Range Monitor readings, which are used to calculate the fuel status relative to limits. GE has not presented these algorithms. We will require GE to submit the algorithms for review.

(5) Increase in Reactor Coolant Inventory

The transient analyzed was inadvertent startup of the high pressure core flooder pump startup (feedwater flow control failure to maximum demand was covered in Category 1).

The transient which could cause unplanned addition to coolant inventory is the inadvertent actuation of the high pressure core flooder system. The high pressure core flooder system actuation has a small effect, because its flow is small compared to the recirculation flow. The transient has little effect on fuel thermal margins and on reactor system pressure. We agree with the applicants' analysis and plant response is therefore acceptable.

(6) Decrease in Reactor Coolant Inventory

The anticipated operational occurrence - the inadvertent opening of a safety relief valve is covered in Category 1.

In response to a staff question concerning credit taken for nonsafety related equipment in analysis of anticipated transients, GE responded with the following list: (a) relief function of safety relief valves, (b) high water level 8 trip, (c) turbine bypass valves, (d) reactor internal pump trip (RPT) on load/turbine trip. It is the staff position, that for ABWR no credit be taken for nonsafety grade equipment in the transient and accident analysis. GDCs 1-4 requires that components important to safety shall be designed to quality standards etc., and GDC 21 requires that the protection system shall be designed for high functional reliability. The events should be reanalyzed taking credit only for safety grade components and equipment. This is an open item.

By letter dated August 23, 1989, the GE informed Gulf States Utilities Company (GSU) of a condition potentially reportable under 10 CFR Part 21, applicable to the River

Bend Station (RBS), involving a slow closure of one main turbine control valve. This low probability event which was not previously considered results from a turbine control valve that GE assumes to close due to an unspecified failure in the turbine control circuit or in the servomechanism hardware. According to GE, if the valve closes in less than 2.3 seconds, a reactor scram is initiated as a result of high neutron flux and no safety limits are exceeded. However, if the valve closes in greater than 2.3 seconds, the reactor scram is initiated by high reactor pressure. During this slow closure case, the minimum critical power ratio (MCPR) safety limit may be exceeded if the maximum combined flow limiter is set for less than 113 percent of rated steam flow. The consequences of this postulated event is based on GE's assessment of a generic BWR/6 analysis. We require GE to address the event for ABWR applicability. This is an open issue.

15.2 Accidents

The applicant analyzed reactor internal pump seizure and shaft break accidents. The cause of reactor internal pump (RIP) seizure and shaft break represents the unlikely event of instantaneous stoppage of the pump motor shaft of one reactor internal pump out of total ten reactor internal pumps. These events produces a very rapid decrease of pump flow as a result of the large hydraulic resistance introduced by the stopped rotor or shaft. Consequently, a decrease in core inlet flow and core cooling capability occurs. However, with only one out of ten RIPs seized, the core flow decrease is small (<10 percent) so the event is mild. The RIP seizure and shaft break do not result in any fuel failure. This satisfies the criteria of 10 CFR Part 100 and is therefore acceptable.

The applicant also analyzed misplaced fuel bundle accident, rod ejection accident and control rod drop accident.

(1) Misplaced Fuel Bundle Accident

Three errors must occur for this event to take place in the equilibrium core loading. First, a bundle must be misloaded into a wrong location in the core. Second, the bundle which was supposed to be loaded where the mislocation occurred is also put in an incorrect location or discharged. Third, the misplaced bundles are overlooked during the core verification process performed following core loading. A fuel loading error, undetected by in-core instrumentation following fueling operations, may result in an undetected reduction in thermal margin during power operations. GE evaluated the consequences of misplaced fuel bundle accident and has concluded that the misplaced fuel bundle accident will not release any radioactive material from the fuel. This satisfies the criteria of 10 CFR Part 100 and is therefore acceptable.

(2) Rod Ejection Accident

The rod ejection accident is caused by a major break on the FMCRD housing, outer tube or associated CRD pipe lines. The consequence of a rod ejection accident is similar to the rod drop accident, in that the fuel enthalpy criteria may be violated if the speed of the ejected rod and/or the reactivity added are large enough. The same criteria of 280 cal/gm is applied to the rod ejection accident.

A redundant brake mechanism is installed in the FMCRD system (2 brakes per FMCRD) to prevent severe consequences resulting from this accident. Even if this accident does happen, the brake effectively terminates this event and prevents any severe consequence from this event.

The results of the rod ejection event show a peak enthalpy less than 70 cal/gm, less than the acceptable criteria of 280 cal/gm. The radiological consequences are bounded by the analysis of the control rod drop accident. Therefore, the plant response is acceptable.

(3) Rod Drop Accident

The locking piston control rod drive mechanism used in current BWRs cannot detect separation of the control rod from the drive mechanism during normal rod movements. In order to prevent damage to the nuclear system process barrier by the rapid reactivity increase which would result from a free fall of a control rod (rod drop accident) from its fully inserted position to the position where the drive mechanism is withdrawn, a velocity limiter is provided on the control rod to restrict the control rod free-fall velocity to acceptable limits.

In contrast to the locking pist n control rod drive, the fine motion control rod drive (FMCRD) is designed to detect separation of the control rod from the drive mechanism. Two redundant and separate Class 1E switches are provided to detect the separation of either the control rod from the hollow piston or the hollow piston from the ball nut.

Actuation of either of these switches will cause an immediate rod block and initiate an alarm in the control room, thereby preventing a rod drop accident from occurring.

The radiological consequences of the control rod drop accident are provided in Section 15.3.2 of this report.

15.3 Radiological Consequences of Accidents

In order to demonstrate the adequacy of the ABWR engineered safety features designed to mitigate the radiological consequences of design-basis-accidents (DBAs), GE assessed the offsite radiological consequences which could result from the occurrence of each of several DBAs and presented the results of the offsite dose calculations in Chapter 15 of the SSAR. To verify GE's assessment and calculations, the staff has independently performed its own assessment for radiological consequences resulting from (1) control rod drop accident, (2) fuel handling accident, (3) main steamline break accident outside containment, and (4) failure of small lines carrying primary coolant outside containment. The staff will complete independent calculations for the offsite radiological consequences and the control room operator doses due to a loss-of-coolant accident (LOCA) at a later date after resolution of the following three specific licensing open items with GE.

(1) Pressure Suppression Pool as a Fission Product Cleanup System.

GE assumed that any elemental and particulate iodine species purged to the suppression pool would be subject to a decontamination factor (DF) of 10. RG 1.3 allows no credit for the retention of iodine by a BWR suppression pool, while SRP Section 6.5.5, "Pressure Suppression Pools as Fission Product Cleanup Systems," issued in December 1988, stated that suppression pools are capable of scrubbing airborne fission products and that to ignore this capability would be an undue conservatism. Therefore, the staff finds that a credit may be given for the removal of suppression pool fission products provided that suppression pool decontamination factors are evaluated in accordance with the methodology prescribed in the revised SRP Section 6.5.5.

To justify the suppression pool DF of 10, GE has committed to provide NRC the fraction of the drywell atmosphere that bypasses the suppression pool by leaking through drywell penetrations. GE's evaluation of the bypass, including a technical justification for the fraction selected, should be based on the complete spectrum of accidents and transients considering the following:

a) The time and rate of release of the source term. This is the critical assumption of the analysis since it will determine whether or not there is a steam driving force present to force the source term into the pool. Without the driving force, the source term will remain in the drywell. Therefore, no credit would be given for pool scrubbing.

(b) Rate of return back to the drywell. After being scrubbed by the pool, the source must have a way to return back to the drywell. The pathway would be via the drywell to wetwell vacuum breakers.

GE's model should consider these two processes and calculate the corresponding source term in drywell as a function of time.

Future applications incorporating the ABWR design will be required to submit to NRC a proposed program for periodic inspection and surveillance tests to confirm suppression pool water depth and drywell leak tightness, consistent with the bypass fraction used in computing the pool decontamination factor and in assessing the radiological consequences. This is an interface requirement which will be included in the ABWR reactor operating license technical specifications.

(2) Primary Containment Leakage Rates

GE assumed that the primary containment leak rate into the reactor building through penetrations and engineered safety feature system components will not be greater than an equivalent release of 0.5 percent by volume per day of the primary containment free air volume for the first 24 hours after a LOCA and half of that value (0.25 percent per day) after 24 hours.

GE's assumption in leakage reduction is based upon the evaluation that the primary containment pressure is reduced by over a factor of one half by twelve hours from the time of a LOCA and therefore the driving force for leakage via the pathway is correspondingly reduced.

RG 1.3 assumes a constant containment leak rate for the duration of a LOCA, although it permits a reduced leak rate with supporting justification. The rationale for a constant containment leak rate is two fold. First, the pressure profile for a BWR does remain at high pressures for a long period of time. Secondly, for most plants, the leakage is only measured by Appendix J to 10 CFR Part 50 at the maximum value.

To allow GE to take credit for the pressure time profile and therefore reduced leakage rate as a function of time, GE should develop a bounding pressure profile and show it to be conservative for all credible accident and transient events over the entire time span. In addition, the leakage rate as a function of pressure must also be demonstrated that it is conservative over the entire range. This profile must be supported by test. The containment leak rates are ABWR interface requirements which will be included in the ABWR reactor operating license technical specifications.

(3) Radioactive Iodine Deposition in the Main Steam Lines and Condensers.

GE analyzed two specific pathways in releasing fission products from the primary containment to the environment. The firstpathway is leakage into the reactor building as described in item (1)(b) above.

The second release pathway is via the main steam line through leakage in the main steam line isolation valves. GE assumed that a pathway exists which permits a direct access from the primary containment atmosphere to the main steam lines, that the main steam line isolation valves leak at the maximum technical specification, and that one isolation valve fails in the open position. Flow from the isolating valves is then directed through the main steam lines into the turbine condenser via the turbine bypass lines.

GE further assumed plate out of elemental and particulate iodine species in the condenser. GE states that the use of the main steamlines and condenser as a fission product mitigation pathway is predicted upon the assumption that these structures remain intact to the extent that they are able to transport the MSIV leakage into the condenser during a LOCA event involving a corresponding seismic event. It further states that though not specifically designed or designated as seismically qualified structures, the use of standard engineering practice in the design of these components results in structures of sufficient strength and flexibility to withstand design basis seismic events.

The staff is currently reviewing the GE proprietary report, NEDO-31643-P, titled "Increasing Main Steam Isolation Valve Leakage Rate Limits and Elimination of Leakage Control Systems" submitted by BWR Owner's Group

(BWROG). The BWROG has submitted a revision to this report for the staff's evaluation. This document is still under review.

Plateout of radioactive iodine on the main steam pipe and condenser surfaces can realistically provide significant dose mitigation. Several technical references indicated that particulate and elemental iodines would be expected to deposit on surfaces, with rates of deposition varying with temperature, pressure, gas composition, surface material, and particulate size. On the basis of our review of these references, the staff is evaluating whether a credit for the fission product attenuation in the main steam lines and for the isolated condenser is appropriate and reasonable for BWRs even though the main steam lines downstream from the MSIV and its condenser are not designed to withstand the Safe Shutdown Earthquake as defined in 10 CFR Part 100, Section III.C. The staff expects to complete the review of the GE topical report and develop a staff position not later than April 1992 and its findings and conclusions will be used to resolve this open item for the ABWR design.

The computed doses resulting from DBAs other than a LOCA are listed in Table 15-1. The computed doses in Table 15-1 are in a form which allows direct comparison with the dose reference values of 10 CFR Part 100.11 in that they are expressed as rems of thyroid and whole body exposure for a 2-hour period at the exclusion area boundary and for a 30-day period at the boundary of the low population zone. Since no specific site is associated with this standard plant, these two boundaries are defined only in terms of hypothetical atmospheric diffusion parameters (X/Q's) proposed by GE as site interface conditions.

The staff used in its evaluation of the radiological consequences Exclusion Area Boundary (EAB) of 800 meters (0.5 mile) and a Low Population Zone (LPZ) radius of 4800 meters (3 miles) with Pasquill F stability and persistent (greater than 95 percent of time) one meter per second wind velocity as proposed by GE. These distances represent approximate median values of EBA and LPZ of current operating reactor sites. The atmospheric diffusion parameters used in the staff's evaluation are given in Table 15-3.

The specific interface requirements which affect the future ABWR license technical specifications are as follows:

1. The primary coolant activity limits of 0.2 micro Ci per gram of dose equivalent I-131 for normal operation, and 4.0 micro Ci per gram as the limiting condition for operation.

2. Primary containment integrated leak rate of less than equivalent release of 0.5 percent by volume per day. The equivalent release includes the leakages from the ECCS systems outside the containment.

3. The main steam line isolation valve leakage specified in the ABWR plant technical specification at the time of issuance of operating licenses.

4. Inspection and surveillance requirements for the ABWR plant suppression pool bypass.

15.3.1 Loss-of-Coolant Accident

A loss-of-coolant accident will be postulated, and the radiological consequences evaluated, using assumptions and methods described in the Appendices of Section 15.6.5 of NUREG-0800 and in RG 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Boiling Water Reactors." This accident is postulated in order to determine the adequacy of the engineered safety features designed to prevent release of fission products into the environment, with the meteorological conditions prevailing at a specific ABWR site. The passive engineered safety features provided for this purpose for the ABWR are the primary containment and a secondary containment (reactor building). These are considered in conjunction with the standby gas treatment system (SGTS) which is an active engineered safety feature system.

The ABWR primary containment comprises a drywell and wetwell and supporting systems to limit fission product leakage during and following a LOCA with rapid isolation of all pipes or ducts which penetrate the containment boundary. The ABWR secondary containment is a multi-compartment, self-contained structure maintained at negative pressure with respect to the environment. Flow through the secondary containment is directed via the standby gas treatment system to the plant stack through HEPA and six-inch thick charcoal filters. Because the reactor building is designed to completely enclose the primary containment, the staff assumed no bypass leakage to the environment from the primary containment except that directed through the main steam lines. The staff further assumed an iodine removal efficiency of

99 percent by the SGTS charcoal filter with an effective negative pressure draw down time of 20 minutes. Prior to this negative pressure drawdown time, the release is considered unfiltered.

The rate at which treated leakage is released following this postulated accident is dependent upon the flow rates needed to maintain the secondary containment free volume below outside atmospheric pressure. These flow rates are in turn dependent upon the primary containment's heat and mass loss to the secondary containment, and upon the inleakage of outside air into the secondary containment. Following the guidance of NUREG-0800, leak rates are taken to be those proposed by GE (0.5 percent equivalent release by volume per day) as limiting conditions of operation in the technical specifications of the operating license.

GE has proposed that the primary containment be built and tested periodically to have a leak rate at design pressure of less than 0.5 percent equivalent release by volume per day. Since this limit includes the ECCS leakage outside the primary containment, GE should clarify the application of this limit to the testing and surveillance requirements of the containment in the ABWR technical specifications. This is an open item.

The staff performed a preliminary calculation of the offsite radiological consequence due to a LOCA using the NRC Code TACT-5 without resolution of three open items discussed in Section 15.3 of this draft SER and found that the resulting offsite doses are well in excess of the dose reference values specified in 10 CFR Part 100.11. The staff will complete the LOCA dose assessment upon resolution of the open items using the assumptions and parameters given in Table 15-2.

15.3.2 Control Rod Drop Accident

GE states the radiological consequences of a control rod drop accident need not be considered because such an accident is extremely unlikely with its improved design. The new improved design employs the fine motion control rod drive system (FMCRD) which has several new features that are unique compared to the current BWR locking piston control rod drives.

GE states that (1) for the rod drop accident to occur, it is necessary for such highly unlikely events as failures of both Class 1E separation-detection devices, or the failure of the rod block interlock, and the failure of the latch mechanism to occur simultaneously with the occurrence of a stuck rod on the same FMCRD, and that (2) therefore, there is no basis to postulate this event to occur because of the low probability of such simultaneous occurrence of these multiple independent events.

Based on past licensing reviews by the staff, a control rod drop accident is expected to result in radiological consequences less than a small fraction of the dose reference values specified in 10 CFR Part 100.11 even with conservative assumptions. The SRP Section 15.4.9 (III) states that unless unusual plant or site features are present or the applicant's calculation shows an unusually large amount of fuel damage, a specific calculation of the radiological consequences for this accident is not necessary.

However, the staff did perform a specific evaluation of this accident, since it is the first application involving a particular standardized design with hypothetical site boundaries. This evaluation should serve to establish a reference point for comparison of future applications incorporating the ABWR design.

To evaluate the radiological consequences of this accident, the staff assumed the highest worth control rod is postulated to become decoupled from its drive mechanism at a fully inserted position in the core. The drive mechanism is withdrawn, but the decoupled control rod is assumed to be stuck in place. At a later moment, the control rod suddenly falls free and drops out of the core. This results in the insertion of a large positive reactivity into the core and a localized power excursion. The termination of this excursion is accomplished by automatic safety features, and no action is required on the operator's part. The rod pattern control function of the rod control and information system (RCIS) limits the worth of any control rod by regulating the withdrawal sequence. The staff estimated that such a rod drop would cause no more than 770 fuel rods to reach the threshold for cladding damage, with no fuel melting.

The staff used the fuel fission product release assumptions consistent with RG 1.77, "Assumptions Used for Evaluating a Control Rod Ejection Accident for Boiling and Pressurized Water Reactors." These assumptions are given in Table 15-4. The computed doses, which were calculated by the staff using NRC Code TACT-5, are listed in Table 15-1, and are well within the dose reference values of 10 CFR Part 100.11. Based on these findings the staff concludes that the ABWR standard design is adequate to control the release of fission products following a postulated control rod drop accident.

15.3.3 Fuel-Handling Accident

The fuel-handling accident is assumed to occur as a consequence of a failure of the fuel assembly lifting mechanism resulting in dropping a raised fuel assembly onto the fuel in the reactor core.

A postulated fuel-handling accident for the ABWR was evaluated in accordance with the guidance of Section 15.7.4 of NUREG-0800, and using assumptions consistent with Positions C.1.a through C.1.k of RG 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling Storage Facility for Boiling and Pressurized Water Reactors." The kinetic energy of a single falling fuel assembly was assumed to break open the maximum possible number of fuel rods using perfect mechanical efficiency. Instantaneous release of noble gases and radioiodine vapor from the gaps of the broken rods was assumed to occur, with the released gases bubbling up through the fuel pool water. Radiation monitors located within the normal ventilation system are designed to isolate that system automatically and direct all fuel building exhaust to the SGTS.

The list of the assumptions used for the ABWR application of the RG 1.25 positions is provided in Table 15-5. The offsite doses which were computed by the staff with NRC Code TACT-5 using these assumptions at hypothetical site boundaries, are listed in Table 15-1, and are well within the dose reference values of 10 CFR Part 100.11. Based on these findings, the staff concludes that the standard ABWR design is adequate to control the release of fission products following a postulated fuel-handling accident.

The spend fuel cask drop accident is assumed to occur as a consequence of an unspecified failure of the cask lifting mechanism thereby allowing the cask to fall approximately 64 feet (provided by GE) from the level of the refueling floor to ground level through the refueling floor maintenance hatch.

GE stated that each cask will have the maximum capacity to contain 1116 spent fuel rods based on the largest capacity cask projected to be available. GE proposed and the staff accepted the minimum fuel storage (decay) time of 120 days prior to cask loading operation to commence after reactor fueling.

The list of the assumptions used for the spent fuel cask drop accident is listed also in Table 15-5 and the offsite doses computed for this accident are given in Table 15-1. The calculated offsite doses are well within the dose reference values specified in 10 CFR Part 100.11 and therefore, the staff also concludes that the standard ABWR design is adequate to control the release of fission products following a postulated spent fuel cask drop accident.

15.3.4 Steamline Break Accident

One of the four main steam lines was postulated to rupture between the outer isolation valve and the turbine control valves. GE has analyzed this hypothetical accident and had concluded that no more than 76,770 lbs of reactor coolant would be lost through the break prior to automatic isolation, of which less than 28,737 lbs would be lost as steam. The staff evaluation of this accident, however, followed the assumptions of Section 15.6.4 of NUREG-0800 in assuming that 140,000 lbs of reactor coolant (as the maximum upper bound value) is lost, with all of the contained iodine becoming airborne.

Two reactor coolant conditions were assumed for the evaluation. In Case 1, the lost coolant was assumed to be contaminated with radioactive iodine to the limit allowed by the Standard Technical Specifications for boiling-water reactors during normal operation, which is a concentration of 0.2 micro Ci per gram of doseequivalent I-131. In Case 2, a concentration of 4.0 micro Ci per gram of doseequivalent I-131 was assumed (Standard Technical Specification limit above which the reactor would be required to be shut down). The acceptance criteria of NUREG-0800 are the dose reference values of 10 CFR Part 100.11 for Case 2, and less than 10 percent of these values for Case 1. Dose-equivalent I-131 is defined as any mixture of iodine isotopes yielding the same inhalation thyroid dose as the stated amount of pure I-131. The staff also considered the amounts of 13 noble gas isotopes which would also be released. The staff calculated the estimated offsite doses using a specific primary coolant activity value, atmospheric dispersion factors, representative breathing rates (standard person), and dose conversion factors. The resulting estimated doses are listed in Table 15-1, and are within the acceptance criteria of Section 15.6 of NUREG-0800.

15.3.5 Small Line Break Accident

GDC 55 specifies provisions to ensure isolation of all pipes carrying reactor coolant that penetrate the containment building. Exempted from these specifications are small-diameter pipe which must be continuously connected to the primary coolant system in order to perform necessary functions. For these lines, generally called instrument lines, methods of mitigating the consequences of a rupture are necessary because of the lack of isolation capability.

This accident postulates that a small steam or liquid line break inside or outside the primary containment and that a small instrument line, instantaneously and circumferentially, breaks at a location where it may not be able to be isolated and where detection is not automatic or apparent.

Based on past licensing review by the staff, a small line break accident is expected to result in radiological consequences less than a small fraction of the dose reference values specified in 10 CFR Part 100.11. Furthermore, the staff believes that these postulated breaks are subsumed by the design basis loss-of-coolant accident radiological consequences as indicated in Section 15.6.2 of NUREG-0800, "Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment."

The foregoing comments not withstanding, the staff did perform a specific evaluation of this accident, since it is the first application involving a particular standardized design with hypothetical site boundaries. This evaluation should serve to establish a reference point for comparison of future applications incorporating the ABWR design.

The assumptions used for the evaluation are provided in Table 15-6. The staff calculated the estimated offsite doses using a specific primary coolant activity value, atmospheric dispersion factors, representative breathing rates (standard person), and dose conversion factors. The computed doses are listed in Table 15-1, and are well within the dose reference values of 10 CFR Part 100.11. Based on these findings, the staff concludes that the ABWR standard design is adequate to control the release of fission products following a postulated small line break accident.

15.4 Anticipated Transient Without Scram

The staff review of Reactor Internal Pump Trip (RPT), Alternate Rod Insertion (ARI) and Standby Liquid Control System (SLCS) systems are not complete. The acceptability of the above systems for ATWS Rule 10 CFR 50.62 compliance will be given in the final safety evaluation for the ABWR.

GE submitted ATWS analysis in Appendix 15E of the SSAR. Our preliminary audit calculations for ATWS indicated the possibility of a delayed shutdown, high power peaking at top of the core and potential for fuel failure, for cases assuming a failure of the hydraulic scram followed by a slow electric scram (the ABWR incorporates electric-hydraulic fine motion control rod drives which use both electric motor and hydraulic pressure for reactor scram). GE's preliminary position is that Linear Heat Generation Rate (LHGR) during ATWS should be low as possible, and has agreed to consider design changes which will reduce the severity of ATWS event. GE has agreed to revise Appendix 15E - ATWS analysis. The staff will report the result of the review of GE's changes in Appendix 15E in the final safety evaluation. This issue is open.

Table 15-1
Radiological Consequences of Design-Basis Accidents

	Exclusion area* 2 hour dose (rem)		Low population zone** 30-day dose (rem)	
Postulated Accident	Thyroid	Whole body	Thyroid	Whole body
Main steamline failure outside containment				
With concomitant iodine spike	1.8	<1	<1	<1
With preaccident iodine spike	24	<1	11	<1
Rod drop accident	1.5	<1	2	<1
Fuel handling accident	5.4	<1	2	<1
Small line break accident	0.3	<1	0.2	<1
LOCA duration				
Containment leakage	TBD	TBD	TBD	TBD
0.0-2.0 hr				
2.0-8.0 hr				
8.0-24.0 hr				
24.0-96.0 hr				
96.0-720.0 hr				
ECCS	TBD	TBD	TBD	TBD
leakage				
0.0-2.0 hr				
2.0-720.0 hr				
MSIV	TBD	TBD	TBD	TBD
leakage				
0.0-2.0 hr				
2.0-720.0 hr				
Total LOCA doses		TBD		

*Exclusion area boundary (EAB) distance = 800 m.
**Low population zone (LPZ) boundary distance = 4,800 m.
TBD = to be determined

Table 15-2
Assumptions Esed to Evaluate the Loss-of-Coolant Accident

Parameter	Value
Power level, MWt	4,005
Operating time, years	3
Core fraction airborne in the drywell, (percent)	
Noble gases	100
Iodines	25
Primary containment leak rate, (percent)/day	TBD
Iodine Chemical Species, (percent)	
Organic	4
Elemental	91
Particulate	5
Suppression Pool Decontamination	
Noble Gas	1
organic Iodine	1
Elemental Iodine	TBD
Particulate Iodine	TBD
Pool Bypass	TBD
Iodine deposition factor, (percent)	
Main steam line	TBD
Main condenser	TBD
Control room free air volume, m^3	28,000
Containment free volume, ft^3	470 000
Reactor enclosure free volume, ft^3	TBD
Reactor enclosure mixing fraction, (percent)	50
Reactor enclosure building drawdown time, sec	1,200
Recirculation system flow rate, ft^3/min	19,700
Emergency system flow rate, ft^3/min	2,100
Standby gas treatment system iodine filter efficiencies, (percent)	
Elemental	99
Organic	99
Particulate	99

TBD = to be determined

Table 15-3
Atmospheric Dispersion (X/Q)
Values Used in Accident Evaluations

Time period	X/Q value (sec/m^3)
0-2 hour (EBA)*	2.2×10^{-4}
0-8 hour (LPZ)**	3.4×10^{-5}
8-24 hour (LPZ)**	1.2×10^{-5}
1-4 day (LPZ)**	4.3×10^{-6}
4-30 day (LPZ)**	9.2×10^{-7}

*Exclusion area boundary (EAB) distance = 800 m.
**Low population zone (LPZ) boundary distance = 4,800 m.

Table 15-4
Assumptions Used in Computing Rod Drop Accident Doses

Power level	4,005 MWt
Peaking factor	1.55
Number of fuel rods perforated	770
Number of fuel rods melted	6
Condenser leak rate	1.0%/day
Fraction of fission product inventory release to coolant:	
Iodines	50%
Released to condenser	10%
Available for release after plateout and partitioning	10%
Noble gases	100%
Released to condenser	100%
Atmospheric diffusion values (sec/m^3)	
0-2 hour, exclusion boundary	$2.2E^{-4}$
0-8 hour, low population zone	$3.4E^{-5}$

Table 15-5
Assumptions Used in Computing Fuel-Handling Accident Doses

Power level	4,005 MWt
Peaking factor	1.55
Number of fuel rods damaged	124
Number of fuel rods in cask	1,116
Filter iodine removal efficiencies:	
organic	99%
Elemental	99%
Shutdown time	24 hours
Inventory released from damaged rods:	
Iodine and noble gases	10%
Kr-85	35%
Iodine fraction:	
Organic	0.25
Elemental	0.75
Atmospheric diffusion values (sec/m^3)	
0-2 hour, exclusion boundary	$2.2E^{-4}$
0-8 hour, low population zone	$3.4E^{-5}$

Table 15-6
Assumptions Used to Evaluate the Main Steamline and Small Line Break Accidents Outside Containment

Parameter	Value
Mass of primary coolant released before main steam isolation valve closure, lbs	140,000
Mass of primary coolant release through small line, lbs	22,000
Fraction of iodine in the primary coolant released, (percent)	100
Fraction of noble gases released, (percent)	100
Primary coolant concentration (dose equivalent I-131), micro Ci per gram Technical Specification limits,	
normal long-term operation	0.2
normal short-term operation	4.0
Atmospheric diffusion values (sec/m^3)	
0-2 hour, exclusion boundary	$2.2E^{-4}$
0 8 hour, low population zone	$3.4E^{-5}$