AIM

Association for Information and Image Management

1100 Wayne Avenue, Suite 1100
Silver Spring, Maryland 20910
301/587-8202

Centimeter

Inches

MANUFACTURED TO AIM STANDARDS
BY APPLIED IMAGE, INC.

1 of 10

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

Main Report

Manuscript Completed: July 1994
Date Published: July 1994

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

MASTER

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

ABSTRACT

This safety evaluation report (SER) documents the technical review of the U.S. Advanced Boiling Water Reactor (ABWR) standard design by the U.S. Nuclear Regulatory Commission (NRC) staff. The application for the ABWR design was initially submitted by the General Electric Company, now GE Nuclear Energy (GE), in accordance with the procedures of Appendix O of Part 50 of Title 10 of the Code of Federal Regulations (10 CFR Part 50). Later GE requested that its application be considered as an application for design approval and subsequent design certification pursuant to 10 CFR § 52.45.

The U.S. ABWR design is similar to the international ABWR design, which was being built at the Kashiwazaki Kariwa Nuclear Power Generation Station, at the time of the staff's review, by the Tokyo Electric Power Company, Inc. The ABWR is a single-cycle, forced-circulation, boiling water reactor (BWR) with a rated power of 3926 megawatts thermal (MWt) and a design power of 4005 MWt. Many features of the ABWR design are similar to those of BWR designs that the staff had previously approved. To the extent feasible and appropriate, the staff relied on earlier reviews for those

ABWR design features that are substantially the same as those previously considered. The SERs for the other BWR designs have been published and are available for public inspection at the NRC Public Document Room, 2120 L Street, N.W., Washington, D.C. 20037. Unique features of the ABWR design include internal recirculation pumps, fine-motion control rod drives, microprocessor-based digital logic and control systems, and digital safety systems.

On the basis of its evaluation and independent analyses, the NRC staff concludes that, subject to satisfactory resolution of the confirmatory items identified in Section 1.8 of this SER, GE's application for design certification meets the requirements of Subpart B of 10 CFR Part 52 that are applicable and technically relevant to the U.S. ABWR standard design. A copy of the report by the Advisory Committee on Reactor Safeguards required by 10 CFR § 52.53 is provided in Chapter 21. A final design approval, issued on the basis of this SER, does not constitute a commitment to issue a permit or license, or in any way affect the authority of the Commission, the Atomic Safety and Licensing Board, and other presiding officers, in any proceeding pursuant to Subpart G of 10 CFR Part 2.

CONTENTS

VOLUME 1: MAIN REPORT

	Page
ABSTRACT	iii
1 INTRODUCTION AND GENERAL DISCUSSION	1-1
1.1 Introduction	1-1
1.2 General Design Description	1-2
1.2.1 GE's Design Process for the U.S. ABWR	1-3
1.2.2 Precertification and Postcertification Design Control Procedures	1-3
1.2.3 The role of the COL Applicant or Licensee	1-3
1.2.4 Control of Design Documentation in Support of the Certified Design	1-3
1.2.5 GE's Agreement With the Japanese	1-3
1.2.6 Scope of Certified Design	1-3
1.2.7 Summary of U.S. ABWR Standard Design	1-4
1.3 Comparison With Similar Facility Designs	1-5
1.4 Identification of Agents and Contractors	1-5
1.5 Summary of Principal Review Matters	1-5
1.6 Index of Applicable Regulations and Exemptions	1-6
1.7 Index of Tier 2* Information	1-7
1.8 Index of Confirmatory Items	1-8
2 SITE CHARACTERISTICS	2-1
2.1 Geography and Demography	2-1
2.1.1 Site Location and Description	2-1
2.1.2 Exclusion Area Authority and Control	2-1
2.1.3 Population Distribution	2-1
2.2 Nearby Industrial, Transportation, and Military Facilities	2-1
2.2.1 and 2.2.2 Identification of Potential Hazards in Site Vicinity	2-1
2.2.3 Evaluation of Potential Accidents	2-1
2.3 Meteorology	2-1
2.3.1 Regional Climatology	2-1
2.3.2 Local Meteorology	2-4
2.3.3 Onsite Meteorological Measurements Program	2-4
2.3.4 Short-Term Dispersion Estimates for Accidental Atmospheric Releases	2-4
2.3.5 Long-Term Diffusion Estimates	2-4
2.4 Hydrologic Engineering	2-4
2.4.1 Hydrologic Description	2-4
2.4.2 Floods	2-4
2.4.3 Probable Maximum Flood on Streams and Rivers	2-4
2.4.4 Ice Effects	2-4
2.4.5 Cooling Water Channels and Reservoirs	2-4
2.4.6 Channel Diversion	2-5
2.4.7 Flooding Protection Requirements	2-5
2.4.8 Cooling Water Supply	2-5
2.4.9 Accidental Release of Liquid Effluents in Ground and Surface Waters	2-5
2.4.10 Technical Specification and Emergency Operation Requirement	2-5
2.5 Geology, Seismology, and Geotechnical Engineering	2-5
2.5.1 Basic Geologic and Seismic Information	2-5
2.5.2 Vibratory Ground Motion	2-5

	Page
2.5.3 Surface Faulting	2-5
2.5.4 Stability of Subsurface Materials and Foundations	2-6
2.6 Site Parameter Envelope	2-9
2.6.1 Wind and Tornado Design Site Parameters	2-10
2.6.2 Water Level (Flood) Design Site Parameters	2-10
2.6.3 Seismology Site Parameters	2-10
2.6.4 Soil Properties Site Parameters	2-11
2.6.5 Precipitation (for Roof Design) Site Parameters	2-11
2.6.6 Design Temperature Site Parameters	2-11
2.6.7 Atmospheric Dispersion (χ/Q)	2-11
 3 DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS	 3-1
3.1 General	3-1
3.1.1 Elimination of Operating Basis Earthquake from Design Consideration	 3-1
3.2 Classification of Structures, Systems, and Components	3-9
3.2.1 Seismic Classification	3-9
3.2.2 System Quality Group Classification	3-11
3.3 Wind and Tornado Loadings	3-13
3.3.1 Wind Design Criteria	3-13
3.3.2 Tornado Design Criteria	3-14
3.4 Water Level (Flood) Design	3-15
3.4.1 Flood Protection	3-15
3.4.2 Water Level (Flood) Design Procedure	3-19
3.5 Missiles	3-20
3.5.1 Missile Selection and Description	3-20
3.5.2 Structures, Systems, and Components To Be Protected From Externally Generated Missiles	 3-27
3.5.3 Barrier Design Procedures	3-29
3.6 Protection Against Dynamic Effects Associated With the Postulated Rupture of Piping	 3-30
3.6.1 Plant Design for Protection Against Postulated Piping Failure in Fluid Systems Outside Containment	 3-30
3.6.2 Determination of Rupture Locations and Dynamic Effects Associated With the Postulated Rupture of Piping	 3-33
3.6.3 Leak-Before-Break Evaluation Procedures	3-36
3.7 Seismic Design	3-36
3.7.1 Seismic Design Parameters	3-37
3.7.2 Seismic System Analysis	3-38
3.7.3 Seismic Subsystem Analysis	3-48
3.7.4 Seismic Instrumentation	3-48
3.8 Design of Seismic Category I Structures	3-49
3.8.1 Concrete Containment	3-49
3.8.2 Steel Components of the Reinforced Concrete Containment	3-51
3.8.3 Concrete and Steel Internal Structures of Steel or Concrete Containment	 3-51
3.8.4 Other Seismic Category I Structures	3-53
3.8.5 Foundations	3-56
3.8.6 Certified Design Material	3-58
3.9 Mechanical Systems and Components	3-58
3.9.1 Special Topics for Mechanical Components	3-58
3.9.2 Dynamic Testing and Analysis of Systems, Components, and Equipment	 3-60

3.9.3	ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures	3-66
3.9.4	Control Rod Drive Systems	3-75
3.9.5	Reactor Pressure Vessel Internals	3-76
3.9.6	Testing of Pumps and Valves	3-77
3.10	Seismic and Dynamic Qualification of Mechanical and Electrical Equipment	3-84
3.10.1	Conclusions	3-87
3.10.2	Methods and Procedures of Analysis or Testing of Supports of Electrical Equipment and Instrumentation	3-87
3.11	Environmental Qualification of Mechanical and Electrical Equipment	3-88
3.11.1	Completeness of Qualification of Electrical Equipment Important to Safety	3-89
3.11.2	Qualification Methods	3-89
3.11.3	Completeness of Information in Tables of SSAR Appendix 3I	3-91
3.11.4	Adequacy of Interface Requirements	3-95
3.11.5	Conclusions	3-95
3.12	Piping Design	3-95
3.12.1	Introduction	3-95
3.12.2	Codes and Standards	3-96
3.12.3	Analysis Methods	3-98
3.12.4	Piping Modeling	3-101
3.12.5	Pipe Stress Analysis Criteria	3-104
3.12.6	Pipe Support Criteria	3-112
3.12.7	High-Energy Line-Break Criteria	3-116
3.12.8	Leak-Before-Break Criteria	3-118
3.12.9	Generic Piping Design ITAAC	3-118
4	REACTOR	4-1
4.1	General	4-1
4.2	Fuel System Design	4-1
4.3	Nuclear Design	4-4
4.4	Thermal Hydraulic Design	4-6
4.5	Reactor Materials	4-8
4.5.1	Control Rod Drive System Structural Materials	4-8
4.5.2	Reactor Internal Materials	4-9
4.6	Functional Design of Fine Motion Control Rod Drive System	4-9
5	REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS	5-1
5.1	Introduction	5-1
5.2	Integrity of Reactor Coolant Pressure Boundary	5-1
5.2.1	Compliance With Code and Code Cases	5-1
5.2.2	Overpressure Protection	5-4
5.2.3	Reactor Coolant Pressure Boundary Materials	5-5
5.2.4	Inservice Inspection and Testing of the Reactor Coolant Pressure Boundary	5-7
5.2.5	Reactor Coolant Pressure Boundary Leakage Detection	5-10
5.3	Reactor Vessel	5-15
5.3.1	Reactor Vessel Materials	5-15
5.3.2	Pressure-Temperature Limits	5-17
5.3.3	Reactor Vessel Integrity	5-18
5.4	Components and Subsystem Design	5-18
5.4.1	Reactor Recirculation System	5-18
5.4.2	Steam Generators	5-19

	Page
5.4.3 Reactor Coolant Piping	5-19
5.4.4 Main Steam Line Flow Restrictions	5-19
5.4.5 Main Steam Isolation Valve Leakage Control System	5-20
5.4.6 Reactor Core Isolation Cooling System	5-20
5.4.7 Residual Heat Removal System	5-22
5.4.8 Reactor Water Cleanup System	5-24
6 ENGINEERED SAFETY FEATURES	6-1
6.1 Engineered Safety Features Materials	6-1
6.1.1 Metallic Materials	6-1
6.1.2 Protective Coating Systems (Paints) and Organic Material	6-2
6.2 Containment Systems	6-3
6.2.1 Primary Containment Functional Design	6-3
6.2.2 Containment Heat Removal System	6-19
6.2.3 Secondary Containment Functional Design	6-20
6.2.4 Containment Isolation System	6-22
6.2.5 Combustible Gas Control in Containment	6-29
6.2.6 Containment Leakage Testing	6-33
6.2.7 Fracture Prevention of Containment Pressure Boundary	6-38
6.2.8 Severe Accident Considerations	6-38
6.3 Emergency Core Cooling System	6-38
6.3.1 System Description	6-39
6.3.2 Evaluation of Single Failures	6-40
6.3.3 Qualification of Emergency Core Cooling System	6-40
6.3.4 Testing	6-42
6.3.5 Performance Evaluation	6-43
6.3.6 Certified Design Material	6-45
6.3.7 Conclusions	6-45
6.4 Control Room Habitability Systems	6-46
6.5 Fission Product Removal and Control Systems	6-48
6.5.1 Engineered Safety Features Atmosphere Cleanup Systems	6-48
6.5.2 Containment Spray System	6-50
6.5.3 Fission Product Control Systems and Structures	6-50
6.6 Inservice Inspection of Class 2 and 3 Components	6-51
6.6.1 Requirements of 10 CFR 50.55a, "Codes and Standards"	6-51
6.6.2 Application of the Codes and Standards Rule to Standard Design Certification	6-52
6.6.3 Staff Evaluation	6-52
6.7 High-Pressure Nitrogen Gas Supply System	6-53
6.8 Main Steam Isolation Valve Leakage Control System	6-53
7 INSTRUMENTATION AND CONTROLS	7-1
7.1 Introduction	7-1
7.1.1 Acceptance Criteria	7-1
7.1.2 Method of Review	7-1
7.1.3 General Findings	7-1
7.1.4 Specific Findings	7-14
7.2 Reactor Protection System	7-15
7.2.1 General System Description	7-15
7.2.2 Safety System Logic and Control and Specific Subsystem Descrip- tions	7-21
7.2.3 Indication of Bypassed and Inoperable Status	7-30
7.2.4 Alarms and Annunciators	7-30

7.2.5	Support Systems	7-31
7.2.6	Defense-in-Depth Analysis	7-32
7.2.7	Setpoints	7-41
7.2.8	Hardware and Software Qualification	7-42
7.2.9	RPS Findings and Conclusions	7-45
7.3	Engineered Safety Features Systems	7-46
7.3.1	System Description	7-46
7.3.2	Safety System Logic and Control and Specific Subsystem Descriptions	7-49
7.3.3	Findings and Conclusions	7-50
7.4	Systems Required for Safe Shutdown	7-51
7.4.1	System Description	7-51
7.4.2	Specific Findings	7-53
7.4.3	Evaluation Findings	7-53
7.5	Safety-Related Display Instrumentation and Information Systems Important to Safety	7-54
7.5.1	System Description	7-54
7.5.2	Findings and Conclusions	7-55
7.6	All Other Instrumentation Systems Required for Safety	7-56
7.6.1	System Description	7-56
7.6.2	Specific Findings and Evaluation	7-58
7.7	Control Systems	7-58
7.7.1	System Description	7-58
7.7.2	Specific Findings and Evaluations	7-61
7.8	COL License Information	7-62
7.9	Appendix 7A - Design Response to Appendix B of ABWR Licensing Review Bases	7-62
7.10	Unresolved Safety Issues, Generic Safety Issues, and Operating Experience	7-63
7.10.1	Unresolved Safety Issues and Generic Safety Issues	7-63
7.10.2	Operating Experience	7-63
7.11	Technical Specifications	7-65
8	ELECTRIC POWER SYSTEMS	8-1
8.1	Introduction	8-1
8.2	Offsite Electric Power System	8-1
8.2.1	Preferred Offsite Circuits Outside the ABWR Scope of Design	8-1
8.2.2	Preferred Offsite Circuits Within the ABWR Scope of Design	8-5
8.2.3	Independence Between Offsite and Onsite Power Systems, Equipment, and Components	8-12
8.2.4	Power Supply for the Reactor Internal Pumps	8-21
8.3	Onsite Class 1E Power System	8-21
8.3.1	Compliance with General Design Criteria	8-24
8.3.2	Physical Independence	8-26
8.3.3	Protection of Electrical Systems, Equipment, and Components	8-35
8.3.4	Electrical Independence	8-47
8.3.5	Lighting Systems	8-53
8.3.6	Design Control	8-55
8.3.7	Testing and Surveillance	8-56
8.3.8	Capacity and Capability	8-57
8.3.9	Station Blackout	8-64
9	AUXILIARY SYSTEMS	9-1
9.1	Fuel Storage and Handling	9-1
9.1.2	Spent Fuel Storage	9-2
9.1.3	Spent Fuel Pool Cooling and Cleanup System	9-3

	Page
9.1.4 Light Load-Handling System (Related to Refueling)	9-9
9.1.5 Overhead Heavy Load-Handling System	9-10
9.2 Water Systems	9-15
9.2.1 Station Service Water System	9-15
9.2.2 Reactor Auxiliary Cooling Water System	9-15
9.2.3 Demineralized Water Makeup System	9-16
9.2.4 Potable and Sanitary Water System	9-16
9.2.5 Ultimate Heat Sink	9-16
9.2.6 Condensate Storage Facility	9-18
9.2.7 Chilled Water Systems	9-18
9.2.8 Makeup Water System (Preparation)	9-18
9.2.9 Makeup Water System (Condensate)	9-18
9.2.10 Makeup Water System (Purified) Distribution System	9-20
9.2.11 Reactor Building Cooling Water System	9-21
9.2.12 HVAC Normal Cooling Water System	9-24
9.2.13 HVAC Emergency Cooling Water System	9-25
9.2.14 Turbine Building Cooling Water System	9-28
9.2.15 Reactor Service Water	9-29
9.2.16 Turbine Service Water System	9-31
9.3 Process Auxiliaries	9-32
9.3.1 Compressed Air Systems	9-32
9.3.2 Process and Post-Accident Sampling Systems	9-36
9.3.3 Non-Radioactive Drain System	9-39
9.3.4 Chemical and Volume Control System	9-40
9.3.5 Standby Liquid Control System	9-40
9.3.6 Instrument Air System	9-42
9.3.7 Service Air System	9-42
9.3.8 Radioactive Drain Transfer System	9-42
9.3.9 Hydrogen Water Chemistry System	9-43
9.3.10 Oxygen Injection System	9-44
9.3.11 Zinc Injection System	9-44
9.4 Heating, Ventilation, and Air Conditioning Systems	9-44
9.4.1 Control Building	9-44
9.4.2 Spent Fuel Pool Area Ventilation System	9-48
9.4.3 Auxiliary Area Ventilation System	9-48
9.4.4 Turbine Island HVAC System	9-48
9.4.5 Reactor Building Ventilation System	9-49
9.4.6 Radwaste Building HVAC System	9-54
9.4.7 Diesel Generator Area Ventilation System	9-55
9.4.8 Service Building Ventilation System	9-55
9.4.9 Drywell Cooling System	9-56
9.5 Other Auxiliary Systems	9-57
9.5.1 Fire Protection System	9-57
9.5.2 Communications Systems	9-67
9.5.3 Lighting Systems	9-67
9.5.4 Diesel Generator Fuel Oil Storage and Transfer System	9-67
9.5.5 Diesel Generator Cooling Water System	9-70
9.5.6 Diesel Generator Starting Air System	9-72
9.5.7 Diesel Generator Lubrication System	9-73
9.5.8 Diesel Generator Combustion Air Intake and Exhaust System	9-75
10 STEAM AND POWER CONVERSION SYSTEM	10-1
10.1 Summary Description	10-1

	Page
10.2 Turbine Generator	10-1
10.2.1 Turbine Generator System	10-1
10.2.2 Turbine Rotor Integrity	10-2
10.3 Main Steam Supply System	10-4
10.3.1 System Description and Operation	10-4
10.3.2 Steam and Feedwater System Materials	10-7
10.4 Other Features	10-7
10.4.1 Main Condenser	10-7
10.4.2 Main Condenser Evacuation System	10-8
10.4.3 Turbine Gland Sealing System	10-9
10.4.4 Turbine Bypass System	10-10
10.4.5 Circulating Water System	10-11
10.4.6 Condensate Purification System	10-12
10.4.7 Condensate and Feedwater System	10-13
10.4.8 Power Cycle Heat Sink	10-14
11 RADIOACTIVE WASTE MANAGEMENT	11-1
11.1 Source Terms	11-2
11.2 Liquid Waste Management System	11-2
11.2.1 System Description and Review Discussion	11-2
11.2.2 Conclusion	11-7
11.3 Gaseous Waste Management System	11-9
11.3.1 System Description and Review Discussion	11-9
11.3.2 Conclusion	11-10
11.4 Solid Waste Management System	11-12
11.4.1 System Description and Review Discussion	11-12
11.4.2 Conclusion	11-13
11.5 Process and Effluent Radiological Monitoring and Sampling Systems	11-15
11.5.1 System Description and Review Discussion	11-15
11.5.2 Conclusion	11-16
12 RADIATION PROTECTION	12-1
12.1 Ensuring That Occupational Radiation Doses Are As Low As Is Reasonably Achievable	12-1
12.1.1 Policy Considerations	12-1
12.1.2 Design Considerations	12-1
12.1.3 Operational Considerations	12-2
12.1.4 COL License Information	12-3
12.2 Radiation Sources	12-3
12.2.1 Contained Sources and Airborne Radioactive Material Sources	12-3
12.2.2 Certified Design Material	12-4
12.2.3 COL License Information	12-4
12.3 Radiation Protection Design	12-4
12.3.1 Facility Design Features	12-5
12.3.2 Shielding	12-6
12.3.3 Ventilation	12-6
12.3.4 Area Radiation and Airborne Radioactive Monitoring Instrumentation	12-7
12.3.5 Certified Design Material	12-8
12.3.6 10 CFR 50.34(f): TMI-Related Items	12-9
12.4 Dose Assessment	12-9
12.5 Organization	12-9
12.5.1 10 CFR 50.34(f): TMI-Related Items	12-10

	Page
13 CONDUCT OF OPERATIONS	13-1
13.1 Organizational Structure of Applicants	13-1
13.2 Training	13-1
13.3 Emergency Planning	13-2
13.4 Review and Audit	13-3
13.5 Plant Procedures	13-3
13.6 Physical Security	13-4
13.6.1 Preliminary Planning	13-4
13.6.2 Security Plan	13-4
13.6.3 Control of Access to Areas Containing Vital Equipment	13-5
14 INITIAL TEST PROGRAM	14-1
14.1 Preliminary Safety Analysis Reports Information	14-1
14.2 Initial Plant Test Programs	14-1
14.2.1 Summary of Test Program and Objectives	14-2
14.2.2 Organization and Staffing	14-3
14.2.3 Test Procedures	14-3
14.2.4 Conduct of Test Program	14-4
14.2.5 Review, Evaluation, and Approval of Test Results	14-5
14.2.6 Test Records	14-5
14.2.7 Conformance of Test Program With Regulatory Guides	14-5
14.2.8 Utilization of Reactor Operating and Testing Experience in the Development of Test Program	14-6
14.2.9 Trial Use of Plant Operating and Emergency Procedures	14-6
14.2.10 Initial Fuel Loading and Initial Criticality	14-6
14.2.11 Test Program Schedule	14-7
14.2.12 Individual Test Descriptions	14-9
14.2.13 COL License Information - Initial Test Program	14-19
14.3 Certified Design Material	14-20
14.3.1 Introduction to CDM	14-23
14.3.2 Certified Design Material for Structures and Systems	14-24
14.3.3 Additional Certified Design Material	14-47
14.3.4 Interface Requirements	14-55
14.3.5 Site Parameters	14-56
14.3.6 Summary	14-56
14.3.7 DFSEI Issues	14-57
15 TRANSIENT AND ACCIDENT ANALYSIS	15-1
15.1 Anticipated Operational Occurrences	15-1
15.2 Trip of All Reactor Internal Pumps and Pressure Regulator Down-Scale Failure	15-5
15.3 Accidents	15-6
15.4 Radiological Consequences of Accidents	15-7
15.4.1 Control Rod Drop Accidents	15-16
15.4.2 Failure of Small Lines Carrying Primary Coolant Outside Containment	15-16
15.4.3 Main Steamline Failure Outside Containment	15-17
15.4.4 Loss-of-Coolant Accidents	15-17
15.4.5 Fuel Handling and Fuel Cask Drop Accidents	15-24
15.4.6 Postulated Radioactive Releases Resulting from Liquid Tank Failures	15-25
15.4.7 Reactor Water Cleanup System Pipe Break Accident	15-25

	Page
15.5 Anticipated Transient Without Scram	15-26
15.5.1 Design Features	15-26
15.5.2 Analysis	15-26
16 TECHNICAL SPECIFICATIONS	16-1
16.1 Introduction	16-1
16.2 Evaluation	16-1
16.3 Conclusion	16-2
17 QUALITY ASSURANCE	17-1
17.1 Quality Assurance During the Design Phase	17-1
17.1.1 General	17-1
17.1.2 Organization	17-1
17.1.3 Quality Assurance Program	17-4
17.2 Quality Assurance During the Operations Phase	17-6
17.3 Reliability Assurance Program	17-6
17.3.1 General	17-9
17.3.2 Scope	17-9
17.3.3 Purpose	17-9
17.3.4 Objective	17-10
17.3.5 GE Organization for D-RAP	17-10
17.3.6 SSC Identification and Prioritization	17-10
17.3.7 Design Considerations	17-11
17.3.8 Defining Failure Modes	17-11
17.3.9 Operational Reliability Assurance Activities	17-11
17.3.10 COL Applicant's Reliability Assurance Process	17-12
17.3.11 D-RAP Implementation	17-12
17.3.12 Conclusion	17-12
18 HUMAN FACTORS ENGINEERING	18-1
18.1 Methodology	18-2
18.1.1 Preliminary Review and Draft Safety Evaluation Report Issues	18-2
18.1.2 Final Standard Safety Analysis Report Review	18-4
18.1.3 Development of Review Criteria	18-5
18.2 Design Goals	18-8
18.2.1 General Discussion in the Standard Safety Analysis Report	18-8
18.2.2 Analysis	18-8
18.2.3 Finding	18-9
18.3 Main Control Room Standard Design Features	18-9
18.3.1 General Discussion in the Standard Safety Analysis Report	18-9
18.3.2 Analysis	18-10
18.3.3 Finding	18-17
18.4 Inventory of Controls, Displays, and Alarms	18-17
18.4.1 General Discussion in the Standard Safety Analysis Report	18-17
18.4.2 Analysis	18-18
18.4.3 Findings	18-20
18.5 Remote Shutdown System	18-20
18.5.1 General Discussion in the Standard Safety Analysis Report	18-20
18.5.2 Analysis	18-20
18.5.3 Finding	18-21
18.6 Local Valve Position Indication	18-21

	Page
18.7	Unresolved and Generic Safety Issues 18-22
18.7.1	General Discussion in the Standard Safety Analysis Report 18-22
18.7.2	Analysis 18-22
18.7.3	Finding 18-24
18.8	Emergency Procedure Guidelines 18-24
18.8.1	Containment Temperature 18-24
18.8.2	Venting 18-24
18.8.3	Drywell Spray Initiation Limit 18-25
18.8.4	Heat Capacity Temperature Limit 18-25
18.8.5	Primary Containment Flooding 18-27
18.8.6	ATWS Stability Strategy 18-27
18.9	Design and Implementation Process 18-27
18.9.1	General Discussion in the Standard Safety Analysis Report 18-28
18.9.2	Analysis 18-28
18.9.3	Design Description Tier 1 ITAAC and DAC Review 18-35
18.10	Conclusion 18-35
19	SEVERE ACCIDENTS 19-1
19.1	Probabilistic Safety Assessment 19-1
19.1.1	Executive Summary 19-1
19.1.2	Introduction 19-6
19.1.3	Advanced Boiling Water Reactor Probabilistic Risk Assessment- Based Safety Insights 19-7
19.1.4	Evaluation of the Quality of the ABWR Probabilistic Risk Assessment 19-41
19.1.5	Open Item Closure 19-42
19.2	Severe Accident Performance 19-42
19.2.1	Introduction 19-42
19.2.2	Deterministic Assessment of Severe Accident Prevention 19-42
19.2.3	Deterministic Assessment of Severe Accident Mitigation 19-45
19.2.4	Containment Performance 19-68
19.2.5	Accident Management 19-69
19.2.6	Capacity of the ABWR Primary Containment Vessel 19-69
19.3	Shutdown Risk 19-74
19.3.1	Introduction 19-74
19.3.2	Evaluation Scope 19-75
19.3.3	ABWR Design Features Minimizing Shutdown Risk 19-76
19.3.4	ABWR Reactor Instrumentation 19-83
19.3.5	Flooding and Fire Protection 19-84
19.3.6	Shutdown Risk Insights 19-86
19.3.7	Technical Findings in NUREG-1449 19-86
19.3.8	Summary 19-88
19.3.9	Conclusion 19-88
20	GENERIC ISSUES 20-1
20.1	Task Action Plan Items 20-17
20.1.1	Issue A-1: Water Hammer (former USI) 20-17
20.1.2	Issue A-7: Mark I Long-Term Program (former USI) 20-18
20.1.3	Issue A-8: Mark II Containment Pool Dynamic Loads - Long-Term Program 20-18
20.1.4	Issue A-9: Anticipated Transient Without Scram (ATWS) (former USI) 20-19
20.1.5	Issue A-10: BWR Feedwater Nozzle Cracking (former USI) 20-19

	Page
20.1.6 Issue A-13: Snubber Operability Assurance	20-20
20.1.7 Issue A-17: Systems Interaction (former USI)	20-20
20.1.8 Issue A-24: Qualification of Class 1E Safety-Related Equipment (former USI)	20-21
20.1.9 Issue A-25: Non-Safety Loads on Class 1E Power Sources	20-22
20.1.10 Issue A-29: Nuclear Power Plant Design for the Reduction of Vulnerability to Industrial Sabotage	20-22
20.1.11 Issue A-31: RHR Shutdown Requirements (former USI)	20-23
20.1.12 Issue A-35: Adequacy of Offsite Power Systems	20-24
20.1.13 Issue A-36: Control of Heavy Loads Near Spent Fuel (former USI)	20-25
20.1.14 Issue A-39: Determination of Safety Relief Valve Pool Dynamic Loads and Temperature Limits (former USI)	20-25
20.1.15 Issue A-40: Seismic Design Criteria - Short-Term Program (former USI)	20-26
20.1.16 Issue A-42: Pipe Cracks in BWR (former USI)	20-26
20.1.17 Issue A-44: Station Blackout (former USI)	20-27
20.1.18 Issue A-47: Safety Implications of Control Systems (former USI)	20-28
20.1.19 Issue A-48: Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment	20-28
20.1.20 Issue B-5: Ductility of Two-Way Slabs and Shells and Buckling Behavior of Steel Containments	20-29
20.1.21 Issue B-10: Behavior of BWR Mark III Containments	20-30
20.1.22 Issue B-17: Criteria for Safety-Related Operator Actions	20-30
20.1.23 Issue B-29: Effectiveness of Ultimate Heat Sinks	20-31
20.1.24 Issue B-32: Ice Effects on Safety-Related Water Supplies	20-31
20.1.25 Issue B-36: Develop Design, Testing, and Maintenance Criteria for Atmosphere Cleanup System Air Filtration and Adsorption Units for Engineered Safety Features Systems and Normal Ventilation Systems	20-32
20.1.26 Issue B-55: Improved Reliability of Target Rock Safety-Relief Valves	20-32
20.1.27 Issue B-56: Diesel Reliability	20-32
20.1.28 Issue B-61: Allowable ECCS Equipment Outage Periods	20-34
20.1.29 Issue B-63: Isolation of Low-Pressure Systems Connected to the Reactor Coolant Pressure Boundary	20-35
20.1.30 Issue B-66: Control Room Infiltration Measurements	20-35
20.1.31 Issue C-1: Assurance of Continuous Long-Term Capability of Hermetic Seals on Instrumentation and Electrical Equipment	20-36
20.1.32 Issue C-8: Main Steam Line Leakage Control Systems	20-36
20.1.33 Issue C-10: Effective Operation of Containment Sprays in a LOCA	20-37
20.1.34 Issue C-17: Interim Acceptance Criteria for Solidification Agents for Radioactive Solid Wastes	20-37
20.1.35 Other Issues	20-38
20.2 New Generic Issues	20-38
20.2.1 Issue 15: Radiation Effects on Reactor Vessel Supports	20-38
20.2.2 Issue 23: Reactor Coolant Pump Seal Failures	20-38
20.2.3 Issue 25: Automatic Air Header Dump on BWR Scram System	20-39
20.2.4 Issue 29: Bolting Degradation or Failure in Nuclear Power Plants	20-39
20.2.5 Issue 40: Safety Concerns Associated With Pipe Breaks in the BWR Scram System	20-40
20.2.6 Issue 45: Inoperability of Instrumentation Due to Extreme Cold Weather	20-40

	Page
20.2.7 Issue 51: Proposed Requirements for Improving the Reliability of Open Cycle Service Water Systems	20-41
20.2.8 Issue 57: Effects of Fire Protection System Actuation on Safety-Related Equipment	20-41
20.2.9 Issue 67.3.3: Steam Generator Staff Actions - Improved Accident Monitoring	20-42
20.2.10 Issue 75: Generic Implications of ATWS Events at the Salem Nuclear Plant	20-42
20.2.11 Issue 78: Monitoring of Fatigue Transient Limits for Reactor Coolant System	20-43
20.2.12 Issue 82: Beyond Design-Basis Accidents in Spent Fuel Pools	20-43
20.2.13 Issue 83: Control Room Habitability	20-44
20.2.14 Issue 86: Long-Range Plan for Dealing with Stress Corrosion Cracking in BWR Piping	20-45
20.2.15 Issue 87: Failure of High Pressure Coolant Injection Steam Line Without Isolation	20-45
20.2.16 Issue 89: Stiff Pipe Clamps	20-46
20.2.17 Issue 103: Design for Probable Maximum Precipitation	20-47
20.2.18 Reserved.	20-47
20.2.19 Issue 105: Interfacing Systems LOCA at LWRs	20-47
20.2.20 Issue 106: Piping and Use of Highly Combustible Gases in Vital Areas	20-49
20.2.21 Issue 113: Dynamic Qualification Testing of Large Bore Hydraulic Snubbers	20-50
20.2.22 Issue 118: Tendon Anchorage Failure	20-51
20.2.23 Issue 120: On-Line Testability of Protection Systems	20-51
20.2.24 Issue 121: Hydrogen Control for Large, Dry PWR Containments	20-52
20.2.25 Issue 124: Auxiliary Feedwater System Reliability	20-52
20.2.26 Issue 128: Electrical Power Reliability	20-52
20.2.27 Issue 130: Essential Service Water Pump Failures at Multiplant Sites	20-54
20.2.28 Issue 142: Leakage Through Electrical Isolators in Instrumentation Circuits	20-55
20.2.29 Issue 143: Availability of Chilled Water Systems	20-56
20.2.30 Issue 145: Actions to Reduce Common Cause Failures	20-57
20.2.31 Issue 151: Reliability of Recirculation Pump Trip During an ATWS	20-57
20.2.32 Issue 153: Loss of Essential Service Water in LWRs	20-58
20.2.33 Issue 155.1: More Realistic Source Term Assumptions	20-59
20.3 Human Factors Issues	20-60
20.3.1 Issue HF 1.1: Staffing and Qualifications - Shift Staffing	20-60
20.3.2 Issue HF 4.4: Procedures - Guidelines for Upgrading Other Procedures	20-60
20.3.3 Issue HF 5.1: Man-Machine Interface - Local Control Stations	20-60
20.3.4 Issue HF 5.2: Man-Machine Interface - Review Criteria for Human Factors Aspects of Advanced Controls and Instrumentation	20-60
20.4 TMI Action Plan Items	20-61
20.4.1 Issue I.A.1.1: Operating Personnel -- Operating Personnel and Staffing - Shift Technical Advisor	20-61
20.4.2 Issue I.A.1.2: Operating Personnel -- Operating Personnel and Staffing - Shift Supervisor Administrative Duties	20-61
20.4.3 Issue I.A.1.3: Operating Personnel -- Operating Personnel and Staffing - Shift Manning	20-61
20.4.4 Issue I.A.1.4: Operating Personnel -- Operating Personnel and Staffing - Long-Term Upgrading	20-62

20.4.5	Issue I.A.2.1: Operating Personnel -- Training and Qualification of Operating Personnel - Immediate Upgrading of Operator and Senior Operator Training and Qualifications	20-62
20.4.6	Issue I.A.2.3: Operating Personnel -- Training and Qualifications of Operating Personnel - Administration of Training Programs	20-62
20.4.7	Issue I.A.2.6(1): Operating Personnel -- Training and Qualifications of Operating Personnel - Long-Term Upgrading of Training and Qualifications; Revise Regulatory Guide 1.8	20-62
20.4.8	Issue I.A.3.1: Operating Personnel -- Licensing and Requalification of Operating Personnel - Revise Scope of Criteria for Licensing Examinations	20-63
20.4.9	Issue I.A.4.1(2): Operating Personnel -- Simulator Use and Development - Initial Simulator Improvement; Interim Changes in Training Simulators	20-63
20.4.10	Issue I.A.4.2: Operating Personnel -- Simulator Use and Development - Long-Term Training Upgrade	20-63
20.4.11	Issue I.C.1: Operating Procedures - Short-Term Accident Analysis and Procedures Revision	20-63
20.4.12	Issue I.C.2: Operating Procedures - Shift Relief and Turnover Procedures	20-63
20.4.13	Issue I.C.3: Operating Procedures - Shift Supervisor Responsibilities	20-64
20.4.14	Issue I.C.4: Operating Procedures - Control Room Access	20-64
20.4.15	Issue I.C.5: Operating Procedures - Procedures for Feedback of Operating Experience to Plant Staff	20-64
20.4.16	Issue I.C.6: Operating Procedures - Procedures for Verification of Correct Performance of Operating Activities	20-64
20.4.17	Issue I.C.7: Operating Procedures - NSSS Vendor Review of Procedures	20-65
20.4.18	Issue I.C.8: Operating Procedures - Pilot Monitoring of Selected Emergency Procedures for NTOL Applicants	20-65
20.4.19	Issue I.C.9: Operating Procedures - Long-Term Program Plan Procedures for Upgrading of Procedures	20-65
20.4.20	Issue I.D.1: Control Room Design - Control Room Design Reviews	20-65
20.4.21	Issue I.D.2: Control Room Design - Plant Safety Parameter Display Console	20-65
20.4.22	Issue I.D.3: Control Room Design - Safety System Status Monitoring	20-65
20.4.23	Issue I.D.5(2): Control Room Design - Plant Status and Post-Accident Monitoring	20-65
20.4.24	Issue I.D.5(3): Control Room Design - On-Line Reactor Surveillance System	20-66
20.4.25	Issue I.F.1: Quality Assurance (QA) - Expand QA List	20-66
20.4.26	Issue I.F.2: Quality Assurance - Develop More Detailed QA Criteria	20-66
20.4.27	Issue I.G.1: Preoperational and Low-Power Testing - Training Requirements	20-66
20.4.28	Issue I.G.2: Preoperational and Low-Power Testing - Scope of Test Program	20-66
20.4.29	Issue II.B.1: Consideration of Degraded or Melted Cores in Safety Review - Reactor Coolant System Vents	20-67
20.4.30	Issue II.B.2: Consideration of Degraded or Melted Cores in Safety Review - Plant Shielding to Provide Access to Vital Areas and Protect Safety Equipment for Post-Accident Operation	20-67

20.4.31	Issue II.B.3: Consideration of Degraded or Melted Cores in Safety Review - Post-Accident Sampling	20-67
20.4.32	Issue II.B.4: Consideration of Degraded or Melted Cores in Safety Review - Training for Mitigating Core Damage	20-67
20.4.33	Issue II.B.8: Consideration of Degraded or Melted Cores in Safety Review - Rulemaking Proceeding on Degraded Core Accidents	20-67
20.4.34	Issue II.D.1: Reactor Coolant System Relief and Safety Valves - Testing Requirements	20-67
20.4.35	Issue II.D.3: Reactor Coolant System Relief and Safety Valves - Relief and Safety Valve Position Indication	20-67
20.4.36	Issue II.E.1.1: System Design -- Auxiliary Feedwater System - Auxiliary Feedwater System Evaluation	20-67
20.4.37	Issue II.E.1.2: System Design -- Auxiliary Feedwater System - Auxiliary Feedwater System Automatic Initiation and Flow Indication	20-67
20.4.38	Issue II.E.1.3: System Design -- Auxiliary Feedwater System - Update the Standard Review Plan and Develop Regulatory Guidance	20-67
20.4.39	Issue II.E.3.1: System Design -- Decay Heat Removal - Reliability of Power Supplies for Natural Circulation	20-68
20.4.40	Issue II.E.4.1: System Design -- Containment Design - Dedicated Penetrations	20-68
20.4.41	Issue II.E.4.2: System Design -- Containment Design - Isolation Dependability	20-68
20.4.42	Issue II.E.4.4: System Design -- Containment Design - Purging	20-68
20.4.43	Issue II.E.5.1: System Design -- Design Sensitivity of Babcock & Wilcox (B&W) Reactors - Design Evaluation	20-68
20.4.44	Issue II.E.6.1: System Design -- In-Situ Testing of Valves - Test Adequacy Study	20-68
20.4.45	Reserved.	20-68
20.4.46	Issue II.F.1: Instrumentation and Controls - Additional Accident Monitoring Instrumentation	20-68
20.4.47	Issue II.F.2: Instrumentation and Controls - Identification of and Recovery from Conditions Leading to Inadequate Core Cooling	20-68
20.4.48	Issue II.F.3: Instrumentation and Controls - Instruments for Monitoring Accident Conditions	20-69
20.4.49	Issue II.G.1: Electrical Power - Power Supplies for Pressurizer Relief Valves, Block Valves, and Level Indicators	20-69
20.4.50	Issue II.J.3.1: General Implications of TMI for Design and Construction Activities -- Management for Design and Construction - Organization and Staffing to Oversee Design and Construction	20-69
20.4.51	Issue II.J.4.1: General Implications of TMI for Design and Construction Activities - Revise Deficiency Reporting Requirements	20-69
20.4.52	Issue II.K.1(5): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- IE Bulletins - Safety-Related Valve Position Description	20-69
20.4.53	Issue II.K.1(10): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- IE Bulletins - Review and Modify Procedures for Removing Safety-Related Systems from Service	20-69
20.4.54	Issue II.K.1(13): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- IE Bulletins - Proposed Technical Specification Changes Reflecting Implementation of All Bulletin Items	20-70

20.4.55	Issue II.K.1.(22): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- IE Bulletins - Describe Automatic and Manual Actions for Proper Functioning of Auxiliary Heat Removal Systems When Feedwater System Not Operable	20-70
20.4.56	Issue II.K.1(23): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- IE Bulletins - Describe Uses and Types of RV Level Indication for Automatic and Manual Initiation Safety Systems	20-70
20.4.57	Issue II.K.2(9): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Commission Orders on B&W Plants - Analysis and Upgrading of Integrated Control System	20-71
20.4.58	Issue II.K.2(10): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Commission Orders on B&W Plants - Hard-Wired Safety-Grade Anticipatory Reactor Trips	20-71
20.4.59	Issue II.K.2(16): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Commission Orders on B&W Plants - Impact of RCP Seal Damage Following SBLOCA With Loss of Offsite Power	20-71
20.4.60	Issue II.K.3(2): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Report on Overall Safety Effect of Power-Operated Relief Valve (PORV) Isolation	20-71
20.4.61	Issue II.K.3(3): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Report Safety and Relief Valve Failures Promptly and Challenges Annually	20-71
20.4.62	Issue II.K.3(11): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Control Use of PORV Supplied by Control Components, Inc., Until Further Review Complete	20-71
20.4.63	Issue II.K.3(13): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Separation of HPCI and RCIC System Initiation Levels	20-72
20.4.64	Issue II.K.3(15): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Modify Break Detection Logic to Prevent Spurious Isolation of HPCI and RCIC Systems	20-72
20.4.65	Issue II.K.3(16): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Reduction of Challenges and Failures of Relief Valves; Feasibility Study and System Modification	20-72
20.4.66	Issue II.K.3(17): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Report on Outage of ECC Systems; Licensee Report and Technical Specification Changes	20-72

20.4.67 Issue II.K.3(18): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Modification of ADS Logic; Feasibility Study and Modification for Increased Diversity for Some Event Sequences	20-73
20.4.68 Issue II.K.3(21): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Restart of Core Spray and LPCI Systems on Low Level; Design and Modification	20-73
20.4.69 Issue II.K.3(22): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Automatic Switchover of RCIC System Suction; Verify Procedures and Modify Design	20-73
20.4.70 Issue II.K.3(23): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Central Water Level Recording	20-73
20.4.71 Issue II.K.3(24): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Confirm Adequacy of Space Cooling for HPCI and RCIC Systems	20-73
20.4.72 Issue II.K.3(25): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Effect of Loss of ac Power on Pump Seals	20-74
20.4.73 Issue II.K.3(27): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Provide Common Reference Level for Vessel Level Instrumentation	20-74
20.4.74 Issue II.K.3(28): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Study and Verify Qualification of Accumulators on ADS Valves	20-74
20.4.75 Issue II.K.3(30): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Revised SBLOCA Methods to Show Compliance with 10 CFR Part 50, Appendix A	20-74
20.4.76 Issue II.K.3(31): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Plant-Specific Calculations to Show Compliance with 10 CFR 50.46	20-74
20.4.77 Issue II.K.3(44): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Evaluation of Anticipated Transients with Single Failure to Verify no Significant Fuel Failure	20-75
20.4.78 Issue II.K.3(45): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Evaluate Depressurization with Other Than Full ADS	20-75

20.4.79	Issue II.K.3(46): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Response to List of Concerns from ACRS Consultant	20-75
20.4.80	Issue III.A.1.1(1): Emergency Preparedness and Radiation Effects -- Improve Licensee Emergency Preparedness - Short Term; Upgrade Emergency Preparedness, Implement Action Plan Requirements for Promptly Improving Licensee Emergency Preparedness	20-75
20.4.81	Issue III.A.1.2: Emergency Preparedness and Radiation Effects -- Improve Licensee Emergency Preparedness - Short Term; Upgrade Emergency Preparedness	20-76
20.4.82	Issue III.A.2.1: Emergency Preparedness and Radiation Effects -- Improve Licensee Emergency Preparedness - Long Term; Amend 10 CFR Part 50 and 10 CFR Part 50, Appendix E	20-76
20.4.83	Issue III.A.2.2: Emergency Preparedness and Radiation Effects -- Improve Licensee Emergency Preparedness - Long Term; Development of Guidance and Criteria	20-76
20.4.84	Issue III.A.3.3: Emergency Preparedness and Radiation Effects -- Improving NRC Emergency Preparedness - Communications	20-76
20.4.85	Issue III.D.1.1: Radiation Protection -- Radiation Source Control - Primary Coolant Sources Outside the Containment Structure	20-77
20.4.86	Issue III.D.3.3: Radiation Protection -- Worker Radiation Protection Improvement - Inplant Radiation Monitoring	20-77
20.4.87	Issue III.D.3.4: Radiation Protection -- Worker Radiation Protection Improvement - Control Room Habitability	20-77
20.5	10 CFR 50.34(f), Additional TMI Requirements	20-77
20.5.1	10 CFR 50.34(f)(1)(i): Consideration of Degraded or Melted Cores in Safety Review - Rulemaking Proceeding on Degraded Core Accidents (TMI Item II.B.8), "Design Alternatives from PRA"	20-77
20.5.2	10 CFR 50.34(f)(1)(ii): System Design -- Auxiliary Feedwater System - Auxiliary Feedwater System Evaluation (TMI Item II.E.1.1)	20-89
20.5.3	10 CFR 50.34(f)(1)(iii): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Commission Orders on B&W Plants - Impact of RCP Seal Damage Following SBLOCA With Loss of Offsite Power (TMI Item II.K.2(16)) and Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Effect of Loss of ac Power on Pump Seals (TMI Item II.K.3(25))	20-89
20.5.4	10 CFR 50.34(f)(1)(iv): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Report on Overall Safety Effect of PORV Isolation (TMI Item II.K.3(2))	20-89
20.5.5	10 CFR 50.34(f)(1)(v): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Separation of HPCI and RCIC System Initiation Levels (TMI Item II.K.3(13))	20-90
20.5.6	10 CFR 50.34(f)(1)(vi): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Reduction of Challenges and Failures of Relief Valves; Feasibility Study and System Modification (TMI Item II.K.3(16))	20-90

20.5.7	10 CFR 50.34(f)(1)(vii): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Modification of ADS Logic; Feasibility Study and Modification for Increased Diversity for Some Event Sequences (TMI Item II.K.3(18))	20-91
20.5.8	10 CFR 50.34(f)(1)(viii): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Restart of Core Spray and LPCI Systems on Low Level; Design and Modification (TMI Item II.K.3(21))	20-91
20.5.9	10 CFR 50.34(f)(1)(ix): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Confirm Adequacy of Space Cooling for HPCI and RCIC Systems (TMI Item II.K.3(24))	20-92
20.5.10	10 CFR 50.34(f)(1)(x): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Study and Verify Qualification of Accumulators on ADS Valves (TMI Item II.K.3(28))	20-92
20.5.11	10 CFR 50.34(f)(1)(xi): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Evaluate Depressurization with Other Than Full ADS (TMI Item II.K.3(45))	20-94
20.5.12	10 CFR 50.34(f)(1)(xii): Evaluation of Alternative Hydrogen Control Systems	20-94
20.5.13	10 CFR 50.34(f)(2)(i): Operating Personnel -- Simulator Use and Development - Long-Term Training Upgrade (TMI Item I.A.4.2)	20-95
20.5.14	10 CFR 50.34(f)(2)(ii): Operating Procedures -Long-Term Program Plan Procedures for Upgrading of Procedures (TMI Item I.C.9)	20-95
20.5.15	10 CFR 50.34(f)(2)(iii): Control Room Design -Control Room Design Reviews (TMI Item I.D.1)	20-95
20.5.16	10 CFR 50.34(f)(2)(iv): Control Room Design -Plant Safety Parameter Display Console (TMI Item I.D.2)	20-95
20.5.17	10 CFR 50.34(f)(2)(v): Control Room Design -Safety System Status Monitoring (TMI Item I.D.3)	20-96
20.5.18	10 CFR 50.34(f)(2)(vi): Consideration of Degraded or Melted Cores in Safety Review - Reactor Coolant System Vents (TMI Item II.B.1)	20-96
20.5.19	10 CFR 50.34(f)(2)(vii): Consideration of Degraded or Melted Cores in Safety Review - Plant Shielding to Provide Access to Vital Areas and Protect Safety Equipment for Post-Accident Operation (TMI Item II.B.2)	20-97
20.5.20	10 CFR 50.34(f)(2)(viii): Consideration of Degraded or Melted Cores in Safety Review - Post-Accident Sampling (TMI Item II.B.3)	20-97
20.5.21	10 CFR 50.34(f)(2)(ix): Consideration of Degraded or Melted Cores in Safety Review - Rulemaking Proceeding on Degraded Core Accidents (TMI Item II.B.8), "Hydrogen Control System"	20-97
20.5.22	10 CFR 50.34(f)(2)(x): Reactor Coolant System Relief and Safety Valves - Testing Requirements (TMI Item II.D.1)	20-98
20.5.23	10 CFR 50.34(f)(2)(xi): Reactor Coolant System Relief and Safety Valves - Relief and Safety Valve Position Indication (TMI Item II.D.3)	20-98

20.5.24	10 CFR 50.34(f)(2)(xii): System Design -- Auxiliary Feedwater System - Auxiliary Feedwater System Automatic Initiation and Flow Indication (TMI Item II.E.1.2)	20-98
20.5.25	10 CFR 50.34(f)(2)(xiii): System Design -- Decay Heat Removal - Reliability of Power Supplies for Natural Circulation (TMI Item II.E.3.1)	20-98
20.5.26	10 CFR 50.34(f)(2)(xiv): System Design -- Containment Design - Isolation Dependability (TMI Item II.E.4.2)	20-98
20.5.27	10 CFR 50.34(f)(2)(xv): System Design -- Containment Design - Purging (TMI Item II.E.4.4)	20-99
20.5.28	10 CFR 50.34(f)(2)(xvi): System Design -- Design Sensitivity of B&W Reactors - Design Evaluation (TMI Item II.E.5.1)	20-100
20.5.29	10 CFR 50.34(f)(2)(xvii): Instrumentation and Controls - Additional Accident Monitoring Instrumentation (TMI Item II.F.1)	20-100
20.5.30	10 CFR 50.34(f)(2)(xviii): Instrumentation and Controls - Identification of and Recovery from Conditions Leading to Inadequate Core Cooling (TMI Item II.F.2)	20-100
20.5.31	10 CFR 50.34(f)(2)(xix): Instrumentation and Controls - Instruments for Monitoring Accident Conditions (TMI Item II.F.3)	20-103
20.5.32	10 CFR 50.34(f)(2)(xx): Electrical Power - Power Supplies for Pressurizer Relief Valves, Block Valves, and Level Indicators (TMI Item II.G.1)	20-103
20.5.33	10 CFR 50.34(f)(2)(xxi): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- IE Bulletins - Describe Automatic and Manual Actions for Proper Functioning of Auxiliary Heat Removal Systems When Feedwater System Not Operable (TMI Item Issue II.K.1.(22))	20-103
20.5.34	10 CFR 50.34(f)(2)(xxii): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Commission Orders on B&W Plants - Analysis and Upgrading of Integrated Control System (TMI Item II.K.2(9))	20-104
20.5.35	10 CFR 50.34(f)(2)(xxiii): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Commission Orders on B&W Plants - Hard-Wired Safety-Grade Anticipatory Reactor Trips (TMI Item II.K.2(10))	20-104
20.5.36	10 CFR 50.34(f)(2)(xxiv): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Central Water Level Recording (TMI Item II.K.3(23))	20-104
20.5.37	10 CFR 50.34(f)(2)(xxv): Emergency Preparedness and Radiation Effects -- Improve Licensee Emergency Preparedness - Short Term; Upgrade Emergency Preparedness (TMI Item III.A.1.2)	20-104
20.5.38	10 CFR 50.34(f)(2)(xxvi): Radiation Protection -- Radiation Source Control - Primary Coolant Sources Outside the Containment Structure (TMI Item III.D.1.1)	20-104
20.5.39	10 CFR 50.34(f)(2)(xxvii): Radiation Protection -- Worker Radiation Protection Improvement -Inplant Radiation Monitoring (TMI Item III.D.3.3)	20-105
20.5.40	10 CFR 50.34(f)(2)(xxviii): Radiation Protection -- Worker Radiation Protection Improvement - Control Room Habitability (TMI Item III.D.3.4)	20-105
20.5.41	10 CFR 50.34(f)(3)(i): Operating Procedures - Procedures for Feed-back of Operating Experience to Plant Staff (TMI Item I.C.5)	20-105
20.5.42	10 CFR 50.34(f)(3)(ii): Quality Assurance (QA) - Expand QA List (TMI Item I.F.1)	20-106

	Page
20.5.43 10 CFR 50.34(f)(3)(iii): Quality Assurance - Develop More Detailed QA Criteria (TMI Item I.F.2)	20-106
20.5.44 10 CFR 50.34(f)(3)(iv): Consideration of Degraded or Melted Cores in Safety Review - Rulemaking Proceeding on Degraded Core Accidents (TMI Item II.B.8), ".91-Meter (3-Foot) Diameter Equivalent Dedicated Containment Penetration"	20-106
20.5.45 10 CFR 50.34(f)(3)(v): Consideration of Degraded or Melted Cores in Safety Review - Rulemaking Proceeding on Degraded Core Accidents (TMI Item II.B.8), "Containment Integrity During an Accident Involving 100-Percent, Fuel Clad Metal-Water Reaction"	20-107
20.5.46 10 CFR 50.34(f)(3)(vi): System Design -- Containment Design - Dedicated Penetrations (TMI Item II.E.4.1)	20-107
20.5.47 10 CFR 50.34(f)(3)(vii): General Implications of TMI for Design and Construction Activities -- Management for Design and Construction - Organization and Staffing to Oversee Design and Construction (TMI Item II.J.3.1)	20-108
20.6 Generic Communications	20-108
21 REPORT OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS	21-1
22 CONCLUSIONS	22-1

VOLUME 1: MAIN REPORT

FIGURES

		Page
2-1	Tornado intensity regions	2-3
7.1-1	Integrated hardware/software development process	7-11
7.1-2	Flow of document through the software life cycle	7-12
7.2-1	Safety system logic and control block diagram	7-16
7.2-2	Safety system logic and control (SSLC) control interface diagram	7-18
7.3-1	Safety system logic and control communications path for engineered safety systems	7-37
7.3-2	Assignment of interfacing safety system logic to safety system logic and control controllers	7-38
17.1	ABWR design organization structure	17-3
19.1-1	ABWR core damage frequency distribution	19-13
19.1-2	Breakdown of ABWR containment release frequency from CET analysis	19-34

TABLES

		Page
2-1	Design-basis tornado characteristics	2-2
2-2	EAB and LPZ Atmospheric dispersion characteristics (Chi/Q)	2-11
3.1	Criteria pertaining to the probability of turbine missile generation for a favorably oriented turbine	3-24
3.2	Margins for straight pipe	3-70
3.3	Reassigned number for tables and figures	3-92
6.1	Single failure evaluation	6-41
6.2	Demonstration of compliance with ECCS criteria	6-43
7.1	SRP Acceptance Criteria	7-3
7.2	Chapter 7 Additional Requirements	7-6
11.1	Principal parameters used in the calculation of gaseous and liquid effluents from ABWR	11-3
11.2	Calculated annual release of radioactive materials in gaseous effluents from single ABWR unit	11-4
11.3	Design capacities of principal components in the liquid and gaseous radwaste treatment systems for ABWR single unit	11-6
15.1	Revised radiological consequences of design-basis accidents	15-9
15.2	Assumptions used to compute rod drop accident doses	15-10
15.3	Assumptions used to evaluate the main steamline and small line break accidents outside the containment	15-11
15.4	Assumptions used to evaluate the loss-of-coolant accident	15-12
15.5	Atmospheric dispersion (χ/Q) values used in accident evaluations	15-13
15.6	Method to evaluate iodine removal in main steamlines and main condenser	15-13
15.7	Iodine releases GBq (curies)	15-14
15.8	Assumptions used in computing fuel handling accident doses	15-14
15.9	Assumptions and estimates of the radiological consequences to control room operators after a loss-of-coolant accident	15-15

17.1	Quality assurance regulatory guide commitments	17-2
18.1	DSER (SECY-91-320) Chapter 18 human factors engineering issues	18-3
18.2	DSER (SECY-91-320) HFE issues to be addressed in the process plan	18-29
19.1-1	ABWR PRA initiating event contributors to CDF (Level 1, internal events)	19-1
19.1-2	Important sequences leading to core damage (internal events)	19-2
19.1-3	Important sequences for external events (seismic, fire, and internal floods)	19-3
19.1-4	Examples of cost-effective PSA-inspired design/procedure modifications to the ABWR design	19-5
19.1-5	SSCs identified by importance measure values	19-15
19.1-6	Sensitivity to test or maintenance (T/M) outages	19-16
19.1-7	Results of GE HCLPF quantification using the MIN/MAX approach (by plant damage class)	19-18
19.1-8	"Dominant" contributors to HCLPF using the MIN/MAX approach (by plant damage class)	19-20
19.1-9	Examples of configurations that meet 1E-5 CCDF goal in Modes 3, 4, and 5	19-30
19.1-10	GE's point estimates and the staff's mean estimates of the internal events risk	19-33
19.2-1	GE ABWR drywell head comparison of membrane stresses	19-71
19.3-1	Comparison of ABWR shutdown TS and BWR-6 TS	19-89
20-1	ABWR - Relevant USIs and GSIs	20-2
20-2	ABWR - CP/ML Rule Items	20-11
20.5.1-1a	Comparison of GE and staff adjusted offsite consequences (person-Sv)	20-79
20.5.1-1b	Comparison of GE and staff adjusted offsite consequences (person-rem)	20-79
20.5.1-2	Cesium and Iodine release fractions, as estimated by the staff and GE	20-80
20.5.1-3	Potential Design Improvements and Associated Costs (Provided by GE)	20-82
20.5.1-4	Summary of GE's assessment of risk reduction for candidate design improvements	20-85

CONTENTS

VOLUME 2: APPENDICES

	Page
APPENDIX A	A-1
LIST OF ABBREVIATIONS	A-1
APPENDIX B	B-1
REFERENCES	B-1
APPENDIX C	C-1
CHRONOLOGY OF CORRESPONDENCE	C-1
APPENDIX D	D-1
FSER CONTRIBUTORS	D-1
APPENDIX E	E-1
STAFF POSITION ON SHELL BUCKLING DUE TO INTERNAL PRESSURE	E-1
APPENDIX F	F-1
STAFF POSITION ON STEEL EMBEDMENTS	F-1
APPENDIX G	G-1
STAFF POSITIONS AND TECHNICAL BASES ON THE USE OF AMERICAN NATIONAL STANDARDS INSTITUTE (ANSI)/AMERICAN INSTITUTE OF STEEL CONSTRUCTION (AISC) N690, "NUCLEAR FACILITIES - STEEL SAFETY-RELATED STRUCTURES"	G-1
APPENDIX H	H-1
DYNAMIC LATERAL SOIL PRESSURES ON EARTH RETAINING WALLS AND EMBEDDED WALLS OF NUCLEAR POWER PLANT STRUCTURES	H-1
APPENDIX I	I-1
EVALUATION OF ABWR PUMP AND VALVE INSERVICE TESTING PLAN (SSAR TABLES 3.9-8 AND 3.9-9)	I-1
APPENDIX J	J-1
HUMAN FACTORS ENGINEERING PROGRAM REVIEW MODEL AND ACCEPTANCE CRITERIA FOR EVOLUTIONARY REACTORS	J-1
APPENDIX K	K-1
IMPORTANT SAFETY INSIGHTS	K-1

VOLUME 2: APPENDICES

FIGURES

		Page
G-1	Structural stability curves for the axially loaded compression numbers.	G-3
J.1	HFB program review model elements	J-4
J.2	HFB program review stages	J-5

1 INTRODUCTION AND GENERAL DISCUSSION

1.1 Introduction

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

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

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

This safety evaluation report (SER) documents the staff's review of the ABWR SSAR, up to and including Amendment 35, and the CDM, up to and including Revision 4. The staff also reviewed the SSAR, CDM, and final technical specifications to ensure that this information was internally consistent. This resolved Confirmatory

Item F1.1-1. This final SER documents the results of the staff's safety review of the U.S. ABWR standard design against the requirements of Subpart B of 10 CFR Part 52 (design certification) and delineates the scope of the technical details considered in evaluating the proposed design. The principal matters of the staff's review are summarized in Section 1.5 of this report.

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

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

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

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

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

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

Introduction and General Discussion

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

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

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

1.2 General Design Description

The U.S. ABWR design is similar to the international ABWR design, which is being built at the Kashiwazaki

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

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

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

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

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

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

1.2.2 Precertification and Postcertification Design Control Procedures

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

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

1.2.3 The role of the COL Applicant or Licensee

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

1.2.4 Control of Design Documentation in Support of the Certified Design

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

1.2.5 GE's Agreement With the Japanese

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

1.2.6 Scope of Certified Design

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

Introduction and General Discussion

1.2.7 Summary of U.S. ABWR Standard Design

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

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

Reactor

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

The reactor core will consist of 872 fuel bundles in an 8-by-8 array and 205 control rods operating at a power density of 50 kW/liter. The control rods, which will enter

from the bottom of the reactor core, will perform dual functions of power distribution shaping and reactivity control. Manipulation of selected patterns of rods will control power distribution, while electro-hydraulic drive mechanisms or hydraulic rapid scram insertion will control reactivity.

Reactor Coolant System

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

Reactor Protection System

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

Containment

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

Electrical Power Distribution Supply

The electrical power distribution system is a complete three load group distribution system with two independent offsite power sources, the main turbine generator, three onsite standby power sources (emergency diesel generators), and a combustion turbine generator located on site. During normal plant operations, the main generator will supply power to the main power transformer (MPT) and

three unit auxiliary transformers (UATs) through the main generator output breaker and an isolated phase bus. When the main generator is off line, power will be supplied to the UATs and the MPT by the preferred offsite power source.

Emergency Core Cooling System

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

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

Control Room

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

Power Conversion

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

1.3 Comparison With Similar Facility Designs

While many features of the ABWR design are similar to those of BWR designs that the staff had previously

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

1.4 Identification of Agents and Contractors

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

1.5 Summary of Principal Review Matters

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

The staff's technical review of GE's application for certification of the U.S. ABWR standard design was performed in accordance with Commission guidance and the applicable requirements of Subpart B of 10 CFR Part 52. This final SER documents the results of the staff's technical review. The staff's evaluation of the technical information required by 10 CFR 52.47(a)(1)(i) was performed in accordance with the standard review plan (SRP, NUREG-0800) and is discussed throughout this report. The evaluation of the technically relevant unresolved safety issues, generic safety issues, and Three Mile Island requirements (10 CFR 52.47(a)(1)(ii) and (iv)) is given in Chapter 20 of this report. Site parameters required by 10 CFR 52.47(a)(1)(iii) are evaluated in Chap-

Introduction and General Discussion

ter 2 and listed in Section 2.6. The design-specific probabilistic risk assessment (10 CFR 52.47(a)(1)(v)) is evaluated in Section 19.1. Inspections, tests, analyses, and acceptance criteria (ITAAC), as required by 10 CFR 52.47(a)(1)(vi) and (viii), are evaluated in Section 14.3. Interface requirements and representative conceptual designs (10 CFR 52.47(a)(1)(vii) and (ix)) are evaluated in Chapters 8 and 9 and are also discussed in Section 14.3. The ABWR technical specifications are evaluated in Chapter 16 of this report. The staff also implemented the Commission's Severe Accident Policy Statement, dated August 8, 1985, and the Commission's SRMs on SECY-90-016 and SECY-93-087 in its resolution of severe accident issues. The staff's evaluation of severe accident issues is given in Section 19.2 of this report.

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

The staff used the safety standards in 10 CFR 52.48 as the basis for its review of the U.S. ABWR standard design. It also followed Commission guidance given in the SRMs for various Commission papers referenced throughout this report and identified in Appendix B. As a result of this guidance, the staff proposed design-specific regulations that

are applicable to the ABWR design and justified exemptions from existing regulations to complete the regulatory framework of safety standards. An index of these safety standards is contained in Section 1.6 of this report.

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

1.6 Index of Applicable Regulations and Exemptions

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

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

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

1.7 Index of Tier 2* Information

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

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

Introduction and General Discussion

1.8 Index of Confirmatory Items

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

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

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

2 SITE CHARACTERISTICS

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

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

2.1 Geography and Demography

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

2.1.1 Site Location and Description

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

2.1.2 Exclusion Area Authority and Control

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

2.1.3 Population Distribution

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

2.2 Nearby Industrial, Transportation, and Military Facilities

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

2.2.1 and 2.2.2 Identification of Potential Hazards in Site Vicinity

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

2.2.3 Evaluation of Potential Accidents

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

2.3 Meteorology

2.3.1 Regional Climatology

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

Site Characteristics

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

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

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

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

as general aviation or nearby explosions. This was DFSER COL Action Item 2.3.1-1. Section 2.3.2.6 of the SSAR includes this action item. This is acceptable.

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

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

Table 2-1 Design-basis tornado characteristics

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

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

Figure 2-1 Tornado intensity regions

Site Characteristics

2.3.2 Local Meteorology

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

2.3.3 Onsite Meteorological Measurements Program

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

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

2.3.4 Short-Term Dispersion Estimates for Accidental Atmospheric Releases

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

2.3.5 Long-Term Diffusion Estimates

The staff will evaluate annual average atmospheric dispersion values for routine releases using the guidance in RG 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases From Light-Water-Cooled Reactors," Revision 1. The staff will use these values to perform its

10 CFR Part 50, Appendix I, and 10 CFR Part 20 evaluations for a COL application. The COL applicant should provide to the NRC annual average atmospheric dispersion values for routine releases. This was DFSER COL Action Item 2.3.5-1. GE has included this action item in Section 2.3.2.10 of the SSAR. This is acceptable.

2.4 Hydrologic Engineering

2.4.1 Hydrologic Description

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

2.4.2 Floods

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

2.4.3 Probable Maximum Flood on Streams and Rivers

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

2.4.4 Ice Effects

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

2.4.5 Cooling Water Channels and Reservoirs

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

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

2.4.6 Channel Diversion

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

2.4.7 Flooding Protection Requirements

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

2.4.8 Cooling Water Supply

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

2.4.9 Accidental Release of Liquid Effluents in Ground and Surface Waters

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

2.4.10 Technical Specification and Emergency Operation Requirement

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

2.5 Geology, Seismology, and Geotechnical Engineering

The COL applicant should provide site-specific, basic geologic and seismic information and site-specific

information on vibratory ground motion, surface faulting, stability of subsurface materials and foundations, slopes, and embankments and dams as described in the following sections.

2.5.1 Basic Geologic and Seismic Information

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

2.5.2 Vibratory Ground Motion

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

2.5.3 Surface Faulting

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

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

Site Characteristics

2.5.4 Stability of Subsurface Materials and Foundations

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

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

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

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

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

- (2) GE evaluated the effects of variations in water table locations on structural response by considering three water table locations: low, intermediate, and high. It assumed the high water table will be located at 0.6 m (2 ft) below grade, while the low water table will be located at 26 m (85 ft) below

grade at the base of the reactor building foundation basemat. The intermediate water table was assumed to be located at 12 m (40 ft) below grade, approximately at the midheight of the reactor building embedment.

- (3) GE considered a range of soil profiles based on the shear wave velocity profiles used in GESSAR II (NUREG-0979, April 1983) and selected six velocity profiles for the SSI analyses.
 - The first soil profile was assumed to consist of seven horizontal layers. The shear wave velocity, V_s , of the soil at a depth, y , below the ground surface was calculated as a function of the effective mean pressure at that depth and a modulus parameter.
 - The second through the sixth soil profiles were selected on the basis of three generalized soil zones: a soil zone for the second profile (sands, silts, clays, and gravelly soils), a transition zone for the third and fourth profiles, and a soft rock and well-cemented soil zone for the fifth and sixth profiles. Their velocity profiles are smooth curves representative of the average variation of shear modulus with depth.
 - The seventh profile represents an HR site with a uniform V_s of 1,525 m/sec (5,000 ft/sec).
 - The eighth profile represents an EHR site with a uniform V_s of 3,050 m/sec (10,000 ft/sec).

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

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

2.5.4.1 Site and Facilities

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

2.5.4.2 Field Investigations

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

2.5.4.3 Laboratory Investigations

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

2.5.4.4 Subsurface Conditions

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

2.5.4.5 Excavation and Backfilling for Foundation Construction

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

2.5.4.6 Effect of Ground Water

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

2.5.4.7 Liquefaction Potential

GE stated in response to NRC staff Question 241.1c that one of the eight conditions in Section 3A.1 of Appendix 3A of the SSAR required that no potential for liquefaction of soils shall exist at the plant as a consequence of the OBE and the SSE as reviewed and concurred in by the NRC staff. That condition further required that the liquefaction potential of the foundation and site soils be investigated and reported for a long-duration, New Madrid-type earthquake. GE clarified the statement regarding the New Madrid-type earthquake by stating that the maximum ground motion will be the same as the SSE

Site Characteristics

and the actual duration of the earthquake chosen for a specific site will be reviewed by the NRC staff and approved at the time of an individual application. GE further stated that, without knowing the exact site location and its seismic hazard, it cannot specify the earthquake magnitude and the number of strong motion cycles for the liquefaction evaluation of a site.

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

2.5.4.8 Response of Soil and Rock to Dynamic Loading

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

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

2.5.4.9 Maximum Soil-Bearing Pressures

In Appendix 3H of the SSAR, GE gives a method for calculating maximum bearing pressure under the reactor building foundation mat for three load cases. Load Case 1 considers dead load, live load, and a combination of horizontal and vertical SSE components with the vertical component acting downward. Load Case 2 is the same as Load Case 1 except that the vertical seismic component acts upward and the live load is omitted. Load Case 3 is the same as Load Case 2, except for the addition of the buoyancy effect. Because the soil bearing pressure based

on static equilibrium in the case of uplift would be very conservative, GE used the energy balance method. GE specified a minimum static soil bearing capacity of 0.72 megapascal (MPa) (15 kips/ft²). The maximum bearing pressure for the ABWR standard plant, which is due to dead load alone, as calculated by GE, is 0.63 MPa (13.1 kips/ft²). In the DFSER, the staff noted that the COL applicant should provide the site-specific maximum soil pressures along with supporting calculations and compare them with the allowable values. This was DFSER COL Action Item 2.5.4.9-1. SSAR Table 2.0-1 lists the minimum static bearing capacity of 0.72 MPa (15 kips/ft²) as the site design parameter. GE has revised Section 2.3.2.33 of the SSAR by stating that the COL applicant will demonstrate that the site has a minimum static bearing capacity at the foundation level of the reactor and control buildings. For other safety-related plant facilities, the COL applicant will demonstrate that the foundation material has adequate bearing capacity to withstand the site-specific loads. This is acceptable.

2.5.4.10 Earth Pressures

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

2.5.4.11 Soil Properties for Seismic Analysis of Buried Pipes

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

2.5.4.12 Static and Dynamic Stability of Facilities

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

2.5.4.13 Subsurface Instrumentation

The COL applicant should describe instrumentation, if any, proposed for monitoring of the performance of the

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

2.5.4.14 Stability of Slopes

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

2.5.4.15 Embankments and Dams

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

2.6 Site Parameter Envelope

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

document for evolutionary plants when developing the final list. This was DFSER Open Item 2.6-1.

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

- All units and dimensions in these tables should be in the metric system with English units or dimensions provided in brackets.
- The following information should be added to the bounding site parameter for tornado:
 - rate of pressure drop: 8.3 kPa (1.2 lb/in²/sec)
- The following additions or changes should be made for the seismology bounding site parameter:
 - Note (10) should be added to the fourth bullet and modified to read: "SSE Time History: Envelope SSE Response Spectra⁽¹⁰⁾"
 - Note (9) should be changed to read: "The minimum bearing capacity should be referred to as the static bearing capacity."
 - New Note (10) should read: "The response spectra of the SSE time history to be used in the free field must envelop the free field design response spectra for all damping values to be used in the response analysis. In addition, the time history should also be justified to show its adequacy by demonstrating sufficient energy at the frequencies of interest through the generation of the power spectrum density (PSD) function, which is greater than the target PSD function throughout the frequency range of significance."

GE responded to this open item by letters to the staff dated April 16 and June 16, 1993. The staff and GE also discussed it in several conference calls and in the meeting on ITAAC (inspections, tests, analyses, and acceptance criteria) on July 27 through 29, 1993. To provide a comprehensive list of site parameters, the following site parameters were added to SSAR Table 2.0-1, "hazards in site vicinity," and "tornado, rate of pressure drop." This is acceptable. To address the issues identified in Sections 1.4 and 4.5.2 of the SER on EPRI's ALWR requirements document for evolutionary plant, SSAR Table 2.0-1 defines minimum "static" bearing pressure and establishes a requirement, "SSE time history: envelope SSE response spectra." This is acceptable. The staff also reviewed the enveloping meteorological dispersion values

Site Characteristics

in Table 2.0-1 of the SSAR. The staff finds that GE determined these enveloping values using acceptable methodology given in RGs 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents From Light-Water-Cooled Power Reactors," Revision 1, and 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," Revision 1. The methods used by GE for determining these enveloping values are described in Sections 2.6.7 and 15.4 of this report. The staff finds the enveloping meteorological dispersion values acceptable. SSAR Section 3.7 addresses the issue noted in suggested Note (10) above. The staff's evaluation of this subject is discussed in detail in Section 3.7.1 of this report. The staff reviewed Table 2.0-1 of the SSAR and determined that it provides an adequate amount of information in support of Table 5.0 of the ABWR CDM. This is acceptable. In its letter of April 15, 1993, GE explained the use of only the Japanese metric units for the SSAR. Subsequently, GE agreed to use the International System of Units for the SSAR in accordance with the Commission-approved staff recommendations on implementing the metrication policy for evolutionary and revolutionary reactors. This is acceptable. On the basis of the above, DFSE Open Item 2.6-1 is resolved. The site parameter evaluation is summarized in the following sections.

2.6.1 Wind and Tornado Design Site Parameters

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

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

- **Maximum Tornado Wind Speed:** 483 km/hr
(300 mi/hr)
- **Translational Velocity:** 97 km/hr
(60 mi/hr)
- **Radius of Maximum Rotational Speed:** 45.7 m
(150 ft)

- **Maximum Atmospheric Pressure Drop:** 13.8 kPa
(2.0 lb/in²)
- **Missile Spectra:** SRP Section 3.5.1.4, Spectrum I

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

2.6.2 Water Level (Flood) Design Site Parameters

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

- **Floods:** The ABWR should be located on the site so that the level of the design-basis flood is no higher than 30.5 cm (1 ft) below the plant grade.
- **Potential Dam Failures (Seismically Induced):** Failure of existing and potential upstream or downstream water control structures should not contribute to the water level exceeding 30.5 cm (1 ft) below grade nor compromise the ultimate heat sink.
- **Probable Maximum Surge and Seiche Flooding:** The probable maximum surge and seiche flooding should be no higher than 30.5 cm (1 ft) below grade (see SSAR Table 2.1-1).
- **Probable Maximum Tsunami:** The probable maximum tsunami flooding should be no higher than 30.5 cm (1 ft) below grade (see SSAR Table 2.1-1).
- **Ground Water:** The ABWR is intended to be compatible with ground water levels up to 61 cm (2 ft) below plant grade.

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

2.6.3 Seismology Site Parameters

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

- **Vibratory Ground Motion:** In SSAR Table 2.0-1, GE specifies the 0.3g peak ground acceleration (PGA) value for the high frequency anchor for the SSE response spectra. The SSE response spectra are con-

structed in accordance with RG 1.60, "Design Response Spectra for Seismic Design of Nuclear Power Plants," Revision 2.

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

2.6.4 Soil Properties Site Parameters

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

- **Minimum Shear Wave Velocity:** The minimum embedment depth for the reactor building should be 26 m (85 ft). The minimum shear wave velocity of soil should be 305 m/sec (1,000 ft/sec).
- **Liquefaction Potential:** No liquefaction potential should exist for soils under and around all seismic Category I structures, including seismic Category I buried pipelines and electrical ducts.
- **Minimum Bearing Capacity (demand):** The minimum (demand) bearing capacity of soil should be 0.72 MPa (15 kips/ft²).

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

2.6.5 Precipitation (for Roof Design) Site Parameters

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

- **Maximum Rainfall Rate:** 49.3 cm/hr (19.4 in./hr)
- **Maximum Snow Load:** 2.35 kPa (50 lb/ft²)

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

2.6.6 Design Temperature Site Parameters

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

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

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

2.6.7 Atmospheric Dispersion (χ/Q)

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

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

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

Site Characteristics

The Table 2-2 bounding atmospheric relative concentration values for the LPZ (the 8-hour period from 0 to 8 hours, the 16-hour period from 8 to 24 hours, the 3-day period from 1 to 4 days, and the 26-day period from 4 to 30 days) were also determined by the most limiting design basis accident loss-of-coolant accident (LOCA) not to exceed the dose reference values given in 10 CFR Part 100 $3E+3$ MSv (300 rem) for the thyroid and $2.5E+2$ MSv (25 rem) for the whole body. As shown in Table 2.0-1 of the SSAR, the 2-hour LPZ atmospheric relative concentration value and an annual average (8760 hours) concentration value were obtained by logarithmic

extrapolation of these calculated LPZ bounding atmospheric relative concentrations.

The staff concludes in Section 15.4 of this report that GE's proposed bounding atmospheric relative concentrations (χ/Q) for the EAB and for the LPZ of the ABWR plant, in conjunction with the engineered safety features systems provided in the ABWR design, are sufficient to provide reasonable assurance that the radiological consequences of a postulated LOCA will be within the dose reference values in 10 CFR Part 100.

3 DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

3.1 General

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

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

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

3.1.1 Elimination of Operating Basis Earthquake from Design Consideration

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

There exist special circumstances for granting these exemptions from the requirements of Appendix A to Part 100 pursuant to 10 CFR 50.12. The purpose of designing SSCs necessary for continued operation without undue risk to the health and safety of the public to withstand an OBE is to ensure that these SSCs remain functional and within applicable stress and deformation limits when subjected to the effects of the OBE vibratory

ground motion. However, Appendix A to Part 100 also requires that these SSCs are designed to withstand the SSE and remain functional. Thus, when these SSCs are designed to remain functional for the SSE, they will also remain functional at a lesser earthquake level (one-third the SSE) provided all design functions at the OBE are accounted for. The basis for selecting one-third the SSE as the earthquake level at which the plant will be required to shutdown and be inspected for damage was that at this level the likelihood of damage and the frequency of earthquakes occurring was judged to be low based on actual earthquake experience. It should be noted that certain design functions had been only verified for the OBE loads in the past. These design functions were the evaluations of (1) fatigue damage caused by earthquake cycles and (2) relative seismic anchor motions in piping systems. With the elimination of the OBE from design, these design functions would not have been explicitly verified. Consequently, for the ABWR these design functions will be verified in conjunction with the SSE using applicable stress and deformation limits as described in Section 3.1.1.2 of this report.

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

3.1.1.1 Background

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

In SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs," the staff further requested that the Commission approve eliminating the OBE from the design of SSCs in both evolutionary and passive advanced reactors designs. The proposed amendment to 10 CFR Part 100, Appendix A, would allow, as an option, that the OBE be eliminated from design certification when

Design of Structures, Components, Equipment, and Systems

the OBE is established at less than or equal to one-third the SSE. In this manner, the OBE serves the function as an inspection-level earthquake below which the effect on the health and safety of the public would be insignificant and above which the licensee would be required to shut down the plant and inspect for damage. The Electric Power Research Institute (EPRI) requested the elimination of the OBE from design and the Advisory Committee on Reactor Safeguards (ACRS) also recommended OBE elimination in its letter of April 26, 1990.

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

- Use two SSE events with 10 maximum stress cycles per event to account for earthquake cycles in the fatigue analyses of piping systems performed until the new guidance is issued; alternatively, the number of fractional vibratory cycles equivalent to that of 20 full SSE vibratory cycles may be used (but with an amplitude not less than one-third of the maximum SSE amplitude) when derived in accordance with Appendix D to IEEE 344-1987.
- The effects of anchor displacements in the piping caused by an SSE be considered with the Service Level D limit.
- Eliminate the OBE from the design of SSCs. When the OBE is eliminated from the design, no replacement earthquake loading should be used to establish the postulated pipe rupture and leakage crack locations.
- The mechanistic pipe break and high-energy leakage crack locations determined by the piping high-stress and fatigue locations may be used for equipment environmental qualification and compartment pressurization purposes.
- With the elimination of the OBE, two alternatives exist that will essentially maintain the requirements provided in IEEE 344-1987 to qualify equipment with the equivalent of five OBE events followed by one SSE event. Of these alternatives, the equipment should be qualified with five one-half SSE events followed by one full SSE event. Alternatively, a number of fractional peak cycles equivalent to the maximum peak cycles for five one-half SSE events may be used in accordance

with Appendix D to IEEE 344-1987 when followed by one full SSE.

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

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

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

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

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

one-third of the SSE, the load combinations with the SSE generally control the design. Therefore, for primary stresses in piping systems, the staff finds that eliminating the OBE from piping stress load combinations will not cause a reduction in existing safety margins because the load combination with the SSE loading is generally controlling.

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

Fatigue

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

Seismic Anchor Motion

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

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

Piping Stress Limits

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

ASME Code Class 1 Piping Stress Limits

- (a) For primary stress evaluation (NB-3654.2), earthquake loads are not required to be evaluated for consideration of Level B Service Limits for Eq. (9).
- (b) For satisfaction of primary plus secondary stress intensity range (NB-3653.1), in Eq. (10), M should be either (1) the resultant range of all loads considering one-half the range of the SSE or (2) the resultant range of moment owing to the full range of the SSE alone, whichever is greater. The use of the SSE is intended to provide a bounding design for the cumulative effects of earthquakes of a lesser magnitude and is therefore to be included in consideration of Level B Service Limits for Eq. (10). A reduced range (with an equivalent number of fractional vibratory peak cycles) of the SSE moment may be used for consideration of Level B Service Limits (but with a range not less than one-third of the maximum SSE moment range).
- (c) For satisfaction of peak stress intensity (NB-3653.2), the load sets developed in NB-3653.1 that are based on the preceding Position b should be used in calculating the peak stress intensity, S , and the alternating stress intensity, S , for evaluating the fatigue effects and cumulative damage.
- (d) For simplified elastic-plastic discontinuity analysis (NB-3653.6), if Eq. (10) cannot be satisfied for all pairs of load sets, then the alternative analysis as described in NB-3653.6 will be followed. For treatment of seismic anchor motion moments, the

Design of Structures, Components, Equipment, and Systems

following condition should be satisfied in consideration of Level D Service Limits:

$$S_{SAM} = C_2 \frac{D_0}{2I} (M_1^* + M_1^{**}) \leq 6.0 S_m$$

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

M_1^* is the same as M_1^* in Eq. (12)

M_1^{**} is the same as M_1 in Eq. (10), except that it includes only moments caused by seismic anchor motion displacements that are caused by an SSE

The combined moment range ($M_1^* + M_1^{**}$) should be either (1) the resultant range of thermal expansion and thermal anchor movements plus one-half the range of the SSE anchor motion or (2) the resultant range of moment owing to the full range of the SSE anchor motion alone, whichever is greater.

ASME Code Class 2 and 3 Piping Stress Limits

- (a) For consideration of occasional loads (NC/ND-3653.1), earthquake loads (i.e., inertia and seismic anchor motion) are not required for satisfying Level B Service Limits for Eq. (9).
- (b) For consideration of thermal expansion or secondary stresses (NC/ND-3653.2), M_c in Eq. (10) is not required to include the moment effects of SAMs caused by an earthquake.
- (c) For consideration of secondary stresses in Level D Service Limit (NC/ND-3655), the following condition will be satisfied:

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

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

M_c is the range of moments owing to thermal expansion

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

moment owing to the full range of the SSE anchor motion alone, whichever is greater.

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

3.1.1.3 Pipe Break Postulation Without OBE

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

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

high-stress threshold limit. The resulting criteria, which were incorporated in SRP Section 3.6.2, resulted in many postulated pipe break locations and caused the installation of numerous pipe rupture mitigation devices in nuclear plants.

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

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

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

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

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

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

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

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

Design of Structures, Components, Equipment, and Systems

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

3.1.1.4 Concrete and Steel Structures

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

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

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

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

SSE Relative Displacements Between Structures

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

Seismic Instrumentation

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

Use of Regulatory Guides 1.143 and 1.27

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

3.1.1.5 Equipment Seismic Qualification

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

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

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

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

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

qualification when the OBE is eliminated from design are acceptable.

3.1.1.6 Pre-Earthquake Planning and Post-Earthquake Operator Actions

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

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

3.1.1.7 EPRI NP-5930

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

1. A free field instrument must be used for determining the CAV and the spectral acceleration level.
2. The response spectrum check is as follows:

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

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

Design of Structures, Components, Equipment, and Systems

4. The instrumentation installed at the nuclear power plant shall be capable of on-line digital recording of all three components of the ground motion and of converting the recorded (digital) signal into the standardized CAV and the 5-percent damped response spectrum. The digitizing rate of the time history of the ground motions shall be at least 200 samples per second and the band-width shall be at least from 0.20 Hz to 50 Hz. The pre-event memory of the instrument shall be sufficient to record the onset of the earthquake.
5. The system must be capable of routinely calibrating the response spectrum check of 0.20g. Also, the CAV of 0.16 g-sec should be calibrated with a copy of the October 1987, Whittier, California earthquake or an equivalent calibration record provided for this purpose by the manufacturer of the instrumentation. In the event that an actual earthquake has been recorded at the plant site, the above calibration shall be performed to demonstrate that the system was functioning properly at the time of the earthquake.

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

3.1.1.8 EPRI NP-6695

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

Section 3.1, Short-Term Actions

Item 3, "Evaluation of Ground Motion Records"

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

Item 4, "Decision on Shutdown"

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

Add Item 7, "Documentation"

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

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

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

Section 4.3.4, "Pre-Shutdown Inspection"

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

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

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

3.1.1.9 Conclusions

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

3.1.2.5.2 Fracture Prevention of Containment Boundary

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

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

3.2 Classification of Structures, Systems, and Components

3.2.1 Seismic Classification

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

assurance (QA) requirements of Appendix B to 10 CFR Part 50 should be applied to all activities affecting the safety-related functions of seismic Category I SSCs. The staff reviewed the ABWR SSAR in accordance with SRP Section 3.2.1, which references RG 1.29.

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

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

For this reason, the staff position, which was discussed in Section II.E of SECY-93-087 and was approved by the Commission in its SRM dated July 21, 1993, is that the main steam piping beyond the outermost isolation valve up to the seismic interface restraint and connecting branch lines up to the first normally closed valve be classified as QG B (SC 2) and seismic Category I. The MSL from the seismic interface restraint up to but not including the turbine stop valve (including branch lines to the first normally closed valve) will be classified as QG B and inspected in accordance with the applicable portions of ASME Section XI. This portion of the steamline may be classified as non-seismic Category I if it has been analyzed using a dynamic seismic analysis method to demonstrate its structural integrity under SSE loading conditions. However, all pertinent QA requirements of Appendix B to 10 CFR Part 50 are applicable to ensure that the quality of

Design of Structures, Components, Equipment, and Systems

the piping material is commensurate with its importance to safety during normal operational, transient, and accident conditions. To ensure the integrity of the remainder of GE's proposed alternative leakage path, the staff position is that (1) the main steam piping between the turbine stop valve and the turbine inlet, the turbine bypass line from the bypass valve to the condenser, and the main steam drain line from the first valve to the condenser are not required to be classified as safety-related or as seismic Category I, but should be analyzed using a dynamic seismic analysis to demonstrate their structural integrity under SSE loading conditions, and (2) the condenser anchorage shall be seismically analyzed to demonstrate that it is capable of sustaining the SSE loading conditions without failure.

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

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

- The MSL, including supports, from the outermost isolation valve to the seismic restraint, including branch lines up to the first valve is SC 2, QG B; QA will be in accordance with Appendix B to 10 CFR Part 50 and seismic Category I.
- The MSL from the seismic restraint to the turbine stop valve, including branch lines up to the first valve is QG B; QA will be in accordance with Appendix B to 10 CFR Part 50 and non-seismic Category I, but dynamically analyzed for the SSE. Although not explicitly stated in SSAR Table 3.2-1, the commitment to QG B requires inservice inspections in accordance with applicable portions of ASME Section XI.
- The turbine stop valve, the MSL from the turbine stop valve to the turbine, the turbine bypass valve, the turbine bypass line from the bypass valve to the condenser and the main steam drain line from the first valve to the condenser, are all non-safety class, non-

seismic Category I, but dynamically analyzed for the SSE.

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

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

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

Note f in SSAR Table 3.2-1 and SSAR Sections 3.7.2.8 and 3.7.3.13 state that equipment, structures, and piping in the ABWR that are non-seismic Category I but that could damage seismic Category I items if their structural integrity failed, are analyzed and designed to ensure their integrity is maintained under seismic loading from the SSE. At the interface between seismic and non-seismic Category I piping systems, the seismic analysis for the seismic Category I system will be extended to either the first anchor point in the non-seismic system or to sufficient distance in the non-seismic system so as not to degrade the validity of the seismic Category I analysis. In the DFSER, the staff noted that this commitment is in conformance with RG 1.29. However, as a part of the resolution of this

issue, the staff noted that the turbine building and any other applicable structure or equipment should be seismically analyzed for the SSE (RG 1.60 ground response spectrum anchored to 0.3g peak acceleration) to ensure that an earthquake will not adversely affect the structural integrity of the main steam piping, bypass line, valves, and instruments mounted on these pipes, and the main condenser. In Amendment 20 to the SSAR, GE added Section 3.7.3.16, which described a static, in lieu of dynamic, analysis for the turbine building. This proposed analysis was in accordance with the Uniform Building Code (UBC) Zone 2A, but was substantially less conservative than the specified SSE for which the main steam and bypass lines should be analyzed. The staff asked GE to provide a clear set of criteria for the turbine building, which is designed to UBC criteria, so that it will neither suffer no loss of function at the specified SSE nor adversely affect safety-significant piping, pipe-mounted equipment, or the condenser itself. This issue was DFSE Open Items 3.2.1-2 and 3.7.2-7. The resolution of this issue is discussed in the "Turbine Building" portion of Section 3.7.2 in this report.

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

resolution of this issue is also discussed in Section 3.12.3.8 of this report.

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

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

3.2.2 System Quality Group Classification

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

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

Design of Structures, Components, Equipment, and Systems

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

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

<u>NRC QG</u>	<u>ABWR SC</u>	<u>Section III</u>
A	1	1
B	2	2
C	3	3
D	NNS	-

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

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

In Section 3.2.2 of the DSER (SECY-91-153), the staff stated its concern about the QA requirements of the new and spent fuel storage racks and the storage container for defective fuel. This was Outstanding Issue 2 in

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

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

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

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

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

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

3.3 Wind and Tornado Loadings

3.3.1 Wind Design Criteria

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

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

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

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

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

GE meets the requirements of GDC 2 with respect to the capability of the structures to withstand design wind loading so that their design reflects the following:

- (1) appropriate consideration for the SSAR Table 2.0-1 basic design wind with an appropriate margin;
- (2) appropriate combinations of the effects of normal accident conditions with the effects of the natural phenomena; and

Design of Structures, Components, Equipment, and Systems

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

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

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

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

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

3.3.2 Tornado Design Criteria

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

SSAR Section 3.3.2 specifies that all seismic Category I structures exposed to tornado forces are designed to resist a maximum tornado wind speed of 483 km/hr (300 mi/hr)

and translational wind velocity of 97 km/hr (60 mi/hr). This also implies a maximum tangential velocity of 386 km/hr (240 mi/hr). Also specified are a simultaneous atmospheric pressure drop to 13.8 kPa (2.00 lbf/in²) at the rate of 8.3 kPa/sec (1.20 lbf/in²/sec) and the radius of maximum tornado is 45.7 m (150 ft). In SECY-93-087, the staff recommended that the Commission approve its position to employ a maximum tornado wind speed of 483 km/hr (300 mph) in the design of evolutionary and passive ALWRs. In its SRM dated July 21, 1993, the Commission approved the staff's position. On the basis of this evaluation, the staff concludes that the ABWR design meets the Commission-approved staff recommendation for design basis tornado and is acceptable. This resolved Outstanding Issue 4 of the DSER (SECY-91-153).

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

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

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

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

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

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

3.4 Water Level (Flood) Design

3.4.1 Flood Protection

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

SSAR Section 3.4.1 discusses the flood protection measures that are applicable to the standard ABWR plant seismic Category I SSC for both external flooding and postulated flooding from plant component failures. The reactor and control buildings are designed to seismic Category I standards. In addition, portion of the radwaste building that is below plant grade is also designed to

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

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

Design of Structures, Components, Equipment, and Systems

that all the safety-related systems and components located inside seismic Category I structures are protected from ground water seepage and external floods and meet the guidance of SRP Section 3.4.1.

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

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

- (1) divisional flood walls at the -8200 mm (-26 ft-10 7/8 in.) elevation of the control and reactor buildings are 0.6 m (24 in.) or thicker,
- (2) doors and penetrations rated as 3-hour fire barriers are assumed to prevent water spray from crossing divisional boundaries,
- (3) floors are assumed to prevent water seepage to lower levels,
- (4) penetrations between floors for pipe, cable, HVAC duct, and other equipment will be designed to prevent water seepage to lower elevations from 200 mm of standing water through the use of seals or curbs,
- (5) equipment access hatches shall prevent water seepage to lower elevations from 200 mm (8 in.) of standing water. Hatches to filter/demineralizer compartments may not be required to prevent water seepage,
- (6) water from a pipe break is assumed to flow under non-watertight doors and spread evenly over the available areas. Water sensitive safety-related equipment is raised 200 mm (8 in.) above the floor. The depth of water is limited to less than 200 mm by using available floor space and limiting the volume of water sources. As mentioned earlier, floor drains, stairways, and elevator shafts provide drain paths to the building basements.

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

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

GE stated that the greatest flood hazard in the MST occurs as a result of a feedwater line break. The amount of water associated with the break will be limited by manually

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

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

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

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

GE also addressed control building flooding as a result of an RSW failure. A break in the RSW line inside the control building will allow water from the open cycle water source to flood the control building. The three redundant divisions of RSW supply cooling water to the reactor building cooling water (RCW) system heat exchangers located in the control building basement. Any flooding from sources above the basement that could adversely affect safe-shutdown equipment will be directed through floor drains, stairways, and elevator shafts to the basement. Each division is physically separated by

watertight doors, is equipped with a sump pump, and contains two sets of safety-grade level sensors in a two-out-of-four logic. The first (lower) set will alarm to alert the control room operator of the presence of excessive water in the room. The second set of sensors will inform the control room operator that a serious flood situation exists in the RCW/RSW room. In addition, these high-level sensors will trip the RSW pump and close the isolation valves for the affected division. As a result, a maximum of ~5 m (~16.4 ft) of water will collect in a divisional room. This amount of water will be contained in the affected divisional room and will not adversely affect the safety function of the redundant divisions of the RCW and RSW. This analysis demonstrates that the safety-related equipment is adequately protected from the effects of a pipe break in the RSW system. In the DFSER, the staff noted that this evaluation was acceptable subject to its inclusion in the SSAR. This was DFSER Confirmatory Item 3.4.1-3. The SSAR now includes this information, and Confirmatory Item 3.4.1-3 is resolved.

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

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

- (1) meet the applicable requirements regarding seismic, flood, fire, and environmental conditions,
- (2) be routed independently or through separate compartments to assure divisional separation,

Design of Structures, Components, Equipment, and Systems

- (3) withstand both hydrostatic and hydrodynamic effects from pipe breaks. Provisions for pressure relief shall be included as necessary,
- (4) ensure the integrity of the piping penetrations at interfacing buildings under design conditions,
- (5) allow periodic inspection of piping, cables, and penetrations,
- (6) include provisions for leak detection and water removal,
- (7) prevent unauthorized access to the tunnel,
- (8) include provisions for preventing fuel oil from accumulating near structures housing safety-related equipment.

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

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

included this information in the SSAR. Therefore, Open Item 3.4.1-1 is resolved.

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

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

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

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

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

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

description and the ITAAC are evaluated in Section 14.3 of this report. Therefore, DFSER Open Item 3.4.1-2 is resolved.

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

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

3.4.2 Water Level (Flood) Design Procedure

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

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

By limiting the design-basis flood elevation to 30.5 cm (1 ft) below grade and the design-basis ground water level to 61.0 cm (2 ft) below grade in the design of plant structures, and by employing the load combinations specified in SSAR Sections 3.8.4.3 and 3.8.5.3, the ABWR design provides sufficient margin to prevent structural damage that is due to the most severe flood or ground water and the associated dynamic effects that were determined appropriate for the flood levels. In addition,

the design of seismic Category I structures includes, in an acceptable manner, load combinations that occur as a result of the most severe flood or ground-water-related loads and the loads resulting from normal and accident conditions.

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

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

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

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

The staff finds that the ABWR flood design provides reasonable assurance that, in the event of floods or high ground water level, the structural integrity of the plant seismic Category I structures will not be impaired and,

Design of Structures, Components, Equipment, and Systems

consequently, seismic Category I systems and components located within these structures will be adequately protected and will be expected to perform necessary safety functions, as required.

3.5 Missiles

3.5.1 Missile Selection and Description

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

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

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

- (1) locating the system or component in a missile-proof structure,
- (2) separating redundant systems or components for the missile path or range,
- (3) providing local shields and barriers for systems and components,
- (4) designing the equipment to withstand the impact of the most damaging missile,
- (5) providing design features to prevent the generation of missiles,
- (6) orienting missile sources to prevent missiles from striking equipment important to safety.

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

3.5.1.1 Internally-generated missiles (Outside Containment)

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

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

- (1) potential missiles that could result from the failure of rotating equipment such as the main turbine, reactor core isolation cooling (RCIC) turbine, pumps, fans, blowers, diesel generators, and compressors
- (2) pressurized high-energy fluid system components considered potential missile sources, including valve bonnets, stems, pressure vessels, thermowells, retaining bolts, and blowout panels

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

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

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

- orienting missile sources to prevent unacceptable consequences of missile generation

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

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

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

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

SSAR Section 3.5.1.1.1.3 and SSAR Figure 3.5-2 provide information that shows the orientation of the turbine-generator in relation to the in-scope buildings which house safety-related equipment and identifies the low-trajectory missile strike zone. On the basis of this information, the staff concludes that no safety-related equipment is located within the turbine-generator missile strike zone. As a

result, this equipment is protected from a missile ejected from the turbine-generator. Therefore, the staff concludes that the ABWR design meets the guidelines of RG 1.115 and the requirements of GDC 4.

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

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

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

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

Verification of adequate separation of redundant divisions of safety-related systems ensures that these systems will not be rendered inoperable as a result of an internally-generated missile outside of containment. In the DFSER, this verification was identified as Open Item 3.5.1.1-1. Subsequently, GE submitted design descriptions and ITAAC for the buildings within the ABWR design scope. These include verifications related to missile protection for

Design of Structures, Components, Equipment, and Systems

safety-related SSCs. The adequacy and acceptability of the design description and ITAAC are evaluated in Section 14.3 of this report. On the basis of the above, DFSE Open Item 3.5.1.1-i is resolved.

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

3.5.1.2 Internally-generated missiles (Inside Containment)

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

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

GE's analysis of failures of rotating equipment indicates that the equipment design prevents such equipment from becoming a source of potential missiles. Pumps are unlikely to achieve an overspeed condition. All electrically-powered rotating equipment such as pumps and fans are ac powered and their speeds are governed by an ac power supply. Since the ac power supply frequency variation is limited to a narrow range, this rotating equipment is unlikely to attain an overspeed condition.

Fan blade casings are designed with sufficient thickness so that a fan blade breaking off at rated speed will not penetrate the fan casing. GE submitted PDE-18-0389, a proprietary missile generation study, which provides the details of the methodology used to ensure that safety-related SSCs will be protected from missiles that may be generated from rotating equipment. On the basis of the information provided in the study, the staff finds this methodology to be an acceptable means of protecting safety-related SSCs from missile damage.

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

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

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

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

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

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

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

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

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

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

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

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

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

The staff concludes that the ABWR design for protecting SSCs from internally generated missiles inside the containment meets the guidelines of SRP Section 3.5.1.2 and

Design of Structures, Components, Equipment, and Systems

the dynamic effect design bases requirements of GDC 4 and is acceptable.

3.5.1.3 Turbine Missiles

The staff requested that GE provide additional information (Q251.13 and Q251.14) on turbine missile generation. GE responded by revising SSAR Section 3.5.1.1.3 to include a layout of the turbine-generator building indicating the ± 25 degree low-trajectory turbine missile ejection zone. The turbine-generator building is favorably oriented with respect to the primary containment building so that any postulated turbine missile will not strike the primary containment building. GE also commits to meeting RG 1.115, "Protection Against Low-Trajectory Turbine Missile," Revision 1, which specifies that the probability of unacceptable damage from turbine missiles be maintained to less than 10^{-7} per reactor-year.

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

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

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

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

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

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

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

GE addressed this item in SSAR Section 3.5.1.1.3, stating that the COL applicant shall submit for NRC approval, within 3 years of obtaining a COL, a turbine system maintenance program, including probability calculations of turbine missile generation based on the methodology approved by the NRC, or commit to volumetrically inspect all low-pressure turbine rotors at the second refueling outage and every other (alternate) refueling outage thereafter until a maintenance program is approved by the staff. This is acceptable.

3.5.1.4 Missiles Generated by Natural Phenomena

The staff reviewed the design of the facility for protecting SSCs important to safety from missiles generated by natural phenomena in accordance with SRP Section 3.5.1.4. The SRP acceptance criteria specify that the design meet GDC 2 and 4. GDC 2 requires that SSCs important to safety be protected from the effects of natural phenomena; GDC 4 requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents

(LOCAs). The design is considered to be in compliance with GDC 2 and 4 if it meets RG 1.76, "Design Basis Tornado for Nuclear Power Plants," Revision 0, Positions C.1 and C.2, and RG 1.117, "Tornado Design Classification," Revision 1, Positions C.1 through C.3.

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

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

Design of Structures, Components, Equipment, and Systems

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

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

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

- (1) those necessary to ensure the integrity of the RCPB,
- (2) those necessary to ensure the capability to shut down the reactor and maintain it in a safe-shutdown condition (this includes both standby and cold shutdown capability), and
- (3) those whose failure could lead to radioactive releases resulting in calculated offsite exposures greater than 25 percent of the guideline exposures of 10 CFR Part 100.

Compliance with Positions C.1 through C.3 of RG 1.117 was identified by the staff as a part of Outstanding Issue 8 in the DSER (SECY-91-153). In the SSAR Section 3.5.1.4 revisions that followed, GE discussed Positions C.1 and C.2. These discussions are acceptable. Subsequent to this, GE informed the staff that Position C.3 of RG 1.117 would also be met. This was DFSE Confirmatory Item 3.5.1.4-2. Systems meeting the

protection guidelines of RG 1.117 have been identified in Table 3.2-1. These systems, with the exception of the standby gas treatment system (SGTS) charcoal adsorbers and certain instrumentation, are located in seismic Category I structures which are designed to withstand winds and missiles generated by the above-mentioned tornado.

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

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

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

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

Design of Structures, Components, Equipment, and Systems

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

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

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

3.5.1.5 Site Proximity Missiles (Except Aircraft)

In SSAR Section 3.5.1.5, GE originally stated that "external missiles other than those generated by tornadoes are not considered design basis . . .," since the resultant event probability is $\leq 10^{-7}$. In the DFSE, the staff identified it as a site-specific action item for the COL applicant to address. This was DFSE COL Action Item 3.5.1.5-1. Subsequently, in SSAR Section 3.5.4.3, Amendment 33, GE stated that the COL applicant shall provide an analysis that demonstrates that the probability

of missiles impacting the ABWR standard plant and causing consequences greater than those permitted in 10 CFR Part 100 guidelines is $\leq 10^{-7}$. This is acceptable.

3.5.1.6 Aircraft Hazards

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

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

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

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

Design of Structures, Components, Equipment, and Systems

information, the staff concludes that the ABWR design meets the guidelines of RG 1.117.

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

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

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

As was stated in Section 3.5.1.1 of this report, GE provided information in SSAR Section 3.5.1.1.3 and Figure 3.5-2 that shows the orientation of the turbine-generator in relation to the in-scope buildings which house safety-related equipment and identifies the low-trajectory missile strike zone. Based on this information, the staff concludes that no safety-related equipment is located within the turbine-generator missile strike zone. As a result, this equipment is protected from a missile ejected from the turbine-generator. Therefore, the staff concludes that the ABWR design meets the guidelines of RG 1.115.

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

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

requirements of GDC 2 and GDC 4. Therefore, the staff finds that the portion of the design within the ABWR scope meets the requirements of GDC 2 and GDC 4.

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

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

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

As was stated earlier, verification of adequate separation of redundant divisions of safety-related systems and location of these systems in structures designed to withstand externally generated missiles ensure that these systems will not be rendered inoperable as a result of the

missiles. In the DFSER, this verification was identified as Open Item 3.5.2-2. Subsequently, GE submitted design descriptions and ITAAC for the buildings within the ABWR design scope. These include verifications of divisional separation for safety-related SSCs and location of these systems in structures designed to withstand externally generated missiles. The adequacy and acceptability of the design description and ITAAC are evaluated in Section 14.3 of this report; therefore, DFSER Open Item 3.5.2-2 is resolved.

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

3.5.3 Barrier Design Procedures

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

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

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

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

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

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

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

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

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

Design of Structures, Components, Equipment, and Systems

effects of externally generated missiles. This is acceptable.

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

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

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

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

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

SSAR. This was identified as Confirmatory Item 3.4.1-1 in the DFSE and is discussed in Section 3.4.1 of this report.

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

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

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

- (1) a summary of the dynamic analyses applicable to high-energy piping systems, including sketches and data on calculated stress intensities, cumulative usage factors, and stress ranges for selecting break locations
- (2) descriptions showing how safety-related systems are protected from jets, flooding, and other adverse environmental effects resulting from failures of moderate energy piping systems
- (3) identification of specific features provided for protecting each of the systems listed in the ABWR SSAR tables

Design of Structures, Components, Equipment, and Systems

- (4) details on protection provided against the effects of piping failures to ensure MSL feedwater and isolation valve functional capability
- (5) typical examples, if any, where protection for safety-related systems and components against the dynamic effects of pipe failures includes enclosures in suitably designed structures or compartments, drainage systems, and equipment environmental qualifications

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

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

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

- (1) The staff concluded that SSAR Table 3.6-2 did not include all the systems, components, and equipment that have to be protected against piping failures. For example, the reactor service water (RSW) system and the equipment and components used to supply reactor building cooling water (RCW) system water to the residual heat removal (RHR) system, fuel pool cooling (FPC) heat exchangers, and the heating, ventilation, and air conditioning emergency chilled water (HECW) system refrigerators were not included in the SSAR table. Also, GE did not explain why certain systems such as the high-pressure core flooders (HPCF), the reactor core isolation cooling (RCIC) process

sampling, and the standby liquid control (SLC) systems were not listed in the SSAR Table 3.6-4. Subsequently, GE revised these tables to include all of these systems or provided justification for not including them in the tables. This was found to be acceptable and this issue was resolved as reflected in the DFSER.

- (2) In the DSER (SECY-91-153), the staff stated that GE had not provided the results of an analysis of the postulated worst-case piping failure of a moderate- or high-energy line (including total failure of non-seismic Category I piping systems) for the RCIC compartment, equipment and valve room, and other applicable areas outside the containment (e.g., RHR piping areas). GE subsequently provided subcompartment pressure analyses for compartments inside and outside of containment. The staff's review of these analyses is found in Section 6.2.1.7 of this report.
- (3) As discussed above, the staff also identified that GE had not provided a subcompartment analysis for the MST. The staff was concerned that the main steam and feedwater lines would be routed in a tunnel through the control building. A steam or feedwater line failure may render the control room uninhabitable and compromise the safe-shutdown capability. Further, postulated leakage cracks in the RSW system piping may adversely affect the control room habitability systems. Subsequently, GE provided subcompartment pressure analyses (including pipe failures in the MST) in SSAR Section 6.2.3. Evaluation of the analysis can be found in Section 6.2.1.7 of this report. The vulnerability of the main control room (MCR) as a result of flooding in the MST and failure of RSW piping in the control building basement was evaluated and discussed in Section 3.4.1 of this report. Other environmental effects of pipe failures are discussed in SSAR Section 3.11. The safety evaluation can be found in Section 3.11 of this report. Based on its review of flood protection for the MCR included in Section 3.4.1 of this report, the staff concludes that the ABWR design provides adequate assurance that the MCR can be protected from the adverse effects associated with a pipe failure. This issue was resolved as reflected in the DFSER.

The staff expressed a concern about the pressure values in SSAR Table 3I.3-15. The staff noted that the values appear to reflect the zone pressures that result during accident conditions, assuming that the blowout panels function properly. A more conservative scenario would assume failure of the blowout panels. In the DFSER, the

Design of Structures, Components, Equipment, and Systems

staff noted GE's commitment to modify the table to ensure that it reflects the highest anticipated pressures resulting from accident conditions, assuming failure of the blowout panels. This was DFSEER Confirmatory Item 3.6.1-2. Subsequently, after discussions with GE, the staff concluded that the blowout panels fail in a safe condition. Specifically, the panels fail open. Therefore, the values in the Appendix 3I tables accurately reflect conservative design conditions; therefore, DFSEER Confirmatory Item 3.6.1-2 is resolved.

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

which serve all three plant divisional areas. The staff has reviewed this arrangement and has concluded that the arrangement is acceptable for the following reasons:

- (1) The interconnecting ductwork can be automatically and manually isolated during plant events.
- (2) The ductwork is located high in the divisional spaces so that flood water cannot propagate from one division to another.
- (3) Should a fire occur in a division, the smoke-removal mode of the HVAC system will minimize the amount of smoke which can propagate to redundant divisions (see SSAR Sections 9.4 and 9.5.1).
- (4) The environmental effects of a postulated pipe failure outside containment (CUW or RCIC) are well within the environmental qualification limits for the equipment in each division.

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

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

As a result of its review of this information, the staff concludes that GE has provided sufficient design information and supporting analyses to ensure that the guidelines of ASB BTP 3-1 are met. Specifically, the ABWR design provides protection of safety-related equipment from the effects of high- and moderate-energy pipe failures by separation of redundant divisions of these systems, by distance, or by locating safety-related systems in enclosures that provide the necessary protection. Furthermore, all safety-related systems and components

meet the guidelines of RG 1.29 and are designed to seismic Category I standards. Piping failures are postulated in accordance with BTP MEB 3-1, pipe restraints are provided in accordance with the guidelines in SRP Section 3.6.2 (see Section 3.6.2 of this report), and the effects of the postulated piping failures, including those from nonseismic systems, have been considered. Based on this information, the staff concludes that the guidelines of BTP ASB 3-1 are met, and therefore, the requirements of GDC 4 are met.

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

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

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

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

GDC 4 requires that SSCs important to safety be designed to be compatible with and to accommodate the effects of the environmental conditions resulting from normal operations, maintenance, testing, and postulated accidents, including loss-of-coolant accidents (LOCAs). It also requires that they be adequately protected against dynamic effects (including the effects of missiles, pipe whipping, and discharging fluids) that may result from equipment failures and from events and conditions outside the nuclear power plant. In accordance with SRP Section 3.6.2, the staff reviewed GE's proposed criteria and methodology for the COL applicant to analyze and ensure adequate protection against and the dynamic effects that breaks in

high-energy fluid systems would have on adjacent safety-related SSCs with regard to pipe whip and jet impingement loadings. Pipe whip need only be considered for those high-energy piping systems having fluid reservoirs with sufficient capacity to develop a jet stream. GE correctly used the criteria for determining high- and moderate-energy lines in SRP Section 3.6.1, Branch Technical Position (BTP) ASB 3-1, in SSAR Section 3.6.2.1. GE listed all high-energy systems in SSAR Tables 3.6-3 and 3.6-4.

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

For ASME Class 1 piping, the staff position for postulating pipe breaks is delineated in SRP 3.6.2, BTP MEB 3-1. Before NRC issued Revision 2 of BTP MEB 3-1 in June 1987, breaks were postulated at intermediate locations between terminal ends of a pipe run if the maximum stress range as calculated by Eq.(10) $> 2.4 S_m$ and if the maximum stress range as calculated by either Eq. 12) or (13) $> 2.4 S_m$, where Eqs. (10), (12), and (13) and S_m are as defined in ASME Section III, Subsection NB 3653. This staff position was implemented in many plants operating today. In Revision 2 of BTP MEB 3-1, the same criteria were maintained for break exclusion in the containment penetration areas. However, for other areas, the criteria were revised to require that breaks be postulated at any intermediate locations when only Eq. (10) $> 2.4 S_m$. The use of Eqs. (12) and (13) was eliminated. This resulted in an inconsistency in the Revision 2 criteria in that they allow higher limits in the containment penetration areas than in other areas. The break exclusion areas should provide a margin greater than (or at least equal to) the margin for areas outside the break exclusion area. To determine the impact of this inconsistency, the staff obtained several independent analyses for both BWRs and PWRs that compared the number of

Design of Structures, Components, Equipment, and Systems

postulated pipe breaks resulting from the use of Revisions 1 and 2 criteria. These analyses indicated that the Revision 2 criteria will result in a significant increase in the number of postulated breaks, which may be counterproductive in terms of improving plant safety. Therefore, the staff recommended that SRP 3.6.2 be revised to reinstate the Revision 1 criteria related to allowing the use of Eqs. (12) and (13) for the postulation of intermediate pipe breaks in ASME Class 1 piping systems. Sections 3.6-1 and 3.6-2 in the SSAR contains criteria that are consistent with the Revision 1 criteria as previously discussed. Therefore, the staff concludes that this is an acceptable deviation from SRP 3.6.2, Revision 2.

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

1. SRP 3.6.2 recommends that an augmented inservice inspection program be implemented for those portions of piping within the break exclusion region. An augmented inservice inspection was DFSER COL Action Item 14.1.3.3.7.2-1. In a letter dated March 17, 1993, GE revised SSAR Section 3.6.2.1.4.2, Item (7) and added SSAR Section 3.6.5.3 to state that the COL applicant shall perform a 100-percent volumetric examination of circumferential and longitudinal pipe welds in the break exclusion region during each inspection interval as defined in Article IWA-2400, ASME Code, Section XI. The staff finds that this action for the design and examination of high-energy piping in the containment penetration area meets SRP Section 3.6.2 and is acceptable. GE has included this action item in Section 3.6.2.1.6.2 of the SSAR. The staff's resolution of this DFSER COL action item is also discussed in Section 3.12.7.2 of this report.
2. SRP 3.6.2 contains design, testing, and examination guidelines for guard pipes in the containment penetration areas. SSAR Section 3.6.2.4 states that the ABWR primary containment does not require guard pipes. This is because the ABWR design does not contain guard pipes as defined in Section 3.6.2.4 of RG 1.70, "Standard Format and Content of Safety Analysis Report for Nuclear Power Plants," Revision 3: "a guard pipe is a device to limit pressurization of the space between dual barriers of

certain containments to acceptable levels." The staff notes that SRP 3.6.2 uses the term "guard pipe" in a broader context than that in RG 1.70 to include all applicable sleeves in the containment penetration area. Section 3.6.2.1.4.2(6) in the SSAR provides design, testing, and examination requirements for such sleeves. These requirements are consistent with the guidelines in SRP 3.6.2 and are acceptable.

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

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

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

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

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

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

1. Section 3.6.2.2.2 of the SSAR was extensively modified to permit alternative analytical approaches in accordance with ANSI/ANS 58.2 Paragraphs 6.3.1 through 6.3.5. These alternative approaches include the use of the ANSYS and PDA computer programs which were used in the GE pipe whip sample analyses. This response adequately addresses the staff's original concern about the use of the PDA program alone for pipe whip analyses.
2. Section 3.6.2.3.3 of the SSAR was modified to include design requirements for pipe whip restraints with crushable material and rigid restraints in addition to U-bar pipe whip restraints. This response results in the SSAR being consistent with the information presented to the staff during the audits mentioned, and is acceptable.
3. Appendix 3L "Procedure for Evaluation of Postulated Ruptures in High Energy Pipes" was added to the SSAR. This appendix defines an acceptable procedure for evaluation of dynamic effects of fluid dynamic forces resulting from postulated ruptures in high energy piping systems. The four major steps in the evaluations included (1) the identification of rupture locations and rupture geometry, (2) the design and selection of pipe whip restraints, (3) the procedure for dynamic time-history analysis with simplified models, and (4) the procedure for dynamic time history analysis using detailed piping models.
4. The sample analysis, GE-NE-123-E070-0493, is intended to be illustrative of the GE pipe break analysis method and, although not a part of the SSAR, it supplements Appendix 3L, which is discussed above. The analysis documented the GE procedures for (1) calculating thrust forces at break locations, (2) performing nonlinear time-history pipe whip analyses, (3) demonstrating compliance with the SRP Section 3.6.2 stress requirements for piping near containment penetration areas, and (4) demonstrating that the GE PDA computer program was adequate for selecting the size of pipe whip restraints. The staff's review found that these GE procedures are acceptable,

and the intended objectives of the sample analysis were achieved.

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

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

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

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

Using the above HELSEA evaluation, the staff finds that an adequate level of protection is provided to ensure that the safety-related function of components, systems, and equipment will not be adversely affected by a postulated break in any ABWR high-energy piping systems. Plant arrangement provides physical separation to the extent practical and the HELSEA evaluation ensures that no more than one redundant train can be damaged. If damage could occur to more than one division of a redundant safety-

Design of Structures, Components, Equipment, and Systems

related system within 9.14 m (30 ft) of any high-energy piping, other protection devices such as barriers, shields, enclosures, deflectors, or pipe whip restraints are used. When necessary, the protection requirements are met through the use of walls, floors, columns, abutments, and foundations. Thus, the staff finds that the HELSA criteria satisfy the intent of the SRP 3.6.2 guideline by ensuring that structures are adequately designed to withstand the consequences of a worst-case pipe break with no adverse impact on the safety-related function of systems, components, and equipment and are acceptable.

3.6.2.1 Conclusions

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

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

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

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

can be maintained in a safe-shutdown condition in the event of a postulated rupture of a high- or moderate-energy piping system inside or outside the containment.

3.6.3 Leak-Before-Break Evaluation Procedures

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

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

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

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

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

3.7 Seismic Design

The staff reviewed the seismic design adequacy of the ABWR standard plant using SRP Sections 3.7.1 through 3.7.4 as the basis, and considered GE's responses to the open items, confirmatory items and COL action items identified in the DSER (SECY-91-153) and the DFSE. In addition, the staff conducted two design calculation audits at GE's office in San Jose, California, and two design calculation audits at Bechtel Power Corporation (consultant to GE) in San Francisco, California. The first

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

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

3.7.1 Seismic Design Parameters

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

In SSAR Section 3.7.1, Amendment 33, GE stated that the three components of the synthetic SSE ground motion time history used for the seismic analysis and design of the ABWR seismic Category I SSCs are adjusted in amplitude and frequency to obtain response spectra for damping ratios of 2-, 3-, 4-, 5-, and 7-percent of critical damping to envelop the SSE design ground response spectra at a sufficient number of frequency points as recommended by SRP Section 3.7.1 plus three additional frequencies at 40,

50, and 100 Hz. The power spectral density function (PSDF) of the two horizontal components of the synthetic SSE ground motion time history envelops the target PSDF specified in Appendix A to SRP Section 3.7.1 for a frequency range of 0.3 to 24 Hz. For the vertical component of the ground motion time history, GE followed the guideline of Appendix A to SRP 3.7.1 and developed a target PSDF for the vertical ground motion. GE also showed that the vertical component of the ground motion time history (synthetic time history) satisfied the PSDF enveloping criteria. The details of developing the target PSDF and the demonstration of PSDF enveloping process are described in a GE submittal dated May 7, 1993, (GE's responses to Item 3 of the audit report dated February 22, 1993). In addition, GE showed that the three components of synthetic time history are statistically independent of each other in that the cross-correlation coefficients at zero-time lag between these components are less than 0.16.

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

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

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

The ABWR seismic Category I structures have reinforced concrete mat foundations that are designed to be supported

Design of Structures, Components, Equipment, and Systems

on soil, rock, or compacted backfill. Appendix 3H to the SSAR, Amendment 33, specifies the following embedment depths from the plant finished grade to the bottom of the basemat for each seismic Category I structure:

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

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

Bounding Site Parameters

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

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

- The site-specific ground motion response spectra are bounded by the RG 1.60 design response spectra anchored to 0.3g.
- For a shallow soil site, the site-specific ground response spectra and the associated time history should be specified as the free field motion at a level that complies with the guidelines of SRP Section 3.7.1.

- The site soil static bearing capacity at the foundation level of the reactor and control building is 0.72 MPa (15 kip/ft²) minimum.

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

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

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

3.7.2 Seismic System Analysis

The staff's review of the seismic analysis of the seismic Category I SSCs includes the seismic analysis methods and acceptance criteria used for the ABWR seismic Category I structures, the reactor pressure vessel (RPV) and containment internal structures design. The radwaste building, which is not a seismic Category I structure and does not house any safety-related equipment and systems, is included in this review because GE elected to design this structure for the SSE seismic loads to ensure that the embedded portion of the building retains its structural

integrity during and after an SSE, and to prevent unacceptable leakage of the radwaste material outside the building. The seismic analyses of the turbine building and the condenser, which is located inside the turbine building but has a separate foundation, are also included in this review although these two structures are not seismic Category I. They are within the scope of this review because a portion of the MSL, which is required to be designed for the SSE based on the discussion in Section 3.2.1 of this report, is inside the turbine building, and one end of this portion of the MSL system and branch lines is anchored at the turbine building and the condenser.

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

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

- (1) As a result of changing the design basis tornado wind speed from 418 km/hr (260 mph) to 483 km/hr (300 mph) and adopting the use of tornado missile Spectrum 'I' per SRP Section 3.5.1.4, the RB superstructure and roof were thickened and the roof purlins strengthened. These structural design upgrades resulted in a need to modify the seismic models.
- (2) The original SSE responses of the ABWR seismic Category I structures, including the FRS, were obtained by multiplying the corresponding OBE analysis results by a factor of two. As a result, a significant design margin was included in the SSE responses. To eliminate the excessive design margin, GE initiated the seismic reanalyses for the RB complex and the control building to take advantage of the Commission-approved staff recommendations on OBE elimination.
- (3) The original building responses used for the seismic input to subsystems such as piping were obtained by enveloping the analysis results from two dynamic models, namely, models with and without the steel

stabilizer truss between the RSW and reinforced concrete containment vessel (RCCV). Because of the protection to be provided for the personnel during reactor hot standby and shutdown for maintenance and refueling, the RSW height was raised to the height of 15 cm (6 in.) below the RCCV top slab. This change eliminated the steel stabilizer truss from design and, thus, the stabilizer truss was deleted to result in the revised model for seismic reanalysis.

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

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

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

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

Design of Structures, Components, Equipment, and Systems

response method of analysis used, GE calculated a composite modal damping ratio using the strain energy technique. The techniques used for the dynamic analyses of structures discussed above are consistent with the guidelines of SRP Section 3.7.2 and, therefore, are acceptable.

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

From comparing the natural frequencies obtained from a 2-D fixed base building model and a 3-D fixed base building model with the embedment effect included, GE found that the torsional effect resulting from the eccentricity between the center-of-mass and center-of-rigidity of the seismic Category I structures (RB, control building, and embedded portion of radwaste building) on the seismic responses is negligible because of the

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

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

During the design calculation audit conducted on February 22 through 25, 1993, the staff was concerned about that, in its calculation of seismic loads from the live loads, GE reduced the live load on a global basis but did not provide the basis for the reduction or provide the criteria for the design of local structural elements such as slabs, beams, and columns. In SSAR Sections 3.8.1.3.1, 3H.1.4.3.1, 3H.2.4.3.1, and 3H.3.4.3.1, Amendment 33,

Design of Structures, Components, Equipment, and Systems

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

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

- (1) the collapse of any non-seismic Category I structure, system, or component will not cause the non-seismic Category I structure, system, or component to strike any seismic Category I structure, system, or component.
- (2) the collapse of any non-seismic Category I structure, system, or component will not impair the integrity of seismic Category I SSCs. This may be demonstrated by showing that the impact loads on the seismic Category I structure or system or component resulting from collapse of an adjacent non-seismic Category I structure or system or component, because of its size and mass are either negligible or smaller than those considered in the design (e.g., loads associated with tornado, including tornado missiles).
- (3) the non-seismic Category I SSCs will be analyzed and designed to prevent their failure under SSE conditions in such a manner that the margin of safety of these SSCs is equivalent to that of seismic Category I SSCs.

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

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

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

- a. Analyses of evaluations of construction deviations and design changes, or
- b. The design basis analyses will be repeated using the as-built condition.

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

Design of Structures, Components, Equipment, and Systems

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

Reactor Building

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

As discussed in SSAR Appendix 3A, Amendment 33, the SSI analysis cases were categorized into two groups. In the first group, 3-D SSI analyses of the RB and the control building were performed individually without considering of structure-to-structure interaction effects (Cases 1 through 16 in SSAR Table 3A-7). In the second group,

2-D SSI analyses of the RB and control building were performed considering individual buildings as well as multiple buildings, including the turbine building, to evaluate the structure-to-structure interaction effects (Cases 17 through 22 in SSAR Table 3A-7). For all these 22 SSI analyses, an embedment of 25.9 m (85 ft) from the finished grade to the bottom of the basement was used.

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

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

case to obtain the site enveloping results. From this review, the staff concludes that GE's modeling of the RB and supporting soil medium, use of the 2-D structural model, and SSI analysis methods comply with the guidelines of SRP Section 3.7.2 and the final seismic response envelopes, including the FRS envelopes, are reasonable and acceptable.

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

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

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

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

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

Design of Structures, Components, Equipment, and Systems

the design of the subsystems located in the RB are acceptable.

Control Building

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

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

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

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

In response to these DFSER open and confirmatory items, GE, following the same procedures that were applied to the RB, conducted a seismic reanalysis of the control building. In this seismic reanalysis, the site conditions and

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

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

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

From the review of Appendices 3A and 3G of SSAR Chapter 3 and the certified design material (CDM) for the GE ABWR design dated March 1992, the staff observed

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

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

Radwaste Building Substructure

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

The seismic analysis was performed using a fixed-base freestanding stick model to represent the structure. This simplified analysis model excludes the effects of both the

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

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

Turbine Building

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

However, the staff noted in the DFSER about the adequacy of using some multiple of the ground response spectrum as the input for the MSL analysis at the turbine building side anchor. This was DFSER Open Item 3.7.2-6. In SSAR Section 3.2.5.3, Amendment 33, GE stated that the dynamic input loads for the design of the MSLs in the turbine building are derived as follows: (1) for locations on the basemat, the amplified response spectra (ARS) (FRS in this report) shall be based upon the RG 1.60 response spectra normalized to 0.6g (i.e., 2 times the SSE ground response spectra) and (2) for locations at the

Design of Structures, Components, Equipment, and Systems

operating deck level (either operating deck or turbine deck), the ARS used shall be the same as those used at the RB end of the MST. SAMs shall be similarly calculated. The staff concludes that the dynamic input loads for the design of the MSLs inside the turbine building are acceptable because (1) a comparison of the response spectra at the RB foundation level with the RG 1.60 response spectra anchored to 0.6g ZPA shows that the RG 1.60 response spectra anchored to the same 0.6g ZPA envelop the response spectra at the RB foundation level and (2) the turbine operating deck is located at approximately the same elevation as the anchor point of the main steamline at the RB side and the response spectra at the RB end were generated using an acceptable analysis approach as discussed. DFSE Open Item 3.7.2-6 is resolved.

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

- (1) The seismic zone shall be "Zone 3."
- (2) For dual systems (i.e., shear wall with braced steel frame), one of the two systems must be designed to be capable of carrying all of the seismic loading without collapse. No credit will be given for the other system for resisting lateral loads.

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

Confirmation of Plant-Specific Seismic Design Adequacy

To confirm the site-specific seismic design adequacy of the standard plant, SSAR Section 2.3.1, Amendment 9, stated that the COL applicant shall demonstrate, according to the procedure specified therein, that it has satisfied the eight

site-dependent conditions specified in SSAR Section 3A.1. These eight site-dependent conditions were:

- (1) The peak ground acceleration is less than 0.30g SSE.
- (2) The site design response spectra are less than or equal to those given in RG 1.60 normalized to the peak ground accelerations in Condition 1.
- (3) There is no potential for liquefaction at the plant site as a result of an SSE as reviewed and concurred with by the NRC staff (the liquefaction potential of the foundation and site soils will be investigated and reported for a long duration, New Madrid-type earthquake).
- (4) There is no potential for fault movement at the plant site as reviewed and concurred with by the NRC staff.
- (5) The embedment depth of the reactor building is 25.9 m (85 ft). The excavation tolerance is ± 15 cm (± 0.5 ft).
- (6) The average shear wave velocity for the top 9 m (30 ft) of soil is 305 m/sec (1000 ft/sec) minimum. The upper bound shear wave velocity is 3048 m/sec (10000 ft/sec).
- (7) For layered soil sites with parameters that have very abrupt variations with depth, an analysis with site-unique properties will be performed to confirm the applicability of the generic analysis.
- (8) The soil-bearing capacity at the site is adequate to accommodate plant design loads.

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

- (1) GE should provide the criteria and the confirmation procedure for the site condition classified as shallow soil site in the SSAR. This was DFSE Open Item 3.7.2-8.
- (2) When a site-specific SSI analysis is performed for the SSE case, the three components of the ground motion time history should satisfy not only the PSDF enveloping criterion, but also the response

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

- (3) In its letter dated August 19, 1991, and in SSAR Section 2.3.1, Amendment 18, GE stated that COL applicants shall consider site-dependent bounding Conditions 6 and 7 above as two individual evaluation parameters when confirming the adequacy of the standard plant design for a specific site. The effect of soil layer depth was not considered or included in the evaluation. The staff was concerned that to compare Conditions 6 and 7 with the site-specific design parameters separately is not sufficient to confirm the design adequacy of the standard plant. The staff believed that these two conditions should be considered together with the depth of soil layers. In addition, the site-specific responses (structural member forces and FRS) should be compared to the response envelopes used for the standard plant design unless the site-specific parameters (shear wave velocity, number of soil layers, and depth of soil layers) can be demonstrated to be comparable to one of the 14 generic site conditions. This was DFSER Open Item 3.7.2-9.
- (4) In SSAR Section 2.3.1.2, Amendment 18, GE stated that the FRS comparison can be made for one damping value only when confirming the seismic design adequacy of piping and equipment. According to the guidelines of SRP Section 3.7.1.1.1.b, the FRS comparison should be performed for all damping values assigned to different piping systems and equipment. This was DFSER Confirmatory Item 3.7.2-8.

To respond to these four items, in SSAR Amendment 30, GE revised the eight site-dependent conditions by (1) moving these conditions from Appendix 3A to Section 2.3, "COL License Information," (2) eliminating Conditions 6 and 7, and (3) adding the condition for the shallow soil site. GE also revised the confirmation procedure by only comparing the site-specific SSE ground response spectrum of 5-percent damping at plant grade in the free-field with the design ground response spectrum (i.e., RG 1.60 response spectrum anchored to 0.3g PGA). The staff reviewed these revisions and determined that it is acceptable to compare the ground response spectra at plant grade in the free-field for the confirmation of the design adequacy and that there is no need to specifically confirm the adequacy of the ground motion time history, local soil layering effects, and FRS for the subsystem analysis and design. The basis for the staff conclusion is that when the site-specific ground motion response

spectrum is developed and the free-field surface ground motion is calculated, the local geotechnical properties (soil layering, shear module, etc.) are all considered. As far as the adequacy of site-specific ground motion time history and the generation of the FRS, the staff will review it on a case-by-case basis. Therefore, DFSER Open Items 3.7.2-8 and 3.7.2-9 and DFSER Confirmatory Items 3.7.2-7 and 3.7.2-8 are resolved.

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

1. Appropriate consideration for the most severe earthquake recorded for the most sites east of the Rocky Mountains with an appropriate margin (GDC 2);
2. Appropriate combination of the effects of normal and accident conditions with the effect of the natural phenomena; and
3. The importance of the safety functions to be performed (GDC 2). The use of a suitable dynamic analysis to demonstrate that SSCs can withstand the seismic and other concurrent loads.

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

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

GE performed the system and subsystem analyses on an elastic and linear basis. Time history methods form the bases for the analyses of all major seismic Category I

Design of Structures, Components, Equipment, and Systems

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

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

3.7.3 Seismic Subsystem Analysis

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

GE performed the subsystem analysis on a linear elastic basis. The modal response spectrum method and equivalent static load method formed the basis for the analyses of all major seismic Category I subsystems in

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

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

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

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

3.7.4 Seismic Instrumentation

The seismic instrumentation system specified for the ABWR plant in SSAR Section 3.7.4 is acceptable and

meets GDC 2 and 10 CFR Part 100 Appendix A. The applicant met these requirements by requiring installation of instrumentation that is capable of adequately measuring the effects of an earthquake.

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

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

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

3.8 Design of Seismic Category I Structures

3.8.1 Concrete Containment

The containment is designed as a reinforced-concrete cylindrical shell structure with an internal steel liner made of carbon steel, except for wetted surfaces where stainless

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

AIM

Association for Information and Image Management

1100 Wayne Avenue, Suite 1100
Silver Spring, Maryland 20910
301/587-8202

Centimeter

Inches

MANUFACTURED TO AIM STANDARDS
BY APPLIED IMAGE, INC.

2 of 10

Design of Structures, Components, Equipment, and Systems

and construction of reinforced concrete containment structural elements (including the tangential shear stress limit discussed above), would involve an unreviewed safety question and, therefore, require NRC review and approval prior to implementation. Furthermore, any requested change to the use of this code must either be specifically described in the COL application or submitted for license amendment after COL issuance.

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

In SSAR Section 3.8.2.4.1.4, Amendment 33, GE stated that the drywell head, which consists of shell, finger pin closure, and drywell-head anchor system, was analyzed using a finite-element stress analysis computer program. The stresses, including discontinuity stresses induced by the combination of external pressure or internal pressure, dead load, live load, thermal effects and seismic loads, were evaluated. GE's analyses and limits for the resulting stress intensities are in accordance with Subarticles NE-3130 and NE-3200 of ASME Code Section III, Division 1 (1989 Edition). The staff found that the analysis and design approach and the use of design code are consistent with the guidelines of SRP Section 3.8.1. This is acceptable. GE also stated in the SSAR that the compressive stress within the knuckle region caused by the internal pressure and the compression in other regions caused by other loads are limited to the allowable buckling stress values in accordance with Subarticle NE-3222 of ASME Code Section III, Division 1 (1989 Edition). GE's evaluation of the potential buckling of the drywell head and the buckling criteria used are consistent with the staff position for shell buckling described in Appendix E of this

report. This is acceptable. To follow the 10 CFR Part 52 design certification process, change to the use of ASME Code (1989 Edition) for the design, fabrication, and construction of drywell head against buckling would involve an unreviewed safety question and, therefore, require NRC review and approval prior to implementation. Furthermore, any requested change to the use of this code must either be specifically described in the COL application or submitted for license amendment after COL issuance.

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

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

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

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

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

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

Design of Structures, Components, Equipment, and Systems

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

2. GE has met the requirements of GDC 2 by designing the concrete containment to withstand a 0.3g SSE with sufficient margin, and the combinations of the effects of normal and accident conditions with the effects of environmental loadings such as earthquakes and other natural phenomena.
3. GE has met the requirements of GDC 4 by assuring that the design of the concrete containment is capable of withstanding the dynamic effects associated with missiles, pipe whipping, and discharging fluids.
4. GE has met the requirements of GDC 16 by designing the concrete containment so that it is an essentially leaktight barrier to prevent the uncontrolled release of radioactive effluent to the environment.
5. GE has met the requirements of GDC 50 by designing the concrete containment to accommodate, with sufficient margin, the design leakage rate, calculated pressure and temperature conditions resulting from accident conditions, and by assuring that the design conditions are not exceeded during the full course of the accident condition. In meeting these design requirements, GE has used the recommendations of regulatory guides and industry standards indicated below. GE has also performed an appropriate analysis that demonstrates that the ultimate capacity of the containment will not be exceeded and establishes acceptable margin of safety for the design.

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

The use of these criteria as defined by applicable codes, standards, guides, and specifications; the loads and loading combinations; the design and analysis procedures; the structural acceptance criteria; the materials, quality control programs, and special construction techniques; and the

testing and inservice surveillance requirements, provide reasonable assurance that, in the event of winds, tornados, earthquakes and various postulated accidents occurring within and outside the containment, the structure will withstand the specified design conditions without impairment of structural integrity or safety function of limiting the release of radioactive material.

3.8.2 Steel Components of the Reinforced Concrete Containmentment

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

3.8.3 Concrete and Steel Internal Structures of Steel or Concrete Containmentment

In the ABWR design, the internal structures inside the containment include the reinforced-concrete diaphragm floor, the reactor pedestal, the reactor shield wall, and other structural components. The diaphragm floor separates the upper drywell from the suppression pool. The reactor pedestal consists of a ledge on a cylindrical shell that forms the reactor cavity, extending from the bottom of the diaphragm to the top of the containment foundation slab. The space enclosed by the cylindrical shell under the reactor is the lower drywell, which is connected to the suppression pool through a series of vertical and horizontal vents in the shell wall. A steel equipment platform is located in the lower drywell and is accessible through a steel personnel tunnel and a steel equipment tunnel from outside the containment. Other internal structures include the DEPSS and the miscellaneous floors. The major code used in the design of concrete internal structures is American Concrete Institute (ACI) Standard 349 (1980 Edition). The use of this code for the design of the seismic Category I reinforced concrete structures is acceptable, except that the staff position on the design requirements for the steel embedments should be satisfied. The staff position on steel embedment design is described in Appendix F of this report. As described in early SSAR Amendments, GE used American National Standards Institute/American Institute of Steel Construction (ANSI/AISC) Standard N-690 (1984 Edition) for the design of all steel internal structures. In the DFSE, the staff noted that this standard had not been approved and accepted by the staff, the use of ANSI/ASCE N-690 was DFSE Open Item 3.8.3-1. In SSAR Section 3.8.3 and Tables 3.8-4 and 3.8-9, Amendment 33, GE provided limitations of using the ANSI/AISC N-690 Standard (1984 Edition) in the design of steel internal structures. The use of this standard with these limitations complies with the staff position described in Appendix G of this report. This is

Design of Structures, Components, Equipment, and Systems

acceptable, and DFSER Open Item 3.8.3-1 is resolved. To follow the 10 CFR Part 52 design certification process, any change to the use of ANSI/AISC Standard N-690 (1984 Edition) and ACI Standard 349 (1980 Edition) for the design and construction of containment internal structural elements would involve an unreviewed safety question and, therefore, require NRC review and approval prior to implementation. Furthermore, any requested change to the use of these codes must either be specifically described in the COL application or submitted for license amendment after COL issuance.

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

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

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

The materials of construction and their fabrication, construction, and installation are in accordance with ACI Standard 349 (1980 Edition) and ANSI/AISC Standard

N-690 (1984 Edition) for the concrete and steel structures, respectively. In the DFSER, the staff stated the concern about the use of the edition of ASME Code. This was DFSER Open Item 3.8.3-2. In Amendment 33 of the SSAR, GE specified that the 1989 version of ASME Code will be used for the design. This is acceptable, and DFSER Open Item 3.8.3-2 is resolved.

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

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

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

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

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

2. GE has met the requirements of GDC 2 by designing the containment internal structure to withstand the 0.3g SSE with sufficient margin and the combinations of the effects of normal and accident conditions with effects of environmental loadings such as earthquakes and other natural phenomena.
3. GE has met the requirements of GDC 4 by assuring that the design of the internal structures is capable of withstanding the dynamic effects associated with missiles, pipe whipping, and discharging fluids.
4. GE has met the requirements of GDC 50 by designing the containment internal structures to accommodate, with sufficient margin, the design leakage rate, calculated pressure, and temperature conditions, resulting from accident conditions, and by assuring that the design conditions are not exceeded during the full course of accident conditions. In meeting these design requirements, GE has used the recommendations of RGs and industry standards indicated below in the next paragraph. GE has also performed an appropriate analysis that demonstrates the ultimate capacity of the structures and establishes acceptable margin of safety for the design.

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

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

3.8.4 Other Seismic Category I Structures

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

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

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

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

Design of Structures, Components, Equipment, and Systems

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

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

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

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

In the early SSAR amendments, GE did not account for the effect of the hydrodynamic load on the RB resulting from a safety/relief valve (SRV) discharge or a LOCA in the containment. Because the RB encloses and is structurally integral with the containment shell at each floor level and at the spent fuel grider, the effect of the hydrodynamic load on the RB as a result of a SRV discharge or a LOCA in the containment should be factored into the design. This was DFSER Open Item 3.8.4-2. In SSAR Appendix 3G, Amendment 33, GE specified that the safety-related SSCs as applicable are also analyzed and

designed for the dynamic excitations originating from the event of operational transients and LOCA. The input loads considered in the structural dynamic response analysis include condensation oscillation (CO), pool chugging (CH), horizontal vent chugging (HV), SRV discharge, and annulus pressurization (AP). The staff's evaluation about the adequacy of these loads described in SSAR Appendix 3B, Amendment 33, is discussed in Section 6 of this report. In the structural analyses, these loads were classified as pipe nozzle break loads, symmetric loads, and asymmetric loads. A multi-stick lumped-mass model that represents the RB, RCCV, reactor shield wall and pedestal, and RPV was used for the analysis. The damping values recommended in RG 1.61 were considered for SRV and LOCA loads. For the pipe nozzle break loading cases, multi-input excitation modal time-history analyses were performed. For the symmetric and asymmetric loading cases, a frequency response method was used. In SSAR Appendix 3G, GE provided the structural responses (accelerations and displacements) and the FRS owing to the hydrodynamic loads applied on the RB as a result of an SRV discharge and a LOCA. From review of the SSAR and the audit conducted on February 22 through 24, 1993, the staff concludes that the structural responses, including the FRS generated and the structural design (RB) against these loads, are reasonable. DFSER Open Item 3.8.4-2 is resolved.

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

Design of Structures, Components, Equipment, and Systems

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

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

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

and appendices in Chapter 3 of the SSAR. This was DFSER Open Item 3.8.4-4. During the audit on February 22 through 25, 1993, the staff reviewed GE's seismic reanalysis results and design calculations and found that the resolution for this open item is acceptable. DFSER Open Item 3.8.4-4 is resolved.

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

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

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

Design of Structures, Components, Equipment, and Systems

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

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

1. GE has met the requirements of Section 50.55a and GDC 1 with respect to assuring that the safety-related structures other than containment are designed, fabricated, erected, and constructed to quality standards commensurate with its safety function to be performed by meeting the guidelines of regulatory guides and industry standards indicated below.
2. GE has met the requirements of GDC 2 by designing the safety-related structures other than containment to withstand the 0.3g SSE with sufficient margin and the combinations of the effects of normal and accident conditions with the effects of environmental loadings such as earthquakes and other natural phenomena.
3. GE has met the requirements of GDC 4 by assuring that the design of the safety-related structures is capable of withstanding the dynamic effects associated with missiles, pipe whipping, and discharging fluids.
4. GE has met the requirements of Appendix B to 10 CFR Part 50 because their QA program provides adequate measures for implementing guidelines relating to structural design audits.

The criteria used in the analysis, design, and construction of all the plant seismic Category I structures to account for anticipated loadings and postulated conditions that may be imposed upon each structure during its service lifetime are in conformance with established criteria, codes, standards, and specifications acceptable to the regulatory staff. These include meeting the guidelines of RG 1.69, Revision 0,

RG 1.91, Revision 1, RG 1.94, Revision 1, RG 1.115, Revision 1, RG 1.142, Revision 1, and RG 1.143, Revision 1, and industry standards ACI-349 and ANSI/AISC N-690, "Specifications for the Design, Fabrication, and Erection of Steel Safety-Related Structures for Nuclear Facilities."

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

3.8.5 Foundations

The ABWR design employs separate reinforced-concrete mat foundations for major seismic Category I structures. The RB foundation, which is integral with the containment foundation, supports the containment structure, reactor pedestal, other internal structures, and the balance of RB structure. Even though the containment structure foundation is integral with the RB foundation, it is a portion of the foundation within the perimeter of the containment structure. Therefore, the foundation was designed as a part of the containment boundary. The concrete foundations were designed to resist various combinations of dead loads, live loads, environmental loads (including winds, tornados, and SSE), and loads generated by postulated ruptures of high-energy pipes. The original detailed design information such as the factor of safety against overturning for the RB was calculated and provided in SSAR Appendix 3H, Amendment 4. However, in the DFSER, the staff noted that no such information was given in the SSAR for the control building and the radwaste building substructure. This was DFSER Open Item 3.8.5-1. In addition, the staff also noted that if foundation waterproofing is used, the COL applicant should evaluate the capability of the foundations to transfer seismic shear forces. This was DFSER COL Action Item 3.8.5-1. In SSAR Appendix 3H, Amendment 33, GE provided the loading condition used for the dynamic overturning, sliding and flotation analyses of the RB, control building, and radwaste building and the evaluation results (safety factors against overturning, sliding, and flotation) of these three buildings. From review of the SSAR Appendix 3H and the design calculation audit conducted on February 22 through 25, 1993, the staff concludes that the reactor, control, and radwaste buildings

Design of Structures, Components, Equipment, and Systems

will be dynamically stable under the specified SSE and DFSER Open Item 3.8.5-1 is resolved. As for COL Action Item 3.8.5-1, which deals with the potential for the foundation mat to slide over the plastic sheets that are generally used as waterproofing material, SSAR Sections 3.8.5.4 and 3.8.6.1, Amendment 33, stated that the COL applicant will evaluate the capability of the foundation to transfer shear loads where foundation waterproofing material is used. This is acceptable. In addition, based on the current practice of foundation construction and the staff's understanding, a layer of gravel will be placed on the excavated foundation surface for the soil site and the excavated rock foundation surface will be roughened before pouring concrete and placing the waterproofing material. The treated foundation surface will increase the friction between the structural foundation and the supporting foundation surface. Therefore, the staff concludes that the treated foundation surface will be capable to transfer the seismic shear loads.

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

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

During the first two design calculation audits, the staff found that the detailed design calculations for the found-

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

1. GE has met the requirements of Section 50.55a and GDC 1 with respect to assuring that the seismic Category I foundations are designed, fabricated, erected, constructed, tested and inspected to quality standards commensurate with its safety function to be performed by meeting the guidelines of regulatory guides and industry standards indicated below.
2. GE has met the requirements of GDC 2 by designing the seismic Category I foundation to withstand the 0.3g SSE with sufficient margin and the combinations of the effects of normal and accident conditions with the effects of environmental loadings such as earthquakes and other natural phenomena.
3. GE has met the requirements of GDC 4 by assuring that the design of seismic Category I foundations is capable of withstanding the dynamic effects associated with missiles, pipe whipping, and discharging fluids.

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

Design of Structures, Components, Equipment, and Systems

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

3.8.6 Certified Design Material

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

3.9 Mechanical Systems and Components

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

3.9.1 Special Topics for Mechanical Components

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

supports, as well as experimental and inelastic analytical techniques.

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

- (1) ASME Service Level A - normal conditions
- (2) ASME Service Level B - upset conditions, incidents of moderate frequency
- (3) ASME Service Level C - emergency conditions, infrequent incidents
- (4) ASME Service Level D - faulted conditions, low-probability postulated events
- (5) testing conditions

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

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

Design of Structures, Components, Equipment, and Systems

program developers and the programs are verified by one or more of the methods recommended in SRP Section 3.9.1.

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

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

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

(1) Postulated Pipe Rupture

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

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

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

- A guillotine failure of the ASME Class 1 weld is assumed. Subsequent to this event, the stretch tube, which normally functions to hold the pump diffuser in place, is the first member to resist ejection of the pump housing. On the basis of an elastic analysis, the stresses in the stretch tube are calculated to be 85 percent of its minimum specified ultimate strength. The stretch tube could be reasonably considered to mitigate this event without failure, however, this failure scenario is continued as discussed below.
- In the event that the stretch tube also fails subsequent to the weld failure, the RIP shaft and the thrust bearing are subjected to the ejection load. The weakest link in this remaining load path is the bearing-to-shaft bolt. An elastic analysis of this bolt also results in stresses that are less than its ultimate strength. Therefore, this bolt would not be expected to fail.
- If the weld, stretch tube, and bearing-to-shaft bolt all sequentially fail, then the external vertical restraints are subjected to the ejection loads. These restraints are stainless steel rods that run from lugs on the vessel to lugs on the RIP motor cover. The structural integrity of the attachment lugs, bolts, and rod clevises under this loading condition is demonstrated by elastic analysis. The use of inelastic analysis methods is limited to the middle slender body of the rod. The allowable strain used in this analysis is identical to that used for pipe whip restraints and is acceptable as discussed in Item 1 above.

Design of Structures, Components, Equipment, and Systems

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

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

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

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

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

GE has met Appendix B to 10 CFR Part 50, and GDC 1 by submitting information that demonstrates the applicability and validity of the design methods and computer programs used for the design and analysis of seismic Category I structures designated as ASME Code, Class 1, 2, 3, and CS and those not covered by the Code within the present state-of-the-art limits and by having design control measures that are consistent with the applicable guidelines of SRP 3.9.1. This is acceptable for

ensuring the quality of the computer programs. If the COL applicant opts to use computer programs different from those used by GE for the design of any safety-related item with the exception of piping systems, the guidelines of SRP 3.9.1 must be met for such programs. The staff's review of the requirements for piping systems is included in its evaluation of the design acceptance criteria (DAC) and ITAAC for generic piping designs in Section 3.12 of this report.

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

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

3.9.2.1 Piping Pre-operational Vibration and Dynamic Effects Testing

Piping vibration, thermal expansion, and dynamic effects testing will be conducted during a pre-operational testing program. The purpose of these tests is to ensure that the piping vibrations are within acceptable limits and that the piping system can expand thermally in a manner consistent with the design intent. During the plant's pre-operational and startup testing program, which is also described in SSAR Section 14.2.12, all ABWR plants will test various piping systems for abnormal, steady-state, or transient vibration and for restraint of thermal growth. Systems to be monitored will include (1) ASME Code, Class 1, 2, and 3 piping systems; (2) high-energy piping systems inside seismic Category I structures; (3) high-energy portions of systems whose failure could reduce the functioning of seismic Category I plant features to an unacceptable safety level; and (4) seismic Category I portions of moderate-energy piping systems located outside the containment. Steady-state vibration, whether flow induced or caused by nearby vibrating machinery, could cause up to 10^{10} cycles

of stress in the pipe during the 60-year design life of the plant. For this reason, the staff requires that the stresses associated with steady-state vibration be minimized and limited to acceptable levels. The test program will consist of a mixture of instrumented measurements and visual observations by qualified personnel. In SSAR Section 3.9.2.1 and Table 1.8-21, GE indicates that detailed test specifications will be prepared in accordance ANSI/OM-1987, Part 3, and ANSI/OM-1986, Part 7.

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

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

3.9.2.2 Seismic Subsystem Analysis

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

The ABWR plant is designed for an SSE ground motion defined by a RG 1.60, "Design Response Spectra for Seismic Design of Nuclear Power Plants," Revision 1,

response spectrum anchored to a peak ground acceleration of 0.3g. Amplified building response spectra are generated for the ABWR plant to account for varying soil properties in the United States by enveloping 14 site conditions. The enveloping amplified building response spectra provided in the SSAR are used for the design and analyses of the ABWR piping systems.

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

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

For the dynamic analysis of seismic Category I piping, each system is idealized as a mathematical model

Design of Structures, Components, Equipment, and Systems

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

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

evaluation of this issue is discussed in Section 3.12.4.2 of this report.

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

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

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

When analyzing piping systems, the size of the mathematical model might exceed the capacity of the computer program when large and small bore piping are included. Thus, the small bore branch lines are generally decoupled from the large bore main piping. Originally, the SSAR did not provide any criteria for the decoupling of the piping systems in the analysis model. However, in a letter to the NRC dated February 24, 1992, GE provided a decoupling criteria in a GE draft document entitled "ABWR SSAR Main Steam, Feedwater and SRVDL Piping Systems Design Criteria and Analysis Methods," Revision 0, dated February 1992. In this document, GE stated that when the ratio between pipe diameters of the branch line to main line is less than one-third, the branch line can be excluded from the piping model of the main line. Since GE is using this criterion for all piping systems in the ABWR plant, the staff requested the basis

for the 1:3 ratio. In addition, GE was requested to define how the mass effect of the decoupled line is accounted for in the model of the main line and how the frequency ratio effect (or resonant amplification of the main line) is accounted for in the modeling and analysis of the branch line. In the DFSER, the staff noted that GE should revise its SSAR to include this information for the staff's review (DFSER Open Items 3.9.2.2-4 and 14.1.3.3.4.4-1). In Amendment 23 to the SSAR, GE revised Sections 3.7.3.3.1.3, 3.7.3.3.1.4, and 3.7.3.8.1.9 to respond to these open items. The staff's evaluation of this issue is discussed in Section 3.12.4.4 of this report.

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

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

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

The staff's evaluation of this issue is discussed in Section 3.12.5.17 of this report.

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

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

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

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

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

Design of Structures, Components, Equipment, and Systems

transients. These tests are discussed in more detail in the rest of this section.

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

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

By letter dated May 10, 1992, the staff documented a summary of this issue, which was addressed in an audit conducted at GE on February 10 through 12, 1992. As noted in the summary, GE stated that Kashiwazaki Kariwa Unit 6 (K-6), which is currently under construction in Japan, contains reactor internals that are almost identical to those in the ABWR design. During the audit, GE presented a set of documents related to the flow-induced vibration assessment program for the K-6 reactor internals.

This information consists of an analysis for vibration prediction, the basis and details of instrumentation for vibration monitoring, specifications for the installation and removal of the monitoring system, and specifications for conducting the pre-operational and startup tests. In addition, GE provided a description of a full-scale, 60-degree flow test of the ABWR RIP system that was conducted in Japan.

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

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

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

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

- (2) Precritical testing with fuel - This vibration measurement series is conducted with the reactor assembly complete but before reactor criticality. Flow conditions include balanced, unbalanced, and transient conditions as for the first test series. The purpose of this series is to verify the anticipated effect of the fuel on the vibration response of internals.
- (3) Initial startup testing - Vibration measurements are made during reactor startup at conditions up to 100 percent of rated flow and power. Balanced, unbalanced, and transient conditions of recirculation system operation are evaluated. The primary purpose of this series is to verify the anticipated effect of two-phase flow on the vibration response of internals.

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

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

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

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

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

tubes. Access is provided to the reactor lower plenum for these inspections.

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

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

Design of Structures, Components, Equipment, and Systems

3.9.2.4 Dynamic System Analysis of Reactor Internals Under Faulted Conditions

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

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

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

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

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

The staff concludes that the ABWR dynamic system and component analysis meets the applicable portions of GDC 2 and 4 and SRP Section 3.9.2 with respect to the design of systems and components important to safety to withstand the effects of earthquakes and the appropriate combinations of the effects of normal and postulated accident conditions with the effects of the SSE by a

dynamic system analysis which provides an acceptable basis for confirming the structural design adequacy of the reactor internals and unbroken piping loops to withstand the combined dynamic loads of a postulated LOCA and the SSE and the combined loads of a postulated main steamline rupture and the SSE. The analysis provides adequate assurance that the combined stresses and strains in the components of the CRD system and reactor internals will not exceed the allowable design stress and strain limits for the materials of construction and that the resulting deflections or displacements at any structural element of the reactor internals will not distort the reactor internals geometry to the extent that core cooling may be impaired. The staff finds the methods used for component analysis to be compatible with those used for the system analysis. The combination of component and system analyses is acceptable.

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

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

3.9.3.1 Loading Combinations, Design Transients, and Stress Limits

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

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

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

As stated earlier, the staff believes the margins built into the ASME fatigue design curves may not be sufficient to

account for variations in the original fatigue test data as a result of various environmental effects. Therefore, the staff's position is that environmental effects should be considered in the fatigue analysis for ASME Code Class 2 and 3 piping. GE should include in its SSAR the proposed approach for accounting for the environmental effects in these fatigue analyses. In the DFSEER, this was identified as a part of DFSEER Open Item 3.9.3.1-1 for ASME Class 1 components. The resolution of this issue is discussed in Section 3.12.5.7 of this report.

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

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

In accordance with NRC Bulletin 88-08, the staff is requesting that licensees and applicants review systems

Design of Structures, Components, Equipment, and Systems

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

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

During an audit conducted at GE offices in San Jose, California, on March 23 through 26, 1992, the staff asked

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

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

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

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

Design of Structures, Components, Equipment, and Systems

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

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

3.9.3.1.1 Intersystem LOCA Design for Piping Systems

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

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

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

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

- (1) The likelihood of rupture (burst) of the pressure boundary is low, based on a goal of approximately 10 percent failure probability for rupture;
- (2) The likelihood of intolerable leakage of flange joints or valve bonnets is reasonably low although some leakage might occur;
- (3) Some piping components might undergo gross yielding and permanent deformation.

Low-Pressure Piping Design

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

Table 3.2 Margins for straight pipe

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

Design of Structures, Components, Equipment, and Systems

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

Valves in Low-Pressure Systems

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

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

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

Other Components in Low-Pressure Systems

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

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

ISLOCA Conclusion

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

As stated in SECY-90-016, for those low-pressure systems for which designing to withstand full reactor pressure is not practical, the design provides (1) the capability for leak testing of the pressure isolation valves, (2) valve position indication that is available in the control room when isolation valve operators are deenergized, and (3) high-pressure alarms to warn control room operators when rising RCS pressure approaches the design pressure of attached low-pressure systems and both isolation valves are not closed.

Using these design guidelines, the staff concludes that--

- (1) the likelihood of the low-pressure piping rupturing under full RCS pressure is low,
- (2) the likelihood of intolerable leakage is low under ISLOCA conditions although some leakage may occur at flanges and valve bonnets, and
- (3) some piping components might undergo gross yielding and permanent deformation under ISLOCA conditions.

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

3.9.3.1.2 Conclusion

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

3.9.3.2 Design and Installation of Pressure-Relief Devices

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

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

tested, the valves will be tested in accordance with Item II.D.1. This is acceptable.

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

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

3.9.3.3 Component Supports

The staff reviewed SSAR Sections 3.9.3.4 and 3.9.3.5 with regard to the methodology used in the design of ASME Code Class 1, 2, and 3 component supports. The review included an assessment of the design and structural integrity of the supports. It addressed three types of supports: plate and shell, linear, and component standard

Design of Structures, Components, Equipment, and Systems

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

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

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

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

for the design of structural steel for use as pipe supports and is acceptable.

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

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

- (1) information that addresses the types of snubbers and their characteristics (as delineated in SRP 3.9.3, Section II.3b) to be used in the ABWR standard plant
- (2) information that addresses the use of seismic restraints other than snubbers (i.e., special engineered pipe supports such as energy absorbers and limit stops, and their modeling assumptions)
- (3) information that addresses the pipe support stiffness values and support deflection limits used in the piping analyses
- (4) information that addresses how the seismic excitation of the pipe supports (especially large frame-type structures) are to be considered in the design of the pipe support anchorage

Design of Structures, Components, Equipment, and Systems

- (5) information that addresses the hot and cold gaps to be used between the pipe and the box-frame-type of support and the coefficient of friction to be used for considering friction forces between the pipes and the steel frames
- (6) criteria that will ensure that the maximum deflections of the piping at support locations for static and dynamic loadings are within an allowable limit to preclude failure of the pipe supports and hangers

In the DFSER, all of these items constituted Open Items 3.9.3.3-1 and 14.1.3.3.6-1. The staff's evaluations of these open items are discussed in Sections 3.12.6.5, 3.12.6.6, 3.12.6.7, 3.12.6.8, 3.12.6.10, 3.12.6.11, and 3.12.6.13 of this report.

The staff noted in the DFSER that GE had not provided any information on the design criteria for the structural design of instrumentation line supports. The staff requested that this information be included in the SSAR. This was DFSER Open Item 3.9.3.3-2 and part of DFSER Open Item 14.1.3.3.6-1. SSAR Subsections 3.7.3.8.1.9 and 3.9.3.4.1 contain GE's response to these open items. The staff's evaluation of this response is discussed in Section 3.12.6.12 of this report.

The industry has taken the position that ANS/AISC N-690 is useful in the design of instrumentation sensing line supports. Its use would have the effect of reducing the QA recordkeeping requirements and code stamping required by Subsection NF of ASME Section III. The staff's position on this issue is that for construction of ASME component supports, ANS/AISC N-690 alone is not an acceptable standard; ASME Code, Section III, Subsection NF, should be used. However, the staff is currently participating in the ASME effort to incorporate N-690 into Subsection NF. Subsequent to a staff-endorsed version of NF incorporating N-690, Subsection NF will also specify the rules acceptable to the staff for construction of ASME Class supports. In the DFSER, the staff stated that when this staff-approved version is available, the COL applicant seeking to use it may submit a request to the staff for approval on a plant-specific basis. This was DFSER COL Action Item 14.1.3.3.6.12-1. The staff's evaluation of this issue is discussed in Section 3.12.6.12 of this report.

On the basis of the evaluation in Section 3.9.3.3, supplemented by the evaluations in applicable portions of Section 3.12.6, of this report, the staff concludes that GE meets the requirements of 10 CFR 50.55a and GDC 1, 2, and 4 with regard to the design and service load combinations and associated stress and deformation limits specified for ASME Code, Class 1, 2, and 3 component supports by

insuring that component supports are designed to quality standards commensurate with their importance to safety, and that these supports can accommodate the effects of normal operation as well as postulated events such as LOCAs and the dynamic effects resulting from the safe shutdown earthquake. The combination of loadings (including system operating transients) considered for each component support within a system, including the designation of the appropriate service stress limit for each loading combination, has met the positions and criteria of RGs 1.124 and 1.130 and are in accordance with NUREG-0484. The specified design and service loading combinations used for the design of ASME Code, Class 1, 2, and 3 component supports in systems classified as seismic Category I provide assurance that in the event of an earthquake or other service loadings due to postulated events or system operating transients, the resulting combined stresses imposed on system components will not exceed allowable stress and strain limits for the materials of construction. Limiting the stresses under such loading combinations provides a conservative design basis to assure that support components can withstand the most adverse combination of loading events without loss of structural integrity.

The staff's evaluation of Class CS components is given in Section 3.9.5 of this report.

3.9.4 Control Rod Drive Systems

The staff's review under SRP Section 3.9.4 included the control rod drive system (CRDS) up to its interface with the control rods. Those components of the CRDS that are part of the primary pressure boundary are classified as SC 1, QG A, and are designed according to ASME Code, Section III, Class 1 requirements and to the quality assurance requirements of 10 CFR Part 50, Appendix B. The CRDS will be capable of reliably controlling reactivity changes either under conditions of anticipated normal plant operational occurrences or under postulated accident conditions. The CRDS in the ABWR design consists of fine motion control rod drive mechanisms and the control rod drive hydraulic system. The staff reviewed the information in SSAR Section 3.9.4 related to the criteria used to ensure the structural integrity of this system during normal operation and under accident conditions. These criteria conform to SRP Section 3.9.4 and are acceptable. Loading combinations for the CRDS are discussed in Section 3.9.3.1 of this report. The functional design and testing of these systems is discussed in Section 4.6 of this report.

The staff's review acceptance criteria are based on meeting (1) GDC 1 and 10 CFR Part 50, Subsection 50.55a requiring that the CRDS be designed to quality standard

Design of Structures, Components, Equipment, and Systems

commensurate with the importance of the safety functions to be performed; (2) GDC 2 requiring that the CRDS be designed to withstand the effects of an earthquake without loss of capability to perform its safety functions; and (3) GDC 14 requiring that the RCPB portion of the CRDS be designed, constructed, and tested for the extremely low probability of leakage or gross rupture.

The staff concludes that the design of the control rod drive system is acceptable for the ABWR and meets GDC 1, 2, and 14, and 10 CFR 50.55a. As stated in the first paragraph in this section, by designing the CRDS up to its interface with the control rods to acceptable loading combinations of normal operation and accident conditions using ASME Class 1 and 10 CFR Part 50, Appendix B, requirements, GE has assured the structural integrity of the CRDS. Therefore, GE meets GDC 1 and 10 CFR 50.55a with regard to designing components important to safety to quality standards commensurate with the importance of the safety functions to be performed. In addition, GE meets GDC 2 and 14 with regard to designing the control rod drive system to withstand the effects of earthquakes and anticipated normal operation occurrences with adequate margins to ensure its structural integrity and functional capability and with an extremely low probability of leakage or gross rupture of the RCPB. The staff's evaluations of the specified design transients, design and service loadings, and combinations of loads, are discussed in Sections 3.9.1 and 3.9.3.1 of this report. By limiting the stresses and deformations of the CRDS under such loading combinations, the design conforms to the appropriate guidelines in SRP Sections 3.9.3 and 3.9.4.

3.9.5 Reactor Pressure Vessel Internals

The staff reviewed, in accordance with SRP Section 3.9.5, the load combinations, allowable stress and deformation limits, and other criteria used in the design of the reactor internals. The staff's review acceptance criteria are based on meeting (1) GDC 1 and 10 CFR Part 50, Subsection 50.55a requiring that the reactor internals shall be designed to quality standards commensurate with the importance of the safety functions to be performed; (2) GDC 2 requiring that the reactor internals shall be designed to withstand the effects of earthquakes without loss of capability to perform its safety functions; (3) GDC 4 requiring that reactor internals shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operations, maintenance, testing, and postulated LOCA; and (4) GDC 10 requiring that reactor internals shall be designed with adequate margins to assure that specified acceptable fuel design limits are not exceeded during anticipated normal operational occurrences. SSAR Section 3.9.5.3.5 states that the core support structures for the ABWR are designed and

constructed in accordance with ASME Code, Section III, Subsection NG. In accordance with Subsection NG-1100, this means that the manufacture and installation of the ABWR core support structures are in accordance with the NG rules required for materials, design, examination, and preparation of reports. This conforms to SRP Section 3.9.5 and is acceptable.

SSAR Section 3.9.5.3.6 gives the design bases for safety class reactor internals other than the core support structures. The design criteria, loading conditions and analyses that provide the basis for the design of these components meet the guidelines of ASME Code, Section III, Subsection NG-3000. These components are constructed so as not to adversely affect the integrity of the core support structures as required by ASME Code, Section III, Subsection NG-1122. These criteria conform to SRP Section 3.9.5 and are acceptable.

In accordance with SSAR Table 3.2-1, the core support structures and all other safety-related reactor internals are designed as Safety Class 2 components and to the quality assurance requirements of 10 CFR Part 50, Appendix B. In addition, as discussed in Section 3.9.1, "Special Topics for Mechanical Components," Section 3.9.2.4, "Dynamic System Analysis of Reactor Internals Under Faulted Conditions," and Section 3.9.3.1, "Loading Combinations, Design Transients, and Stress Limits," of this report, the SSAR contains acceptable criteria for the design of all safety-related reactor internals under normal, upset, emergency, and faulted loading conditions. Implementation of these criteria to the design of the reactor internal structures and components provides reasonable assurance that, in the event of an earthquake or of a system transient during normal plant operation, the resulting deflections and associated stresses imposed on these structures and components will not exceed allowable stresses and deformations under such loading combinations. This provides an acceptable design basis for ensuring that these structures and components will withstand the most adverse loading events that were postulated to occur during their service lifetime without loss of structural integrity or impairment of function.

The staff concludes that the design of reactor internals for the ABWR is acceptable and meets GDC 1, 2, 4, and 10 and 10 CFR 50.55a.

On the basis of these evaluations related to designing all safety-related reactor internals (1) as Safety Class 2; (2) to the quality assurance requirements of 10 CFR Part 50, Appendix B; and (3) to acceptable ASME Code, Section III, rules, GE meets GDC 1 and 10 CFR 50.55a with regard to designing the reactor internals to quality

standards commensurate with the importance of the safety functions to be performed.

On the basis of these evaluations related to designing all safety-related reactor internals to acceptable loading combinations and stress limits when the internals are subjected to the loads associated with normal, upset, emergency, and faulted conditions, GE meets GDC 2, 4, and 10 with respect to designing components important to safety to withstand the effects of earthquakes and the effects of normal operation, maintenance, testing, and postulated LOCAs with sufficient margin to ensure that their capability to perform their safety functions is maintained and the specified fuel design limits are not exceeded.

3.9.6 Testing of Pumps and Valves

SRP Section 3.9.6 provides guidance for review of inservice testing (IST) of certain safety-related pumps and valves typically designated as ASME Code Class 1, 2, or 3. The staff's review acceptance criteria are based on meeting (1) GDC 37 as related to periodic functional testing of the ECCS to assure the leak tight integrity and performance of its active components; (2) GDC 40 as related to periodic functional testing of the containment heat removal system to assure the leak tight integrity and performance of its active components; (3) GDC 43 as related to periodic functional testing of the containment atmospheric cleanup systems to assure the leak tight integrity and the performance of the active components, such as pumps and valves; (4) GDC 46 as related to periodic functional testing of the cooling water system to assure the leak tight integrity and performance of the active components; (5) GDC 54 as related to piping systems penetrating containment being designed with the capability to test periodically the operability of the isolation and determine valve leakage acceptability; and (6) 10 CFR Part 50, Subsection 50.55a(f) as related to including pumps and valves whose function is required for safety in the inservice testing program to verify operational readiness by periodic testing.

In Section 3.9.3 of this report, the staff discusses the design of safety-related pumps and valves for the ABWR. The load combinations and stress limits used in the design of pumps and valves ensure that the integrity of the component pressure boundary will be maintained. In addition, a licensee will periodically test the performance and measure performance parameters of safety-related pumps and valves in accordance with ASME Code Section XI, as required by 10 CFR 50.55a(f). Periodic measurements of various parameters will be compared to baseline measurements to detect long-term degradation of the pump or valve performance. The tests, measurements, and

comparisons will ensure the operational readiness of these pumps and valves. However, as discussed in SECY-90-016, the staff determined that ASME Code Section XI requirements do not assure the necessary level of component operability that is desired for evolutionary LWR designs. Accordingly, in SECY-90-016, as supplemented by the staff's April 27, 1990, response to comments by the ACRS, the staff recommended criteria to the Commission to be used to supplement Section XI of the ASME Code. In its SRM of June 26, 1990, on SECY-90-016, the Commission approved the staff's recommendations. The staff's proposed applicable regulations, as described in Section 1.6 of this report, for inservice testing are as follows.

All pumps and valves of the standard design subject to the test requirements set forth in 10 CFR 50.55a(f) are subject to the following additional limitations and modifications.

- (1) Piping design must incorporate provisions for full flow testing at maximum design flow of pumps and check valves;
- (2) Check valve testing must incorporate the use of advanced non-intrusive techniques to address degradation and performance characteristics;
- (3) Provisions must be established to determine the frequency necessary for disassembly and inspection of pumps and valves to detect unacceptable degradation that cannot be detected through the use of advanced non-intrusive techniques; and
- (4) Provisions must be incorporated to test motor-operated valves under design basis differential pressure.

The staff's evaluation of GE's responses to ASME Code Section XI issues, including the above applicable regulations, is discussed in the following sections.

SSAR Section 3.9.6 states that IST of safety-related pumps and valves will be performed in accordance with the requirements of ASME/ANSI OMa-1988 Addenda to ASME/ANSI OM-1987, Parts 1, 6, and 10. SSAR Table 1.8-21 also indicates that the applicable code of record for the ABWR is the ASME Code 1989 Edition. ASME/ANSI Part 6 of Operations and Maintenance (OM), "Inservice Testing of Pumps," and Part 10, "Inservice Testing of Valves," are referenced in Section XI, ASME Code 1989 Edition. The 1988 Addenda and the 1989

Design of Structures, Components, Equipment, and Systems

Edition of Section XI have been incorporated by reference into 10 CFR 50.55a and are acceptable for the evolutionary LWR IST provided the analysis of leakage rates and corrective action requirements of OM Part 10, paragraph 4.2.2.3 are applied to containment isolation valve testing. The staff finds the containment isolation valve leak testing to be acceptable as discussed in Section 3.9.6.2.4 of this report.

SSAR Table 3.9-8 lists the IST parameters and frequencies for safety-related pumps and valves. This table was provided as part of the design certification and contains the information regarding the ability to test pumps and valves in accordance with the ASME Code. The staff's assessment of the IST plan is contained in Section 3.9.6.3 of this report. However, the development of a complete plant-specific IST program will be the responsibility of the COL applicant. The comprehensive pump and valve IST program will provide additional information beyond that contained in the IST plan. The IST program will include the tests performed on each pump and valve and the Code requirement met by each test; test parameters and frequency of the tests; the normal, safety, and fail-safe position of each valve; component type for each pump and valve; and P&ID coordinates for each pump and valve. In addition, the COL applicant will submit any requests for relief, which the NRC staff will review on the basis of the ASME Code edition referenced in 10 CFR 50.55a(f), the ABWR design, and the inservice testing methods available at the time of the COL application. This was DFSE COL Action Item 3.9.6.3-1. SSAR Section 3.9.7.3 includes this action item and states that the COL applicant will provide a detailed pump and valve IST plan. This is acceptable.

GE's description of the primary elements of the inservice testing plan for design certification in SSAR Section 3.9.6 are discussed below.

3.9.6.1 Testing of Safety-Related Pumps

In response to the staff's concern regarding the adequacy of design and qualification for safety-related pumps, GE stated in SSAR Section 3.9.6.1 that for each safety-related pump, the design basis and required operating conditions (including tests) under which the pump will be required to function will be established. GE further stated that the COL applicant will establish the design and qualification requirements and will provide acceptance criteria for these requirements. For each size, type, and model the COL applicant will perform testing encompassing design conditions that demonstrate acceptable flow rate and corresponding head, bearing vibration levels, and pump internals wear rates for the operating time specified for

each system mode of pump operation. From these tests the COL applicant will also develop baseline hydraulic and vibration data for evaluating the acceptability of the pump after installation. The staff finds that the planned actions provide a reasonable assurance for the adequacy of the design and qualification for safety-related pumps and are, therefore, acceptable.

In response to the staff's policy concern about the adequacy of minimum-flow systems for safety-related pumps, GE states in SSAR Section 3.9.6.1 that safety-related pumps and piping configurations can accommodate inservice testing at a flow rate at least as large as the maximum design flow for the pump. This commitment is responsive to the guidelines on this issue contained in SECY-90-016 and meets Item 1 of the staff's recommendations, as an applicable regulation, on the pump and valve inservice testing issues as described in Sections 1.6 and 3.9.6 of this report. This is acceptable. In addition, GE also stated in the SSAR that it would evaluate the sizing of each minimum recirculation flow path to ensure that its use under all analyzed conditions will not result in degradation of the pump and to periodically measure the minimum recirculation flow rate to verify that it is in accordance with the design specification. In response to another staff request, GE states in the same SSAR section that it would provide the safety-related pumps with instrumentation to verify that the net positive suction head (NPSH) is greater than or equal to the NPSH required during all modes of pump operation. The staff finds this acceptable because they meet SRP 3.9.6 testability requirements for safety-related pumps.

In Section 3.9.6.1 of the DSER (SECY-91-235), the staff identified an open item relating to disassembly and inspection of safety-related pumps (Outstanding Issue 14). In its letter dated December 19, 1991, GE stated that the COL applicant will develop a program to establish the frequency and the extent of disassembly and inspection for all safety-related pumps, including the basis for the frequency and the extent of each disassembly. GE further stated that the program may be revised throughout the plant life to minimize disassembly, based on past disassembly experience. Subsequent to these clarifications, the staff concluded this program to be the responsibility of the COL applicant. Therefore, Outstanding Issue 14 was closed and was reidentified as DFSE COL Action Item 3.9.6.1-1. SSAR Section 3.9.6.1 includes this action item. This action item is responsive to the guidelines on this issue contained in SECY-90-016 and meets Item 3 of the staff's recommendations, as an applicable regulation, on the pump and valve inservice testing issues as described in Sections 1.6 and 3.9.6 of this report. This is acceptable.

3.9.6.2 Testing of Safety-Related Valves

3.9.6.2.1 Check Valves

In response to the staff's concern about the adequacy of design and qualification for safety-related check valves, SSAR Section 3.9.6.2.1 states that for each check valve with active safety-related function, the design basis and required operating conditions (including testing) under which the check valve will be required to perform will be established. The SSAR further states that the COL applicant will establish the design and qualification requirements and will provide acceptance criteria for these requirements. By testing each size, type, and model, the COL applicant will ensure the design adequacy of the check valve under design conditions, including severe transient loadings expected during the life of the valve such as waterhammer or pipe break. This testing of each size, type, and model will include test data from the manufacturer, field test data for dedication by the COL applicant, empirical data supported by test, or test (such as prototype) of similar valves that support qualification of the required valve where similarity must be justified by technical data. The COL applicant will ensure that the maximum loading on the check valve under design basis and the required operating conditions is within the structural capability limits for the individual parts of the check valve. In addition, the COL applicant will ensure proper check valve applications in the piping system design. Specific considerations will include selection of valve size and type based on the system flow conditions, installed location of valve with respect to source of turbulence, and correct orientation of valve in the system (e.g., vertical versus horizontal) as recommended or required by the valve manufacturer. The qualification acceptance criteria will include baseline data developed during qualification testing and will be used for verifying the acceptability of the check valve after installation. Furthermore, during the pre-operational testing, the COL applicant will test each safety-related check valve in the open and/or closed direction, as required by the safety function, under all normal operating system conditions. The testing requirements and acceptance criteria, including leak-tightness, disk stability, and correct valve sizing, are provided in the SSAR. The staff finds these planned actions provide a reasonable assurance for the adequacy of the design, qualification, and pre-operational testing for safety-related check valves and are, therefore, acceptable.

In response to the staff's concern about the full-flow testing of check valves, SSAR Section 3.9.6.2.1 states that ABWR safety-related piping systems incorporate provisions for testing to demonstrate the operability of the check valves under design-basis conditions. In response to another staff request, the SSAR states that advanced non-

intrusive techniques will be used in the implementation of inservice testing program to periodically assess degradation and the performance characteristics of check valves. These are responsive to the guidelines on these issues contained in SECY-90-016 and meets Items 1 and 2 of the staff's recommendations, as an applicable regulation, on the pump and valve inservice testing issues as described in Sections 1.6 and 3.9.6 of this report and are, therefore, acceptable.

In Section 3.9.6.2.1 of the DSER (SECY-91-235), the staff identified an open item relating to disassembly and inspection of safety-related check valves (Outstanding Issue 15). In its letter dated December 19, 1991, GE stated that the COL applicant referencing the ABWR design will develop a program to establish the frequency and the extent of disassembly and inspection on the basis of suspected degradation of safety-related check valves, including the basis for the frequency and the extent of each disassembly. GE also stated that the program may be revised throughout the plant life to minimize disassembly based on past disassembly experience. Subsequent to these classifications, the staff determined this to be the responsibility of the COL applicant. Therefore, Outstanding Issue 15 was closed and was reidentified as DFSER COL Action Item 3.9.6.2.1-1. GE has included this action item in Section 3.9.6.2.1 of the SSAR. This action item is responsive to the guidelines on this issue contained in SECY-90-016 and meets Item 3 of the staff recommendations, as an applicable regulation, on the pump and valve inservice testing issue as described in Sections 1.6 and 3.9.6 of this report and is, therefore, acceptable.

3.9.6.2.2 Motor-Operated Valves

In response to the staff's concern regarding the adequacy of design and qualification for safety-related motor-operated valves (MOVs) SSAR Section 3.9.6.2.2 states that for each MOV with active safety-related function, the design basis and required operating conditions (including testing) under which the MOV will be required to perform are established for the development and implementation of the design, qualification, and pre-operational testing. For the design and qualification of MOVs, the SSAR provides commitments as follows. The COL applicant will establish the design and qualification requirements and will provide acceptance criteria for these requirements. The COL applicant will test each size, type, and model to determine the torque and thrust requirements to operate the MOV and will ensure the adequacy of the torque and thrust that the motor operator can deliver under design conditions. The COL applicant will also test each size, type, and model under a range of differential pressure and flow conditions up to the design conditions. These design conditions

Design of Structures, Components, Equipment, and Systems

include fluid flow, differential pressure (including pipe break), system pressure, fluid temperature, ambient temperature, minimum voltage, and minimum and maximum stroke time requirements. This testing of each size, type, and model will include test data from the manufacturer, field test data for dedication by the COL applicant, empirical data supported by test, or test of similar valves (such as prototype) that support qualification of the required valve where similarity must be justified by technical data. From this testing, the COL applicant will demonstrate that the results of testing under in situ or installed conditions can be used to ensure the capability of the MOV to operate under design conditions. The COL applicant will also ensure that the structural capability limits of the individual parts of the MOV will not be exceeded under design conditions. Furthermore, the COL applicant will ensure that the valve specified for each application is not susceptible to pressure locking and thermal binding. For the pre-operational testing of MOVs, the SSAR provides commitments as follows. The COL applicant will test each MOV in the open and closed directions under static and maximum achievable conditions up to design basis conditions, using diagnostic equipment that measures torque and thrust and motor parameters. The COL applicant will test the MOV under various differential pressure and flow up to maximum achievable conditions and perform a sufficient number of tests to determine the torque and thrust requirements at design conditions. The specific testing parameters and acceptance criteria for demonstrating that the adequacy of the MOV functional performance has been met are provided in the SSAR. The staff finds these commitments regarding the design, qualification, and pre-operational testing for safety-related MOVs provide a reasonable assurance for demonstrating the adequacy of the MOV capability for the design basis conditions and are, therefore, acceptable.

The commitments just discussed also relate to the ITAAC information applicable to MOVs discussed in Section 3.9.6.4 of this report. To follow the 10 CFR Part 52 design certification process, any change to these commitments would involve an unreviewed safety question and, therefore, requires NRC review and approval prior to implementation. Furthermore any requested change to these commitments must either be specifically described in the COL application or submitted for license amendment after COL issuance.

In the DFSER, the staff noted that the COL applicant should address the design, qualification, and pre-operational testing for the MOVs as discussed in SSAR Section 3.9.6.2.2, prior to plant startup. This was COL Action Item 3.9.2.2-1. SSAR Section 3.9.7.3 includes the above information. The staff finds this to be acceptable.

In the DSER (SECY-91-235), the staff identified an open item relating to justifying the use of prototype qualification testing of MOVs (Outstanding Issue 16). The staff's specific concern is the need for proper justification of the applicability of prototype test data. In response to the staff's concern, GE committed to revise the SSAR to reference GL 89-10, Supplement 1, staff responses to Q22 and Q24 through Q28, which contain staff guidelines to justify prototype testing. Therefore, Outstanding Issue 16 was resolved and was reidentified as DFSER Confirmatory Item 3.9.6.2.2-1. As described in SSAR Section 3.9.6.2.2, testing of MOVs of each size, type, and model will include test of similar valves (such as prototype) that support qualification of the required valve where similarity must be justified by technical data. The staff found that this commitment for prototype qualification testing is acceptable, and there is no need to reference GL 89-10, Supplement 1, for conducting MOV prototype qualification testing. As a result, DFSER Confirmatory Item 3.9.6.2.2 is withdrawn and closed.

In response to the staff's concern about the periodic testing for MOVs, GE stated that the SSAR will be revised to address the COL applicant's program to periodically test MOVs. In the DFSER, the staff noted that the COL applicant will determine the optional frequency for valve stroking during inservice testing. This was part of DFSER COL Action Item 3.9.6.2.2-2. SSAR Section 3.9.6.2.2 states that periodic testing per GL 89-10 Paragraphs D and J will be conducted under adequate differential pressure and flow conditions that allow a justifiable demonstration of continuing MOV capability for design-basis conditions. SSAR Section 3.9.7.3 also states that the COL applicant will determine the optimal frequency of this periodic verification of the continuing MOV capability for design basis conditions and will include this requirement in the development of the detailed IST program. This is responsive to the guidelines on the MOV testing contained in SECY-90-016 and meets Item 4 of the staff's recommendations, as an applicable regulation, on the pump and valve inservice testing issues as described in Sections 1.6 and 3.9.6 of this report and is acceptable. GE further stated that the ASME Code provides criteria limits for the test parameters identified in SSAR Table 3.9-8 for the ASME Code inservice testing.

In GL 89-10, the staff recommended that MOVs in a safety-related system should either be designed to prevent mispositioning or be subjected to qualification testing to demonstrate capability to recover from mispositioning. This was DFSER Confirmatory Item 3.9.6.2.2-2. The BWR Owners' Group subsequently submitted a backfit appeal on that GL 89-10 recommendation. The staff, with the assistance of Brookhaven National Laboratory, reviewed and reevaluated the issue, using probabilistic risk

assessment techniques, and determined that the recommendations for MOVs mispositioning is not necessary for BWRs. The staff subsequently issued GL 89-10, Supplement 4, and stated that the staff no longer considered the recommendations for inadvertent operation of MOVs to be within the scope of GL 89-10 for BWRs. Therefore, the DFSER Confirmatory Item 3.9.6.2.2-2 is withdrawn and resolved.

In the DSER (SECY-91-235), the staff identified an open item regarding disassembly and inspection of safety-related MOVs (Outstanding Issue 17). In its letter dated December 19, 1991, GE stated that the COL applicant referencing the ABWR design will develop a program to establish the frequency and the extent of disassembly and inspection based on suspected degradation of safety-related MOVs, including the basis for the frequency and the extent of each disassembly. GE further stated that the program may be revised throughout the plant life to minimize disassembly based on past disassembly experience. Subsequent to the above clarifications, the staff determined this program to be the responsibility of the COL applicant. Therefore, Outstanding Issue 17 was closed and was reidentified also as part of DFSER COL Action Item 3.9.6.2.2-2. GE has included this information in SSAR Section 3.9.6.2.2. This is responsive to the guidelines on this issue contained in SECY-90-016 and meets Item 3 of the staff's recommendations, as an applicable regulation, on the pump and valve inservice testing issues as described in Sections 1.6 and 3.9.6 of this report and is, therefore, acceptable.

SSAR Section 3.9.6.2.2 states that the inservice testing of MOVs will rely on diagnostic techniques that are consistent with the state of the art which will permit an assessment of the performance of the valve under actual loading conditions. This is responsive to the staff's guidelines on this issue and is, therefore, acceptable.

3.9.6.2.3 Power-Operated Valves

In response to the staff's concern about the adequacy of design and qualification for safety-related power-operated valves (POVs) other than MOVs, SSAR Section 3.9.6.2.3 states that for each POV with active safety-related function, the design basis and required operating conditions (including testing) under which the POV will be required to perform will be established. The SSAR further states that COL applicant will establish the design and qualification requirements and will provide acceptance criteria for these requirements. By testing each size, type, and model, the COL applicant will determine the force requirements to operate the POV and will ensure the adequacy of the force that the operator can deliver under design conditions. The COL applicant will also test each size, type, and

model under a range of differential pressure and flow conditions up to the design conditions. This testing of each size, type, and model will include test data from the manufacturer, field test data for dedication by the COL applicant, empirical data supported by test, or test of similar valves (such as prototype) that support qualification of the required valve where similar must be justified by technical data. From this testing, the COL applicant will demonstrate that the results of testing under in-situ conditions can be used to ensure the capability of the POV to operate under design conditions. The COL applicant will also ensure that the structural capability limits of the individual parts of the POV will not be exceeded under design conditions. The COL applicant will ensure that packing adjustment limits are specified for the valve for each application such that it is not susceptible to stem binding. SSAR Section 3.9.6.2.3 also states that during pre-operational testing, the COL applicant will test each POV in the open and closed directions under static and maximum achievable conditions, using diagnostic equipment that measures or provides information to determine total friction, stroke time, seat load, spring rate, and travel under normal pneumatic or hydraulic pressure and minimum pneumatic or hydraulic pressure. The COL applicant will test the POV under various differential pressure and flow up to maximum achievable conditions, including design basis conditions. The COL applicant will perform a sufficient number of tests to determine the force requirements at design conditions. The specific testing parameters and acceptance criteria for demonstrating that the adequacy of the POV functional performance has been met are provided in the SSAR. The staff finds that these commitments provide a reasonable assurance for the adequacy of the design, qualification, and pre-operational testing for safety-related POVs and, are therefore, acceptable.

SSAR Section 3.9.6.2.3 also states that ABWR safety-related piping systems incorporate provisions for testing to demonstrate the operability of the POVs under design-basis conditions. Inservice testing will incorporate the use of advanced non-intrusive techniques to periodically assess degradation and the performance characteristics of POVs. These are responsive to the staff's guidelines on these issues and are acceptable. The SSAR further states that a program will be developed by the COL applicant to establish the frequency and the extent of disassembly and inspection based on suspected degradation of safety-related POVs, including the basis for the frequency and the extent of each disassembly. The SSAR also states that the program may be revised throughout the plant life to minimize disassembly based on past disassembly experience and identified the corresponding COL applicant actions in SSAR Section 3.9.7.3. This is responsive to the guideline on this issue contained in SECY-90-016 and

Design of Structures, Components, Equipment, and Systems

meets Item 3 of the staff's recommendations, as an applicable regulation, on the pump and valve inservice testing issues as described in Sections 1.6 and 3.9.6 of this report. This is acceptable.

3.9.6.2.4 Isolation Valve Leak Tests

In response to the staff's concern about the leak-tight integrity of isolation valves, SSAR Section 3.9.6.2.4 states that the leak-tight integrity of each valve relied on to provide a leak-tight function will be verified. These valves include (1) pressure isolation valves that provide isolation of pressure differential from one part of a system to another or between systems, (2) temperature isolation valves whose leakage may cause unacceptable thermal stress fatigue or stratification in the piping and thermal loading on supports or whose leakage may cause steam binding of pumps, and (3) containment isolation valves that perform a containment isolation function, including valves that are not a part of the 10 CFR Part 50, Appendix J, Type C, testing program but whose leakage may cause loss of water inventory of a suppression pool. The staff's evaluations of SSAR Section 3.9.6.2.4 are as follows.

In the DSER (SECY-91-235), the staff noted a concern about the scope of containment isolation valves (Outstanding Issue 18). In response, GE stated that the scope of containment isolation valves will be in accordance with GDC 54 and agreed to amend the SSAR accordingly. This was DFSER Confirmatory Item 3.9.6.2.4-1. SSAR Section 3.9.6.2.4 includes this information. This is acceptable, and DSER Outstanding Issue 18 and DFSER Confirmatory Item 3.9.6.2.4-1 are resolved. The staff's evaluation of this issue is also discussed in Section 6.2.4 of this report.

In the DSER (SECY-91-235), the staff requested that GE comply with the analysis of leakage rates and corrective action requirements of ASME/ANSI OM Part 10, paragraph 4.2.2.3, for containment isolation valves (Outstanding Issue 19). In response, GE stated that leakage rate testing of containment isolation valves will be in accordance with ASME Code Section XI. However, this was not completely acceptable. OM Part 10, which is referenced in ASME Code Section XI, 1989 Edition, only requires containment isolation valves to be tested in accordance with Appendix J to 10 CFR Part 50 and does not require that corrective action be based on exceeding individual valve leakage limits. The staff's position is that the analysis of leakage rates and corrective action requirements of paragraph 4.2.2.3 in OM Part 10 are also applicable to containment isolation valves. Subsequently, the staff requested GE to revise its response in accordance with the staff's position and reidentified Outstanding

Issue 19 as DFSER Open Item 3.9.6.2.4-1. SSAR Section 3.9.6.2.4 provides the requirements for containment isolation valves leakage rate testing in accordance with the staff's position, which is acceptable, and DSER Outstanding Issue 19 and DFSER Open Item 3.9.6.2.4-1 are resolved.

Several safety systems connected to the RCPB have design pressures below the rated RCS pressure. Also, some systems that are rated at full reactor pressure on the discharge side of pumps have pump suction below RCS pressure. To protect these systems from RCS pressure, two or more isolation valves will be placed in series to form the interface between the high-pressure RCS and the low-pressure system. SSAR Table 3.9-9 provides a list of RCS pressure isolation valves. In the DSER (SECY-91-235), the staff reported that its review of Table 3.9-9 was in progress (Outstanding Issue 20). Subsequently, the staff reviewed the RCS pressure isolation valves list provided in the SSAR and determined that this list contains all of the RCS pressure isolation valves and was acceptable, and that DSER Outstanding Issue 20 was resolved.

In the DFSER, the staff also identified a concern about the periodic leak testing of all pressure isolation valves. The staff's position is that the leak tight integrity of these valves must be ensured by periodic leak testing to prevent exceeding the design pressure of the low-pressure systems. This was DFSER TS Item 3.9.6.2.4-1. In response, GE stated that the periodic leak rate testing of the RCS pressure isolation valves in Table 3.9-9 will be performed in accordance with the surveillance requirements of the ABWR TS. GE also stated that the final proposed ABWR TS, to be considered and approved under the design certification program, will reflect the relevant surveillance requirements of the new BWR Standard Technical Specifications. This was DFSER TS Item 3.9.6.2.4-1. Subsequently, GE included this information in the ABWR TS Section 3.4.4, "RCS Pressure Isolation Valve Leakage," dated November 12, 1992. The periodic leak rate testing requirements for ABWR RCS pressure isolation valves are the same as the relevant surveillance requirements of the new BWR Standard TS and are acceptable. As a part of the resolution of the intersystem loss-of-coolant accident (ISLOCA) issue, low pressure piping systems that interface with the RCPB are required to be designed to withstand full RCS pressure to the extent practicable. The staff's evaluation of the ISLOCA design for ABWR piping systems has been found to be acceptable as discussed in Section 3.9.3.1.1 of this report.

3.9.6.3 Review of SSAR Table 3.9-8, Inservice Testing Plan

As discussed previously in Section 3.9.6 of this report, GE made certain commitments regarding the testability of safety-related pumps and valves in the ABWR design. In SSAR Section 3.9.6, GE stated that Code testing flexibility in the ASME/ANSI OM Part 6 and Part 10 produced no need for relief requests. In SSAR Table 3.9-8, GE listed the inservice testing parameters and frequencies for safety-related pumps and valves. In the DSER (SECY-91-235), the staff identified the review of SSAR Table 3.9-8 as Outstanding Issue 13. However, in the DFSER, Outstanding Item 13 was replaced by DFSER Open Item 3.9.6.3-1, which is discussed below.

The staff's review of SSAR Table 3.9-8 to ensure that GE's commitments regarding the ability to test pumps and valves can be met resulted in a list of questions transmitted to GE in a letter dated May 4, 1992. The staff stated that those questions should not be used to determine a comprehensive list of problem areas in SSAR Table 3.9-8 and that GE should systematically review and revise its ABWR IST plan, emphasizing the design configuration to provide assurance that its commitment regarding the ability to test pumps and valves can be met. Subsequently, a meeting with GE was held on June 8, 1992, in San Jose, California. As a result of that meeting, GE made several commitments that included addressing those questions identified in the staff's letter of May 4, 1992; revising SSAR Table 3.9-8 as well as the associated P&IDs; and performing a systematic review of its IST plan. GE also indicated that some exceptions to the Code requirements may be needed after it completes a systematic review of the IST plan. For the staff review of any Code exceptions, GE committed to identify the Code requirement, to provide a basis to justify the need for the code exceptions, and to describe its proposed alternative testing method. The staff also informed GE that it would be unacceptable to leave for the COL phase certain aspects of the final design related to testability of pumps and valves. Specifically, details of the piping configuration related to additional lines, valves, and instrumentation must be provided before FDA. The development and submittal of an acceptable IST plan was DFSER Open Item 3.9.6.3-1.

In response to the staff's request, GE subsequently revised SSAR Table 3.9-8. The staff, with the assistance of Science Applications International Corporation (SAIC), reviewed and evaluated the revised ABWR IST Plan as presented in SSAR Table 3.9-8. On the basis of that evaluation, the staff determined that the ABWR pump and valve IST plan provided a reasonable assurance that GE's commitment as described in SSAR Section 3.9.6 regarding the ability to test pumps and valves can be met and is

acceptable, and that DFSER Open Item 3.9.6.3-1 is resolved. Details of that evaluation are included as Appendix I to this report. Appendix I consists of the Technical Evaluation Report prepared by SAIC that has been modified by the staff to reflect subsequent changes made to Table 3.9-8 after SSAR Amendment 20. Those changes were made as a result of later design changes as well as resolutions to the staff's concerns identified in the DSER (SECY-91-235).

In the DFSER, the staff also identified an open item that GE should establish criteria to be used by the COL applicant for developing pump and valve design specifications to ensure that these components are capable of performing their design-basis functions. This was DFSER Open Item 3.9.6.3-2. GE provided these criteria in Sections 3.9.6.1, 3.9.6.2.1, 3.9.6.2.2, and 3.9.6.2.3 of the SSAR. The staff has found the criteria for developing pump and valve design specifications to be acceptable as discussed in Sections 3.9.6.1, 3.9.6.2.1, 3.9.6.2.2, and 3.9.6.2.3 of this report, and DFSER Open Item 3.9.6.3-2 is resolved.

3.9.6.4 Certified Design Material

In the DFSER, the staff identified an Open Item 3.9.6.2.3-1 that requested GE to submit an acceptable generic ITAAC for demonstrating MOV capability. In response to the staff's request, GE submitted systems-specific ITAACs that include criteria applicable to MOVs. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Section 14.3 of this report, and DFSER Open Item 3.9.6.2.3-1 is resolved.

In Section 3.9.6.4 of the DFSER, the staff indicated the need for an acceptable ITAAC for POVs other than MOVs. The staff's specific concern was the inadequacy of the POV capability for the design-basis conditions. That was DFSER Open Item 3.9.6.4-1. In response to the staff's concern, GE provided information about the design, qualification, pre-operational and inservice testing requirements for safety-related POVs in SSAR Section 3.9.6.2.3. As discussed in Section 3.9.6.2.3 of this report, the staff finds that GE's commitments provide a reasonable assurance for demonstrating the adequacy of the POV capability for the design-basis conditions. Therefore, the staff subsequently determined that this particular ITAAC is not necessary. The staff also determined that although it was concluded that other POVs would not need Tier 1 treatment, the SSAR does not contain sufficient information on the design and qualification and on the pre-operational testing of other POVs. In response to the staff's concern, SSAR Section 3.9.6.2.3 includes this information. The staff's

Design of Structures, Components, Equipment, and Systems

evaluation of that issue has been found to be acceptable as discussed in Section 3.9.6.2.3 of this report. Furthermore, on the basis of past operational experience and staff's inspections, the staff requested GE to provide an ITAAC for check valves. In response to the staff's request, GE submitted system-specific ITAACs that include criteria applicable to check valves. The adequacy and acceptability of the ABWR check valves ITAAC is evaluated in Section 14.3 of this report, and DFSE Open Item 3.9.6.4-1 is resolved.

3.9.6.5 Conclusion

Based on the evaluations described above, the staff concludes that the pump and valve IST program described in the SSAR is acceptable and meets the requirements of GDC 37, 40, 43, 46, and 54 and 10 CFR 50.55a(f). This conclusion is based on the commitments made regarding the testability of safety-related pumps and valves in the ABWR design, which will ensure the leaktight integrity and the operational readiness to perform necessary safety functions throughout the life of the plant. The pump and valve testing will include baseline preservice testing and visual inspection for leaks and other signs of distress. The staff further concludes that the ABWR standard plant meets the Commission-approved staff positions for inservice testing of pumps and valves contained in SECY-90-016 and its applicable regulation for inservice testing of pumps and valves as discussed in Sections 1.6 and 3.9.6 of this report. This is acceptable. This conclusion is based on that the commitments made in SSAR Sections 3.9.6.1, 3.9.6.2.1, 3.9.6.2.2, and 3.9.6.2.3 for safety-related pumps and valves will ensure (1) the full-flow testing of pumps and valves; (2) the use of advanced non-intrusive techniques for check valve testing; (3) the development of a disassembly and inspection program for pumps and valves; and (4) the adequacy of the MOV capability for the design-basis conditions.

3.10 Seismic and Dynamic Qualification of Mechanical and Electrical Equipment

SSAR Sections 3.9.2.2 and 3.10 provide information on the seismic and dynamic qualification of safety-related mechanical and electrical equipment. Section 3.9.3.2 also contains information related to pump and valve operability assurance. This information includes

- rationale used to determine if tests, analyses, or combinations of both will be performed
- criteria used to define the seismic and other relevant dynamic load input motions

- the proposed demonstration of the adequacy of the qualification program

The staff's review acceptance criteria are based on meeting (1) GDC 1 and 30 as related to qualifying equipment to appropriate quality standards commensurate with the importance of the safety functions to be performed; (2) GDC 2 and Appendix A to 10 CFR Part 100 as related to qualifying equipment to withstand the effects of natural phenomena such as earthquakes; (3) GDC 4 as related to qualifying equipment being capable of withstanding the dynamic effects associated with external missiles and internally generated missiles, pipe whip, and jet impingement forces; (4) GDC 14 as related to qualifying equipment associated with the reactor coolant boundary so as to have an extremely low probability of abnormal leakage, or rapidly propagating failure and of gross rupture; and (5) Appendix B to 10 CFR Part 50 as related to qualifying equipment using the quality assurance criteria provided.

GE will use the seismic qualification methodology in Section 4.4 of GE report NEDE-24326-1 (proprietary) for both mechanical and electrical equipment. This program conforms to IEEE 323 as modified and endorsed by RG 1.89, "Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plant," Revision 1. The program also meets the criteria in IEEE 344 as modified by RG 1.100, "Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants," Revision 2. In Tables 1.8-20 and 1.8-21 of the SSAR, GE agrees to use RG 1.100, Revision 2, June 1988, and IEEE 344, 1987. Section 9 of IEEE 344, 1987, recognizes the use of "experience data" as a method for seismic qualification of equipment. As used in IEEE 344, experience data includes both seismic experience and previous qualifications. In accordance with RG 1.100, Revision 2, the method of qualification will be reviewed by the staff on a case-by-case basis. In Amendment 27 to the SSAR, GE revised Section 3.10.1.1, "Selection of Qualification Method" to permit the use of experience data for seismic qualification of seismic Category I instrumentation and electrical equipment. In a letter dated May 14, 1993, GE provided a markup of the SSAR, which further revised Section 3.10.1.1 and added Section 3.10.5.3 to state that if dynamic qualification of seismic Category I instrumentation or electrical equipment is accomplished by experience, the COL applicant will provide the following to the NRC for review and approval:

- identification of the specific equipment
- the details of the methodology and the corresponding experience data for each piece of equipment

This information is in SSAR Sections 3.10.1.1 and 3.10.5.3. The staff concludes that the above commitment is consistent with the applicable portion of RG 1.100, Revision 2, and is acceptable.

The staff reviewed NEDE-24326-1 (proprietary) and approved the qualification methodology therein in an SER sent to GE on October 23, 1983. In response to the staff's request for information in Q 271.2, GE stated that the methodology in NEDE-24326-1 (proprietary) also conforms to the above commitments as shown in Tables 1.8-20 and 1.8-21 of the SSAR. This response is acceptable.

The methodology in NEDE-24326-1 (proprietary), supplemented by the information in SSAR Sections 3.9.2.2, 3.9.3.2, and 3.9.6, provides test and/or analysis criteria used to demonstrate the operability of active pumps and valves (i.e., those ASME Class 1, 2, or 3 components that must perform a mechanical motion to shut down the plant or mitigate the consequences of a postulated event). The staff concludes that the criteria described are consistent with the guidelines in SRP 3.10 and applicable portions of SECY-90-016, and are acceptable. To provide a more detailed basis for the staff's conclusion, in the following list are applicable guidelines from SRP 3.10 together with references to commitments in the SSAR which address each guideline.

- (1) Tests and analyses are required to confirm the operability of all mechanical and electrical equipment during and after an earthquake of magnitude up to and including the SSE, and for all static and dynamic loads from normal, transient and accident conditions. Prior to SSE qualification, demonstrate that the equipment can withstand excitation less than the SSE without loss of structural integrity. Analyses alone, without testing, are acceptable as a basis for qualification only if the necessary functional operability of the equipment is assured by its structural integrity alone. When complete testing is impractical, a combination of tests and analyses is acceptable.

Equipment that has been previously qualified by means of tests and analyses equivalent to those described here are acceptable provided that proper documentation of such tests and analyses is submitted.

Commitments to most of the above criteria can be found in the SSAR Subsections 3.9.2.2, 3.9.3.2, and 3.10.1.1(B) and in NEDE-24326-1 (proprietary), Sections 4.4.2.5.1, 4.4.3.3 and 4.4.4.

In addition, consistent with staff positions on recently licensed plants, Sections 3.9.3.2.1.1 and 3.9.3.2.3.1.4 in the SSAR, Amendment 29, and the markups of SSAR Sections 3.9.1.4.5, 3.9.1.4.11, and 3.9.3.2.5.1.2, and Table 3.9-2, Footnote (7) in a letter dated May 11, 1993, provide commitments that operability of active ASME Class 1, 2, and 3 valves and Class 2 and 3 pumps is further assured by limiting the stresses to the material elastic limit when the component is subjected to (1) the combination of normal operating loads, (2) SSE and other RB vibration loads, and (3) dynamic system loads (LOCA). Specifically, the average membrane stress resulting from this faulted condition (Service Level D) loads is limited to 75 percent of the material yield stress, and the maximum membrane plus bending stress is limited to 110 percent of the yield stress. Implementation of this acceptance criteria will provide assurance that valve bodies or pump cases will not distort to the extent that operability of the component is impaired. SSAR Sections 3.9.1.4 and 3.9.3.2, and Table 3.9-2 include this information. This is acceptable.

- (2) Equipment should be tested in the operational condition. Operability should be verified during and/or after the testing, as applicable to the equipment being tested. Loadings simulating those of plant normal operation, such as thermal and flow-induced loading, if any, should be concurrently superimposed upon the seismic and other pertinent dynamic loading to the extent practicable. Particular attention should be paid, in operability qualification of mechanical equipment subjected to flow-induced loading, to incorporate degraded flow conditions such as those that might be encountered by the presence of debris, impurities, and contaminants in the fluid system. An example of this may be the operability of the containment sump pump recirculating water full of debris.

Commitments to most of the above criteria can be found in the SSAR Subsections 3.9.2.2 and 3.9.3.2 and in NEDE-24326-1 (proprietary), Sections 4.4.2.5, 4.4.2.5.1 and 4.4.2.5.2.

- (3) The characteristics of the required seismic and dynamic input motions should be specified by response spectrum or time history methods. These characteristics, derived from the structures or systems seismic and dynamic analyses, should be representative of the input motions at the equipment mounting locations.

Design of Structures, Components, Equipment, and Systems

Commitments to the above criteria can be found in the SSAR Subsection 3.7.3.1 and in NEDE-24326-1 (proprietary), Section 4.4.4.1.4.6.2.

- (4) For seismic and dynamic loads, the actual test input motion should be characterized in the same manner as the required input motion, and the conservatism in amplitude and frequency content should be demonstrated (i.e., the test response spectrum should closely resemble and envelope the required response spectrum over the critical frequency range).

Commitments to the above criteria can be found in the SSAR Subsection 3.9.2.2.1 and NEDE-24326-1 (proprietary), Section 4.4.2.5.3(b).

- (5) Since seismic and the dynamic load excitation generally have a broad frequency content, multi-frequency vibration input motion should be used. However, single frequency input motion, such as sine beats, is acceptable provided the characteristics of the required input motion indicate that the motion is dominated by one frequency (e.g., by structural filtering effects), or the anticipated response of the equipment is adequately represented by one mode, or in the case of structural integrity assurance, the input has sufficient intensity and duration to produce sufficiently high levels of stress for such assurance. Components that have been previously tested to IEEE 344-1971 should be reevaluated to justify the appropriateness of the input motion used, and requalified if necessary.

Commitments to these criteria can be found in the SSAR Subsection 3.9.3.2.3.1.4 and in NEDE-24326-1 (proprietary), Sections 4.4.2.5.3 and 4.4.2.5.6.

- (6) For the seismic and dynamic portion of the loads the test input motion should be applied to one vertical axis and one principal horizontal axis (or two orthogonal horizontal axes) simultaneously unless it can be demonstrated that the equipment response in the vertical direction is not sensitive to the vibratory motion in the horizontal direction, and vice versa. The time phasing of the inputs in the vertical and horizontal directions must be such that a purely rectilinear resultant input is avoided. An acceptable alternative is to test with vertical and horizontal inputs in-phase, and then repeat the test with inputs 180 degrees out-of-phase. In addition, the test must be repeated with the equipment rotated 90 degrees horizontally.

Components that have been previously tested to IEEE 344-1971 should be requalified using biaxial test input motions unless justification for using a single axis test input motion is provided.

Commitments to the above criteria can be found in the SSAR Subsection 3.9.3.2.3.1.4 and NEDE-24326-1 (proprietary), Section 4.4.2.5.4.

- (7) Dynamic coupling between the equipment and related systems, if any, such as connected piping and other mechanical components, should be considered. The fixture design should simulate the actual service mounting and should not cause any extraneous dynamic coupling to the test item. A commitment to this criteria can be found in the SSAR Subsections 3.9.2.2.1 and Item 7 of 3.10.1.1.
- (8) For pumps and valves, the loads imposed by the attached piping should be properly taken into account. To assure operability under combined loadings, the stresses resulting from the applied test loads should envelope the specified service stress limit for which the component's operability is intended. As discussed in this Section 3.10, the SSAR Subsections 3.9.3.2.1.1, 3.9.3.2.3.1.4, and 3.9.3.2.5.1.2 contain criteria which addresses this issue.
- (9) Selection of damping values for equipment to be qualified should be made in accordance with RG 1.61, "Damping Values for Seismic Design of Nuclear Power Plants," Revision 0, and IEEE 344-1987. Higher damping values may be used if justified by documented test data with proper identification of the source and mechanism. SSAR Sections 3.7, 3.9.2.2, 3.9.3.2, and 3.10.2 contain criteria that addresses this issue.
- (10) Section 3.10.2.1 of the SSAR states that the methodology for qualifying relays shall be such that testing is performed in both the open and closed positions.

NEDE-24326-1 (proprietary) provides qualification methodology only and contains no plant-specific information. In the DFSER, the staff noted that each COL applicant referencing this document should ensure that specific environmental parameters along with seismic and dynamic input response spectra are properly defined and enveloped in the methodology for its specific plant and implemented in its equipment qualification program. This was DFSER COL Action Item 3.10-1. SSAR Sections 3.9.3.2.3.2 and 3.9.3.2.5.2 of the SSAR state that

documentation will be prepared to clearly show that the criteria outlined in SSAR Section 3.9.3.2 to demonstrate active pump and valve operability has been satisfied. This will be included as a part of the certified stress report for the pump or valve assembly. In addition, SSAR Section 3.10.5 states that the COL applicants shall maintain equipment qualification records in a permanent file that shall be readily available for audit. These are acceptable actions. The staff will audit these files to review the results of tests and analyses that were performed to (1) ensure that the criteria in the SSAR were properly implemented, (2) ensure that adequate qualification was demonstrated for all equipment and their supports, and (3) verify that all applicable loads were properly defined and accounted for in the testing and analyses performed.

3.10.1 Conclusions

On the basis of these evaluations, the staff concludes that GE has defined appropriate seismic and dynamic qualification of mechanical and electrical equipment and pump and valve operability programs. These programs meet applicable portions of GDC 1, 2, 4, 14, and 30, Appendix B to 10 CFR Part 50, and Appendix A to 10 CFR Part 100 and are acceptable. This conclusion is based on the following:

SSAR Table 3.2-1 identifies all ABWR safety-related mechanical and electrical equipment as (1) safety Class 1, 2, or 3, (2) seismic Category I, and (3) designed to the quality assurance requirements of 10 CFR Part 50, Appendix B. As discussed in Sections 3.2.1 and 3.2.2 of this report, the staff concludes that Table 3.2-1 is acceptable. On the basis of these evaluations, the staff concludes that GE meets GDC 1, 30, and 10 CFR Part 50, Appendix B, as they relate to qualifying safety-related mechanical and electrical equipment to appropriate quality standards commensurate with the importance of the safety function to be performed.

The qualification program, which will be implemented for mechanical, instrumentation, and electric equipment meets the requirements and recommendations of IEEE 344-1987 and the regulatory positions of RGs 1.61, 1.89, 1.92, 1.100, and SRP 3.9.3, provides adequate assurance that such equipment will function properly under all imposed design and service loads, including the loadings imposed by the safe shutdown earthquake, postulated accidents, and loss-of-coolant accidents. On the basis of this program, complemented by the staff's evaluations of (1) seismic classifications in Section 3.2.1 of this report, (2) protection from external missiles and internally generated missiles in Section 3.5 of this report, (3) analyses to withstand dynamic effects of postulated pipe breaks in Section 3.6.2 of this report, and (4) loading combinations and stress

limits in Section 3.9.3.1 of this report, GE meets GDC 2, Appendix A to 10 CFR Part 100, GDC 4 and 14, as they relate to qualifying equipment to (1) withstand the effects of natural phenomena such as earthquakes, (2) be capable of withstanding the dynamic effects associated with external missiles, internally generated missiles, and pipe whip and jet impingement forces, and (3) demonstrate that equipment associated with the reactor coolant pressure boundary has a low probability of abnormal leakage, rapidly propagating failure, or gross failure.

To follow the 10 CFR Part 52 design certification process, any change to the commitments involving seismic and dynamic qualification of mechanical and electrical equipment discussed in Section 3.10 of this report would involve an unreviewed safety question and, therefore, requires NRC review and approval prior to implementation. Furthermore, any requested change to these commitments must either be specifically described in the COL application or submitted for license amendment after COL issuance.

3.10.2 Methods and Procedures of Analysis or Testing of Supports of Electrical Equipment and Instrumentation

SSAR Section 3.10.3, Amendment 33, described the procedures and criteria for the seismic qualification and design of the nuclear steam supply system (NSSS) electrical equipment supports; seismic Category I supports for battery racks, instrument racks, control consoles, cabinets, and panels; seismic Category I local instrument supports; and seismic Category I instrument tubing supports. GE provided the methods and criteria used for the design of the seismic Category I electrical raceway (cable trays and conduit) supports in SSAR Section 3.8.4.4.2, Amendment 33. The following covers only the staff's evaluation of the procedures and criteria for the design of the seismic Category I electrical raceway supports.

SSAR Section 3.10.3.2.2, Amendment 23, described the procedures and criteria for the design of the seismic Category I electrical raceway supports. GE used the response spectrum method to analyze the composite system of the electrical raceways and supports and calculate the seismic loads and the RB vibration (RBV) loads resulting from a safety relief valve (SRV) discharge or LOCA inside the containment. The input to the dynamic analysis is the seismic and RBV FRS generated for the supporting floor. In case the supports are attached to a wall or to two different locations, the input is the upper bound FRS envelope obtained by superimposing the FRS of both floors or locations. In addition, in many cases GE combined several FRS by superposition to generate an upper bound

Design of Structures, Components, Equipment, and Systems

FRS envelope as the input to facilitate the design. The staff found the analysis methods and approaches for the design of the cable trays, conduit, and their supports acceptable.

However, SSAR Section 3.8.4.4.2, Amendment 33, revised the analysis methods by stating that all seismic Category I cable trays and conduit supports are designed by one of the methods discussed in SSAR Subsection 3.7.3 or by the design-by-rule methods as approved by the NRC. On the basis of the staff's review and evaluation discussed in Section 3.9.2.2 of this report, the methods provided in SSAR Section 3.7.3 are acceptable for the analysis and design of the cable trays, conduit, and their supports. As for the use of the design by rule methods, SSAR Section 3.7.3.8.2, Amendment 33, stated that for distributive systems such as cable trays, conduit, and HVAC ducts, an alternative to qualification by analysis described in SSAR Subsection 3.7.3.8.1 is the design-by-rule method approved by the NRC at the time of COL application. This is also acceptable to the staff. The basis to accept the use of the design-by-rule method is provided as follows.

According to SSAR Section 3.8.4.2.4, Amendment 33, the design of seismic Category I electrical raceway supports uses codes, standards, and specifications applicable to the building structures to which they are attached. These codes include ANSI/AISC Standard N-690 (1984 Edition), AISI SG-673, "Specification for the Design of Cold-Formed Steel Structural Members," and National Electrical Manufacturers Association (NEMA), "Fittings and Supports for Conduit and Cable Assemblies." The supports are designed and located to withstand the dynamic loads generated from the analyses in three directions by means of vertical, transverse, and longitudinal support and bracing systems. As discussed in SSAR Section 3.8.4.3.3, Amendment 32, the design considers the dead loads, live loads, and seismic loads plus other RBV dynamic loads. SSAR Section 3.8.4.4.3.1, Amendment 33, also discussed two methods used in the analysis and design of cable tray supports:

- (1) Rigid support with flexible trays - In this method, trays were modeled as flexible elastic systems and analyzed by the response spectrum method. The resulting reactions were used for the design of the supports.
- (2) Flexible support with flexible trays - In this method, the composite system of trays and supports were modeled and analyzed by computer as a multidegree of freedom elastic system. The support motion was prescribed by the appropriate floor response spectrum. The resulting responses were used to obtain design loads for the supports.

Since the conduit systems are more flexible and have comparatively less dead load, a rigid support approach (Method (1) above) applied for the cable tray design was used.

According to the staff review of SSAR Section 3.8.4, Amendment 33, and the design audits conducted by the staff, the supports, including those for the non-seismic Category I cable trays and conduits, were designed to meet seismic Category I requirements. These design criteria and procedures meet the guidelines of SRP Section 3.8.4 and are acceptable. In the early amendments of the SSAR, GE did not provide the design procedures and criteria for the seismic Category I cable trays and conduit. This was DFSER Open Item 3.10.3-1. SSAR Section 3.8.4, Amendment 33, provided the design procedures and criteria for the seismic Category I cable trays and conduit, which are acceptable. Therefore, DFSER Open Item 3.10.3-1 is resolved.

As for the concern about the potential interaction between the non-seismic Category I cable trays and conduit and the seismic Category I cable trays and conduit, SSAR Section 3.7.5.4 describes the COL applicant's action for the as-built plant assessment. The staff's review and evaluation are discussed in Section 3.7.2 of this report. On the basis of the previous discussion, the staff concludes that the procedures and criteria for the design of seismic Category I raceway supports are acceptable.

3.11 Environmental Qualification of Mechanical and Electrical Equipment

The staff reviewed the ABWR design environmental qualification requirements for mechanical and electrical equipment in accordance with SRP Section 3.11, Revision 2. Equipment that is used to perform a necessary safety function must be demonstrated to be capable of maintaining functional operability under all service conditions postulated to occur during its installed life for the time it is required to operate. This requirement, which is embodied in GDC 1 and 4 of Appendix A and Criteria III, XI, and XVII of Appendix B to 10 CFR Part 50, is applicable to equipment located inside as well as outside the containment. More detailed requirements and guidance related to the methods and procedures for demonstrating this capability for electrical equipment are in 10 CFR 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants;" NUREG-0588, Revision 1, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment," which supplements IEEE 323 and various NRC regulatory guides and industry standards, and RG 1.89, "Environmental Qualification of Certain Electric

Design of Structures, Components, Equipment, and Systems

Equipment Important to Safety for Nuclear Power Plants," Revision 1.

The NRC staff issued NUREG-0588 in December 1979 to promote a more orderly and systematic implementation of equipment qualification programs by industry and to provide guidance to the NRC staff for its use in ongoing licensing reviews. The portions in the NUREG series report provide guidance on (1) how to establish environmental service conditions, (2) how to select methods that are considered appropriate for qualifying equipment in different areas of the plant, and (3) other areas such as margin, aging, and documentation. A final rule on environmental qualification of electrical equipment important to safety for nuclear power plants became effective on February 22, 1983. This rule, 10 CFR 50.49, specifies the requirements to be met for demonstrating the environmental qualification of electrical equipment important to safety located in a harsh environment. RG 1.89, Revision 1 (June 1984), identifies the guidelines that have to be met for complying with this rule. In conformance with 10 CFR 50.49, electrical equipment for BWRs referencing the ABWR standard design may be qualified according to the criteria specified in Category I of NUREG-0588, Revision 1, and RG 1.89, Revision 1.

The qualification requirements for mechanical equipment are principally contained in Appendices A and B to 10 CFR Part 50. The qualification methods defined in NUREG-0588 can also be applied to mechanical equipment.

To document the degree to which the environmental qualification program for the ABWR complies with the NRC environmental qualification requirements and criteria, GE provided SSAR Section 3.11, "Environmental Qualification of Safety-Related Mechanical and Electrical Equipment," and SSAR Appendix 3I (proprietary), in a response dated January 13, 1989, to the staff's request for additional information dated September 12, 1988.

3.11.1 Completeness of Qualification of Electrical Equipment Important to Safety

The following three categories of electrical equipment important to safety must be qualified in accordance with the provisions of 10 CFR 50.49(b)(1), (b)(2), and (b)(3).

- safety-related electrical equipment (relied on to remain functional during and following design-basis events)
- non-safety-related electrical equipment whose failure under the postulated environmental conditions could prevent satisfactory accomplishment of the safety functions by the safety-related equipment

- certain post-accident monitoring equipment (Category I and II accident-monitoring instrumentation as specified in RG 1.97, Rev. 3)

In the SSAR, GE stated that the design of the information systems important to safety will be in conformance with the guidelines of Revision 3 of RG 1.97. However, the footnote for Subsection 50.49(b)(3) references Revision 2 for selection of the types of post-accident monitoring equipment. In issuing Revision 3, the NRC staff stated that the conformance with Revision 3 would not alter the implementation of Subsection 50.49. Therefore, conformance with Revision 2 is not required because conformance with Revision 3 meets the underlying purpose of the rule. As a result, an exemption from Subsection 50.49(b)(3) is justified by the special circumstances set forth in Subsection 50.12(a)(2)(ii).

In the early SSAR amendments, GE stated that for the ABWR, all three categories of electrical equipment mentioned above and located in a harsh environment will be environmentally qualified. GE also identified an interface requirement that requires COL applicants to list all electrical equipment within the scope of 10 CFR 50.49 in their plant-specific environmental qualification documents (EQDs). GE's approach for selecting and identifying electrical equipment required to be environmentally qualified for the ABWR was considered acceptable. However, following a further review, the staff determined that this item should be reclassified as a COL action item and that the staff will review specific details provided by the COL applicant to demonstrate compliance with 10 CFR 50.49(b)(1), (b)(2), and (b)(3). In the DFSER, the staff noted that the details will include a list of systems and their components that are included in the plant environmental qualification program and the design features for preventing the potential adverse consequence identified in IE Information Notice 79-22, "Qualification of Control Systems." This was DFSER COL Action Item 3.11.1-1. GE has included this action in SSAR Section 3.11.1. This is acceptable.

3.11.2 Qualification Methods

3.11.2.1 Electrical Equipment in a Harsh Environment

The environmental qualification program presented in GE Topical Report NEDE-24326-1 (proprietary) outlines the methodology to qualify NSSS system safety-related electrical equipment subject to a harsh environment. GE adopted this program for the ABWR (SSAR Section 3.11.2).

Design of Structures, Components, Equipment, and Systems

The staff reviewed the topical report and found that the qualification methodology conforms to 10 CFR 50.49 and its associated standards, except for the position on the time margin. NUREG-0588 states that the time margin for certain categories of equipment (these categories are identified in this NUREG report) should be a minimum of 1 hour. The topical report has not addressed this requirement. While GE addressed the time margin in SSAR Section 3.11.1, Amendment 14, the staff noted in the DSER (SECY-91-153) that the report made no reference to the 1-hour time margin requirement discussed in NUREG-0588. Therefore, the staff identified the time-margin issue as Outstanding Issue 12 in the DSER (SECY-91-153) and required it to be resolved in accordance with NUREG-0588, Revision 1 or as amplified in RG 1.89, Revision 1.

In response to this request, GE revised its position on time-margin in Section 3.11.1 of the SSAR, Amendment 17. However, it was not clear that the intent is to comply with the guidance of NUREG-0588, Revision 1, Category 1, paragraph 3, as amplified in RG 1.89, Revision 1, Regulatory Position C.4. Therefore, in the DFSE, the staff noted that GE should confirm that SSAR Section 3.11.1 will be updated to reflect compliance with this guidance. This was DFSE Confirmatory Item 3.11.2.1-1. Subsequently, GE addressed this issue in Section 3.11.1 of Amendment 32 to the SSAR by stating that: "Some mechanical and electrical equipment may be required by the design to perform an intended safety function within minutes of the occurrence of the event but less than 10 hours into the event. Such equipment shall be shown to remain functional in the accident environment for a period of at least 1 hour in excess of the time assumed in the accident analysis unless a time margin of less than 1 hour can be justified. Such justification will include for each piece of equipment: (1) consideration of a spectrum of breaks; (2) the potential need for the equipment later in the event or during recovery operations; (3) determination that failure of the equipment after performance of its safety function will not be detrimental to plant safety or mislead the operator; and (4) determination that the margin applied to the minimum, operability time, when combined with other test margins, will account for the uncertainties associated with the use of analytical techniques in the derivation of environmental parameters, the number of units tested, production tolerances, and test equipment inaccuracies." This is consistent with the staff position on time margin as stated in NUREG-0588, Revision 1, and is acceptable. Therefore, DFSE Confirmatory Item 3.11.2.1-1 is resolved. As discussed in the next section, the time-margin issue for mechanical equipment is similarly resolved.

3.11.2.2 Safety-Related Mechanical Equipment in a Harsh Environment

Although no detailed requirements exist for mechanical equipment, GDC 1 and 4 and Criteria III and XVII of Appendix B to 10 CFR Part 50 contain the following requirements related to equipment qualification:

- Components shall be designed to be compatible with the postulated environmental conditions, including those associated with LOCAs.
- Measures shall be established for the selection and review for the suitability of application of materials, parts, and equipment that are essential to safety-related functions.
- Design control measures shall be established for verifying the adequacy of design.
- Equipment qualification records shall be maintained and shall include the results of tests and materials analyses.

In the early SSAR amendments, GE stated that the qualification program for safety-related mechanical equipment for the ABWR design will include all safety-related mechanical equipment identified in SSAR Section 3.2. GE further stated that the mechanical equipment qualification program to be applied to the ABWR will use applicable portions of the NRC-approved Topical Report NEDE-24326-1 (proprietary) and RG 1.89, Revision 1, and will be consistent with the program for qualification of mechanical equipment in a harsh environment described in the NRC-approved GESSAR (GE SSAR) II design. The ABWR program scope looks not only at the metallic components of the equipment but also at the nonmetallic components. Metallic components that form a pressure boundary are considered to be qualified by the nature of their pressure retention capability as demonstrated by the application of an ASME Boiler and Pressure Vessel stamp. Nonmetallic components, such as greases, gaskets, and lubricants, will be shown to be capable of performing their intended functions under accident environments. The design of safety-related mechanical equipment associated with the ABWR will be performed under the same internal procedural controls as that used for the design of mechanical components associated with the GESSAR II design. These controls ensure that components are designed to be compatible with their postulated operating environments, that measures are established for the selection and review of the suitability of application of the material, parts, and equipment that are essential to safety-related functions, and that there are design control measures for verifying the adequacy of the design. As stated in NEDE-24326-1 (proprietary), a complete set of

qualification records are developed for each safety-related component.

In the early SSAR amendments, GE also identified an interface requirement that required COL applicants to provide in their plant-specific EQDs (1) a list of all safety-related mechanical equipment located in harsh-environment plant zones and (2) the methodology used to qualify the equipment located in harsh as well as mild-environment plant zones. However, following a further review, the staff determined that this item should be reclassified as DFSER COL Action Item 3.11.2.2-1. GE has included this action in Sections 3.11.1 and 3.11.6 of the SSAR. This is acceptable.

The staff concludes that the information GE provided on the selection and identification of mechanical equipment required to be environmentally qualified and the qualification methods for the equipment for the ABWR standard design is acceptable. The staff concludes that on the basis of SSAR Section 3.11.1, the time-margin issue as related to safety-related mechanical equipment is resolved similarly to that for the electric equipment discussed in the previous section.

3.11.3 Completeness of Information in Tables of SSAR Appendix 3I

SSAR Section 3.11 defines all the environmental conditions (normal, abnormal, test, accident, and post-accident) to which the applicable equipment may be exposed during plant operation. SSAR Appendix 3I contains the tables specifying the design limits or time-based profile of thermal environmental parameters (pressure, temperature, and relative humidity) and/or design limits for radiation environmental parameters (gamma dose rate and total gamma integrated dose) for each plant area or zone in the area under normal and/or abnormal and accident environmental conditions. The tables also include the neutron flux during normal operating conditions for different zones of the primary containment. The areas for which the environmental data are tabulated are the primary containment, secondary containment portion of the RB, remaining portions of the RB, turbine building, control building, radwaste building, service building, and outdoor area. Except for the radwaste building and the outdoor area, all other areas are further subdivided into zones on the basis of thermal and radiation environmental conditions determined for the zones. The SSAR considers a postulated RC (steam or water) pressure boundary pipe rupture as the limiting accident for calculating the design limits or the time-based profile for thermal environmental parameters during accident conditions for all zones. The

SSAR considers the design-basis LOCA as the limiting accident for calculating the design limits for radiation environmental parameters during accident conditions for all applicable zones. The total normal and accident doses in a zone is based on integrating the ambient dose rate in the zone over a 60-year period and the accident dose rate in the zone over a 6-month period, respectively.

GE states that the environmental conditions identified in the tables in Appendix 3I are upper-bound envelopes for these conditions in various areas or zones to which the applicable equipment has to be designed and qualified. The environmental parameters specified in these tables are for the upper-bound envelopes. GE states that the parameters do not include margins that may be required to satisfy equipment qualification requirements. GE further notes that these tables include identification of significant enveloping abnormal conditions and each enveloping accident event that affects the zone environment. GE provides these tables for use by COL applicants in developing their plant-specific environmental qualification programs for equipment important to safety. The staff reviewed the tables in Appendix 3I of the early SSAR amendments and found a number of deficiencies. They were collectively identified in the DSER (SECY-91-153) as Outstanding Issue 13. The reclassification of this issue to a number of DFSER open issues, confirmatory issues, and COL action items and corresponding resolutions are discussed below.

In the DSER (SECY-91-153), the staff noted that the tables in SSAR Appendix 3I did not include the chemical environmental conditions (chemical composition and the resulting pH) to which the applicable equipment may be exposed during accident conditions. Subsequently, GE responded that reactor water quality characteristics for the design-basis loss-of-coolant accident (DBLOCA) are contained in SSAR Section 3I.3.2.3. Additionally, in a facsimile dated June 1, 1991, GE stated that SSAR Section 3I.3.2.3 would be updated to include information on water quality characteristics for normal operations. This was DFSER Confirmatory Item 3.11.3-1. GE has included this information in SSAR Section 3I.3.2.3. The staff reviewed this information and found it acceptable. This resolved DFSER Confirmatory Item 3.11.3-1.

The staff notes that, in the SSAR, the table and figure numbers have been reassigned as shown in Table 3.3 of this report, from those in the previous SSAR amendments. The staff reviewed these reassignments and determined them acceptable. In the discussions below, the reassigned numbers are considered unless otherwise noted.

Design of Structures, Components, Equipment, and Systems

Table 3.3 Reassigned number for tables and figures

Previous No.	No.in SSAR Amnt. 32	Title of Table
3I.3-A	3I-1	Plant Environment Data and Location Cross Reference Table of Figure Numbers
3I.3-1	3I-2	Thermodynamic Environment Conditions Inside primary Containment Vessel, Plant Normal Conditions
3I.3-2	3I-3	Thermodynamic Environment Conditions Inside RB (Secondary Containment), Plant Normal Operation Conditions
3I.3-3	3I-4	Thermodynamic Environment Conditions Inside RB (Outside Secondary Containment), Plant Normal Operating Conditions
3I.3-4	3I-5	Thermodynamic Environment Conditions Inside Control Building, Plant Normal Operating Conditions
3I.3-5	3I-6	Thermodynamic Environment Conditions Inside Turbine Building, Plant Normal Operation Conditions
3I.3-9	3I-7	Radiation Environment Conditions Inside Primary Containment Vessel, Plant Normal Operating Conditions
3I.3-10	3I-8	Radiation Environment Conditions Inside RB (Secondary Containment), Plant Normal Operation Conditions
3I.3-11	3I-9	Radiation Environment Conditions Inside RB (Outside Secondary Containment), Plant Normal Operating Conditions
3I.3-12	3I-10	Radiation Environment Conditions Inside Control Building, Plant Normal Operation Conditions
3I.3-13	3I-11	Radiation Environment Conditions Inside Turbine Building, Plant Normal Operating Conditions
3I.3-14	3I-12	Thermodynamic Environment Conditions Inside Primary Containment Vessel, Plant Accident Conditions
3I.3-15	3I-13	Thermodynamic Environment Conditions Inside RB (Secondary Containment), Plant Accident Conditions
3I.3-16	3I-14	Thermodynamic Environment Conditions Inside RB (Outside Secondary Containment), Plant Accident Conditions
3I.3-18	3I-15	Thermodynamic Environment Conditions Inside Control Building, Plant Accident Conditions
3I.3-19	3I-16	Radiation Environment Conditions Inside Primary Containment Vessel, Design-basis accident
3I.3-20	3I-17	Radiation Environment Conditions Inside RB Design-basis accident (Secondary Containment)
3I.3-21	3I-18	Radiation Environment Conditions Inside RB Design-basis accident Conditions (Outside Secondary Containment)
3I.3-22	3I-19	Radiation Environment Conditions Inside Control Building Design-basis accident Conditions
Figure 3I.2-1	Figure 3I-1	Zones in Primary Containment Vessel

In the DSER (SECY-91-153), the staff noted that the tables in Appendix 3I did not include the beta radiation dose rate and the integrated beta dose for applicable zones. In response, GE stated that accurate radiation environments should include consideration of the source term and the design and location and materials of construction of the equipment in the various environmental zones. GE further stated that while the source term is known, the COL applicant will determine the design, specific location, and materials of construction of various pieces of equipment. As a result, GE developed an ITAAC for this issue ("Table 3.73.11C: Equipment Qualification for Radiation"). The staff reviewed the ITAAC and, in the DFSE, determined that the acceptance criteria should be modified to state: "The maximum expected lifetime exposure for each piece of equipment within the scope of 10 CFR 50.49 shall not exceed the demonstrated qualified value as determined in accordance with the requirements of 10 CFR 50.49 paragraph (f)." This was DFSE Open Item 3.11.3-1. In the SSAR, GE provided revised tables for Appendix 3I that included beta radiation dose rates and integrated beta doses for applicable zones. The revised tables replace Tables 3I.3-9 through 3I.3-13 and Tables 3I.3-19 through 3I.3-22. The staff reviewed the revised tables and found them acceptable. This resolved DFSE Open Item 3.11.3-1.

In the DSER (SECY-91-153), the staff noted that the Appendix 3I tables did not identify whether the subject zone is environmentally mild or harsh and also did not list the typical equipment located in each zone. In response, GE proposed a change to SSAR Section 3.11.2 to define a mild environment as: "Mild environment is that which, during or after a design-basis event will at no time be significantly more severe than that existing during normal and abnormal events." The staff understands that IST is included as a normal or abnormal condition (IST is not an environmental qualification program requirement). This proposed definition is consistent with the requirements of 10 CFR 50.49. In the DFSE, the staff noted that it would verify the incorporation of the proposed definition for a mild environment. This was DFSE Confirmatory Item 3.11.3-2. GE has included this information in SSAR Section 3.11.2. This is acceptable and resolved DFSE Confirmatory Item 3.11.3-2. However, for current generation operating reactors, the staff's definition of what constitutes a mild radiation environment for electronic components, such as semi-conductors or any electronic component containing organic materials, is different from what it is for other equipment. The staff's position is that a mild radiation environment for electronic equipment is a total integrated dose of < 10 Gy (10E3 R). For other equipment it is < 100 Gy (10E4 R). With the expected significant increase in the quantity and variety of electronic components in newer generation plants, the staff has

increasing concerns about the efforts being made to ensure that these components are environmentally qualified and the capability of the component to be environmentally qualified. As a result, in the DFSE, the staff commented that GE should confirm that its position on the environmental qualification of electronic components is consistent with the staff's. This was DFSE Open Item 3.11.3-2. In response, GE addressed this item in the SSAR by stating that electronic equipment subject to radiation exposure in excess of 1000 R and other equipment in excess of 10,000 R is qualified in accordance with 10 CFR 50.49. This is acceptable and resolved DFSE Open Item 3.11.3-2. In addition, to respond to DSER Outstanding Issue 13, GE proposed changes to Tables 3I.3-1 through 3I.3-22 to include references to P&ID and IED drawings that will identify typical equipment for each zone. In the DFSE, the staff found these proposed changes acceptable but noted that the changes should be incorporated into the SSAR. This was DFSE Confirmatory Item 3.11.3-3. The proposed changes are incorporated into the SSAR. Therefore, DFSE Confirmatory Item 3.11.3-3 is resolved.

In the DSER (SECY-91-153), the staff noted that the environmental conditions during abnormal plant operational conditions were placed under abnormal/accident conditions in the Appendix 3I tables. Additionally, the staff was not certain whether GE considered the adverse environmental conditions resulting from abnormal events such as SRV discharges and loss of non-safety-related HVAC and their durations in developing the environmental data for these tables for applicable zones. In response, SSAR Amendment 14 revised the Appendix 3I tables to properly include these abnormal occurrences in the normal plant operating conditions that are used for determining the qualified life of the equipment required to be qualified. This is acceptable.

In the DSER (SECY-91-153), the staff noted that the Appendix 3I tables did not explicitly identify the limiting accident (e.g., high-energy line break such as MSL, reactor core isolation cooling, RHR, or CUW line break and DBLOCA inside the containment) for each zone (e.g., steam tunnel, RHR pump room) that results in the most severe environment, particularly thermal, in the zone. SSAR Section 3.11.1 states: "The environmental conditions shown in the Appendix 3I tables are upper-bound envelopes used to establish the environmental design and qualification bases of safety-related equipment. The upper-bound envelopes indicate that the zone data reflects the worse case expected environment produced by a compendium of accident conditions." The staff interprets that the SSAR considers a spectrum of break sizes and the mass and energy releases from the considered break sizes and thereby developed the environmental qualification

Design of Structures, Components, Equipment, and Systems

profiles based on the most limiting combination of the considered break sizes. This is acceptable.

In the DSER (SECY-91-153), the staff noted that the Appendix 3I tables did not contain information on the environmental conditions resulting from spray or submergence or on the consequent wetting of equipment in applicable zones arising from piping failures nor the duration of the spray or submergence. In response, GE indicated in SSAR Section 3I.3.2.3 that containment spray may continue up to 100 days. In addition, GE modified SSAR Section 3I.3.2.3 by stating that "equipment will be qualified for submergence or will not be submerged except where submergence is mitigated by safety function performed by barrier separated redundant equipment." In the DFSER, the staff noted that during construction, the COL applicant should ensure that issues identified in Information Notice 89-63 related to flooding above the flood level and equipment wetting are addressed. This was DFSER COL Action Item 3.11.3-1. GE has included this information in Section 3.11.1 of the SSAR. This is acceptable.

In the DSER (SECY-91-153), the staff noted that the Appendix 3I tables did not contain (1) radiation environmental data under normal plant operating conditions for the radwaste building, outdoor area, and control building zones; (2) thermal environmental data under accident conditions for the radwaste building, service building zones, and outdoor area; and (3) radiation environmental data under accident conditions for the turbine building zones, radwaste building, service building zones, and outdoor area. In the DSER (SECY-91-153), the staff also indicated that if some of these areas identified are not expected to house any equipment required to be qualified and therefore not requiring environmental data, this should be stated. Subsequently, GE identified the various environmental zones within the scope of 10 CFR 50.49. However, the radiation environment that would result from normal operations and a design-basis accident was not determined. GE further commented that an accurate determination of radiation levels in the various harsh environmental zones requires consideration of specific equipment design details such as geometry and materials of construction and equipment location within each zone; therefore, radiation dose and dose rates should be determined by the COL applicant. This determination should also include the radiation contribution from recirculation fluid lines near the applicable areas. In the DFSER, the staff evaluated GE's position discussed above and stated that it would evaluate the relevant radiation zones on a plant-specific basis. This was DFSER COL Action Item 3.11.3-2. However, upon further evaluation, GE decided to include the relevant radiation zone data, including doses and dose rates, in the tables in

Appendix 3I. This information is included in the SSAR, which is acceptable, and DFSER COL Action Item 3.11.3-2 is deleted.

In the DSER (SECY-91-153), the staff noted that the Appendix 3I tables did not contain sufficient information on thermal environmental conditions (e.g., duration of different conditions) in various zones under normal plant operating conditions to develop a meaningful time-based thermal environmental profile for the zones. GE provided a proposed table (Table 3I.3-A) and a proposed amendment to all the tables in Appendix 3I that contain thermodynamic environmental conditions for both normal operating conditions and design-basis accidents. In the DFSER, the staff noted that the proposed information would be sufficient to develop time-based profiles for the various identified zones. This was DFSER Confirmatory Item 3.11.3-4. GE has included this information in Appendix 3I of the SSAR. This is acceptable and resolves DFSER Confirmatory Item 3.11.3-4.

In the DSER (SECY-91-153), the staff noted inconsistencies in the units used to specify the pressures (e.g., kg/cm², mm Aq). Consistent units are provided in the SSAR. This is acceptable.

In the DSER (SECY-91-153), the staff noted that the meaning of the statement "the pressure will be kept negative or positive" is not clear (see Note 2 to Tables 3I.3-3 through 3I.3-7). This statement was subsequently interpreted by the staff, in the DFSER, to mean that in various areas of the plant site, such as the control building, the RB, and the primary containment building, where the atmospheric pressure may be required to be negative that it will be maintained below 0.0 kPag (0.0 psig) and if the atmospheric pressure is required to be positive, that it will be maintained above 0.0 kPag (0.0 psig).

The staff noted in the DSER (SECY-91-153) that the integrated gamma accident dose in the primary containment for the ABWR was given as 6×10^5 Gy (6×10^7 rads), which is less than the typical value of about 2×10^6 Gy (2×10^8 rads) quoted in the safety analysis reports of several operating reactors (e.g., Perry: 2.7×10^6 Gy (2.7×10^8 rads); River Bend: 1.7×10^6 Gy (1.7×10^8 rads); Clinton: 2.0×10^6 Gy (2×10^8 rads); Nine Mile Point: 1.4×10^6 Gy (1.4×10^8 rads). It was not clear why the ABWR integrated gamma accident dose is lower than the corresponding doses quoted for several operating reactors. As a response to the above, GE's position, which was provided in Section 5.3.2.1.5 of SSAR Amendment 15, did not adequately address this issue. In the DFSER, the staff noted that, to resolve this issue, GE must fully explain why the ABWR integrated gamma

accident dose is lower than the corresponding doses quoted for several operating reactors. This was DFSER Open Item 3.11.3-3. Subsequently, GE determined that the integrated gamma dose for an accident inside primary containment for ABWR is 2×10^6 Gy (2×10^8 rads). Table 3I.16 of Appendix 3I of the SSAR provides this information. The staff concludes in its engineering judgement that this is a reasonable value and is consistent with values determined at other BWRs. This is acceptable and resolved DFSER Open Item 3.11.3-3.

On the basis of its review of the tables in SSAR Appendix 3I the staff concludes that the tables in Appendix 3I are acceptable.

3.11.4 Adequacy of Interface Requirements

As a result of staff review of interface requirements, these requirements have been reclassified as ITAAC items that will be reviewed as part of the ITAAC program. In these instances COL applicants must (1) present a summary of environmental conditions and qualified conditions for each applicable item of equipment located in a harsh-environment zone in the system component evaluation work sheets as described in Table I-1 of GE Topical Report NEDE-24326-1-P and compile these sheets in their plant-specific environmental qualification documents and (2) record and maintain in an auditable file the results of all qualification tests for applicable equipment.

Additionally, although not identified as an interface requirement, the DFSER states that COL applicants should develop a surveillance and maintenance program for each applicable equipment item located in a mild-environment zone to ensure its operability during its design life. The vendors of equipment located in a mild environment are required to submit a certificate of compliance certifying that the subject equipment was qualified according to the requirements identified to ensure its capability to perform its safety-related function in its applicable environment. This was DFSER COL Action Item 3.11.3-3. SSAR Sections 3.11.2 and 3.11.6.3 include this action. This is acceptable.

3.11.5 Conclusions

On the basis of the evaluation discussed, the staff finds that the program for environmental qualification of electrical equipment for the ABWR standard design is in compliance with all the requirements of 10 CFR 50.49 and is, therefore, acceptable.

3.12 Piping Design

The DFSER "Piping Design" section, which was under Section 14.1.3.3, is revised and incorporated into a new section, Section 3.12, of this report. In the following, except for identification numbers for the DFSER open, confirmatory, COL action, and TS items, all other designations for subsections, paragraphs, and the like are reassigned and keyed to Section 3.12 of this report, as appropriate.

3.12.1 Introduction

This section provides the staff's safety evaluation of GE's design acceptance criteria (DAC) approach for the ABWR piping design. The staff used the SRP guidelines to evaluate the piping design information in the ABWR SSAR and performed a detailed audit of the piping design criteria, including sample calculations. The staff evaluated the adequacy of the structural integrity and functional capability of safety-related piping systems. The review was not limited only to the ASME Boiler and Pressure Vessel Code Class 1, 2, and 3 piping and supports, but also included buried piping, instrumentation lines, the interaction of non-seismic Category I piping with seismic Category I piping, and any safety-related piping designed to industry standards other than the ASME Code. The staff's evaluation of the adequacy of the ABWR piping design analysis methods, design procedures, acceptance criteria, and related ITAAC that are to be used for the completion and verification of the ABWR piping design is provided in the following sections of this report. The staff's evaluation includes

- applicable codes and standards
- analysis methods to be used for completing the piping design
- modeling techniques
- pipe stress analyses criteria
- pipe support design criteria
- high-energy line break criteria
- LBB approach applicable to the ABWR
- generic piping design ITAAC

The staff must arrive at a final safety determination that, if the COL applicant successfully completes the piping design and analyses and the ITAAC as required by 10 CFR Part 52, using the design methods and acceptance criteria discussed herein, there will be adequate assurance that the piping systems will perform their safety-related functions under all postulated combinations of normal operating conditions, system operating transients, postulated pipe breaks, and seismic events.

Design of Structures, Components, Equipment, and Systems

3.12.2 Codes and Standards

GDC 1 requires that structures, systems, and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function. 10 CFR 50.55a requires that systems and components of boiling and pressurized water-cooled nuclear power reactors must meet the requirements of the ASME Code. It specifies the latest edition and addenda endorsed by the NRC and any limitations. RGs 1.84 and 1.85 list ASME Code Cases that the NRC staff finds acceptable.

In SSAR Tables 1.8-21 and 3.2-3, GE identified the ASME Code, Section III, and the specific edition and addenda that will be used for the design of ASME Code, Class 1, 2, and 3 pressure retaining components. In SSAR Table 5.2-1, the Code Cases that may be used are also identified.

3.12.2.1 ASME Boiler and Pressure Vessel Code

For the ABWR design certification, GE has established that the ASME Boiler and Pressure Vessel Code, Section III, will be used for the design of ASME Code Class 1, 2, and 3 pressure retaining components and their supports. The specific edition and addenda are provided in SSAR Tables 1.8-21 and 3.2-3. The ASME Code is considered Tier 1 information; however, the specific edition and addenda are considered Tier 2 information. The specific edition and addenda are considered Tier 2 information because of the continually evolving technical nature associated with the design and construction practices (including inspection and examination techniques) of the Code. Fixing a specific edition and addenda during the design certification stage may result in inconsistencies between design and construction practices during the detailed design and construction stages. The ASME Code involves a consensus process to reflect the evolving design and construction practices of the industry. Although the reference to a specific edition of the Code for the design of ASME Code class components and their supports is suitable to reach a safety finding during the design certification stage, the construction practices and examination methods of an updated Code that would be effective at the COL application stage must be consistent with the design practices established at the design certification stage.

The staff finds that the specification of the ASME Code as Tier 1 information and the specific edition and addenda as Tier 2 information is appropriate because it would provide the means for the COL applicant to revise or supplement the referenced Code edition with portions of the later Code editions and addenda needed to ensure consistency between the design for the ABWR pressure retaining components and their supports and construction practices. In this manner, the updated reference Code to be used at the time of the COL application is ensured to be consistent with the latest design, construction, and examination practices at that time. However, where the staff finds that there may be a need to specify certain design parameters from a specific Code edition or addenda during its design certification review, particularly when that information is of importance to establish a significant aspect of the design or is used by the staff to reach its final safety determination, such considerations, if necessary, are reflected in the various sections of this safety evaluation.

Therefore, all ASME Code Class 1, 2, and 3 pressure retaining components and their supports must be designed in accordance with the requirements of ASME Code, Section III, using the specific edition and addenda provided in the ABWR SSAR. However, the COL applicant should also ensure that the design is consistent with the construction practices (including inspection and examination methods) of the ASME Code edition and addenda as endorsed in 10 CFR 50.55a in effect at the time of COL application. The portions of the later Code editions and addenda must be identified to the NRC staff for review and approval with the COL application. This was DFSEER COL Action Item 14.1.3.3.2.1-1. GE has included this action in SSAR Section 3.9.7.4, which is acceptable.

3.12.2.2 ASME Code Cases

The only acceptable ASME Code cases that may be used for the design of ASME Code Class 1, 2, and 3 piping systems in the ABWR standard plant are those either conditionally or unconditionally approved in RGs 1.84 and 1.85 in effect at the time of design certification as listed below. However, the COL applicant may submit with its COL application for staff review and approval future code cases that are endorsed in RGs 1.84 and 1.85 at the time of COL application provided they do not alter the staff's safety findings on the ABWR certified design.

- In RG 1.84, the staff has conditionally endorsed ASME Code Case N-411, "Alternative Damping Values for Response Spectra Analysis of Classes 1, 2, and 3 Piping, Section III, Division 1." This Code Case is acceptable for the ABWR. The acceptability of the Code Case and its application is further discussed in Section 3.12.5.4 of this report.

Design of Structures, Components, Equipment, and Systems

Other ASME Code Cases requested by GE that are applicable to the ABWR piping and support design are listed below.

- ASME Code Case N-71-15, "Additional Materials for Subsection NF, Classes 1, 2, 3, and MC Component Supports Fabricated by Welding, Section III, Division 1." This Code Case has been endorsed by the staff in RG 1.85.
- ASME Code Case N-122, "Stress Indices for Structure Attachments, Class 1, Section III, Division 1." This Code Case has been endorsed by the staff in RG 1.84.
- ASME Code Case N-247, "Certified Design Report Summary for Component Standard Supports, Section III, Division 1, Classes 1, 2, 3 and MC." This Code Case has been endorsed by the staff in RG 1.84.
- ASME Code Case N-249-9, "Additional Material for Subsection NF, Classes 1, 2, 3 and MC Component Supports Fabricated Without Welding, Section III, Division 1." This Code Case has been endorsed by the staff in RG 1.85.
- ASME Code Case N-309-1, "Identification of Materials for Component Supports, Section III, Division 1." This Code Case has been endorsed by the staff in RG 1.84.
- ASME Code Case N-313, "Alternate Rules for Half-Coupling Branch Connections, Section III, Division 1." This Code Case has been endorsed by the staff in RG 1.84.
- ASME Code Case N-316, "Alternate Rules for Fillet Weld Dimensions for Socket Welded Fittings, Section III, Division 1, Class 1, 2, 3." This Code Case has been endorsed by the staff in RG 1.84.
- ASME Code Case N-318-3, "Procedure for Evaluation of the Design of Rectangular Cross Section Attachments on Class 2 or 3 Piping, Section III, Division 1." This Code Case has been conditionally endorsed by the staff in RG 1.84 and is discussed further in Section 3.12.5.16 of this report.
- ASME Code Case N-319, "Alternate Procedure for Evaluation of Stress in Butt Weld Elbows in Class 1 Piping, Section III, Division 1." This Code Case has been endorsed by the staff in RG 1.84.
- ASME Code Case N-391, "Procedure for Evaluation of the Design of Hollow Circular Cross Section Welded Attachments on Class 1 Piping, Section III, Division 1." This Code Case has been endorsed by the staff in RG 1.84.
- ASME Code Case N-392, "Procedure for Evaluation of the Design of Hollow Circular Cross Section Welded Attachments on Classes 2 and 3 Piping, Section III, Division 1." This Code Case has been endorsed by the staff in RG 1.84.
- ASME Code Case N-393, "Repair Welding Structural Steel Rolled Shaped and Plates for Component Supports, Section III, Division 1." This Code Case has been endorsed by the staff in RG 1.84.
- ASME Code Case N-414, "Tack Welds for Class 1, 2, 3 and MC Components and Piping Supports." This Code Case has been endorsed by the staff in RG 1.84.
- ASME Code Case N-430, "Requirements for Welding Workmanship and Visual Acceptance Criteria for Class 1, 2, 3 and MC Linear-Type and Standard Supports." This Code Case has been endorsed by the staff in RG 1.84.

All of the above Code Cases are listed in Table 5.2-1 of the SSAR. In addition, in Sections 3.9.3.4 and 3.9.3.5 of the SSAR, Code Case N-476, "Class 1, 2, & 3, and MC Linear Component Supports - Design Criteria for Single Angle Members, Section III, Division 1, Subsection NF," is referenced as augmenting ASME Subsection NF rules for the design of component supports. As stated in Section 3.9.3.3 of this report, the staff finds this Code Case acceptable. Therefore, the staff concludes that, since all of these Code Cases either meet the guidelines of RGs 1.84 or 1.85, or have been reviewed and endorsed by the staff, they are acceptable for use on the ABWR design.

3.12.2.3 Design Specifications

ASME Code, Section III, requires that a design specification be prepared for Class 1, 2, and 3 components such as pumps, valves, and piping systems. The design specification is intended to become a principal document governing the design and construction of these components and should specify loading combinations, design data, and other design data inputs. The Code also requires a design report for ASME Code, Class 1, 2, and 3 piping and components. In the SSAR, GE committed to construct all safety-related components, such as vessels, pumps, valves and piping systems, to applicable requirements of the ASME Section III. During its review of the SSAR, the staff reviewed selected documents related to design specifications and design reports. Those documents were not specifically for the ABWR, but were provided by GE and reviewed by the staff as a demonstration of how design

Design of Structures, Components, Equipment, and Systems

specifications and design reports will be prepared for ABWR plants. The staff determined that the demonstration documents, with modifications, would meet Code requirements. However, because the documents were not specifically for the ABWR, they would have to be modified before the staff can conclude that the design specification and design report requirements in ASME Code, Section III, Subsection NCA, have been met. In order for the staff to reach this conclusion, the COL applicant should submit representative design documents (e.g., design specifications) for NRC staff review as part of the COL application (DFSER COL Action Items 3.9.3.1-2 and 14.1.3.3.2.3-1). SSAR Section 3.9.7 states that COL applicants referencing the ABWR design will make available to the staff design specifications and design reports required by the ASME Code for vessels, pumps, valves, and piping systems for the purpose of audit. This is acceptable.

3.12.2.4 Conclusions

On the basis of these discussions and the evaluation of SSAR Section 3.9.3.1 and Tables 1.8-21, 3.2-3, and 5.2-1, the staff concludes that the piping systems important to safety are designed to quality standards commensurate with their importance to safety. The staff's conclusion is based on the following:

- (a) GE satisfies the requirements of GDC 1 and 10 CFR 50.55a by specifying appropriate codes and standards for the design and construction of safety-related piping and pipe supports, and
- (b) GE identified ASME Code Cases that may be applied to ASME Code, Class 1, 2, and 3 piping and pipe supports, which are acceptable to the staff.

3.12.3 Analysis Methods

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

3.12.3.1 Experimental Stress Analysis

SSAR Section 3.9.1.3 identifies several components for which experimental stress analysis is performed in conjunction with analytical evaluation. Such components in the piping area include the piping seismic snubbers and pipe whip restraints. The staff's evaluation of the experimental stress analysis methods is discussed in Section 3.9.1 of this report. The staff's evaluation of the analysis methods used to qualify these components is discussed further in Sections 3.12.6 and 3.12.7 of this report.

3.12.3.2 Modal Response Spectrum Method

GE performed system and subsystem analyses on an elastic basis. Modal response spectrum and time history methods form the basis for the analyses of all major seismic Category I piping systems and components. SSAR Section 3.7.3.8 describes the piping dynamic analysis procedure using the modal response spectrum method. First a mathematical model is constructed to reflect the dynamic characteristics of the piping system. The mode shapes and natural frequencies of the piping model are computed. Using a given direction of earthquake motion, the modal participation factors for each mode are calculated. Using the appropriate response spectrum curve, the spectral accelerations for each mode are determined. For a piping system supported at points with different dynamic excitations, an enveloped response spectrum of all attachment points is used. From the mode shapes, participation factors and spectral accelerations of each mode, the modal responses are calculated. They include the modal forces, shears, moments, stresses and deflections. For a given direction, the modal responses are combined in accordance with the methods described in SSAR Section 3.7.3.7.

The modal response calculations are performed for each of the three earthquake directions (two horizontal and the vertical). The total seismic response from the simultaneous application of the three-directional components of earthquake loading are obtained by combining the maximum codirectional responses of each of the three components by the square-root-of-the-sum-of-squares (SRSS) method as described in SSAR Section 3.7.3.6.

For piping systems that are anchored and restrained to floors and walls of buildings that have differential movements during a seismic event, additional forces and moments are induced in the system. Additional static analyses are performed to determine these loads as described in SSAR Section 3.7.3.8.1.8. The maximum differential displacements are applied to the piping anchors and restraints. Three analyses are performed: two in the horizontal directions and one in the vertical direction. The resulting stresses are placed in the secondary stress category because they are displacement induced and self-limiting. These secondary loads are combined with the primary (inertia) loads by the SRSS method.

The staff reviewed the SSAR description of the modal response spectrum method and found that it is consistent with the applicable guidelines in SRP Section 3.9.2 and is acceptable.

3.12.3.3 Independent Support Motion Method

As an alternative to the enveloped response spectrum method, the independent support motion (ISM) analysis method may be used. The theory and development of the governing equations of motion for this method are presented in SSAR Subsection 3.7.2.1.4. Additional requirements associated with the application of this method are described in the SSAR Subsection 3.7.3.8.1.10, "Multiply Supported Equipment and Components with Distinct Inputs." This section states that when this method of analysis is used, the following conditions must be met: (1) ASME Code Case N-411-1 damping is not used; (2) a support group is defined by supports which have the same time history input. This usually means all supports located on the same floor, or portions of a floor, of a structure; and (3) The responses from motions of supports in two or more different groups are combined by the SRSS procedure.

The staff finds that contingent upon these conditions, this alternative to the enveloped response spectrum method is acceptable. More details of the evaluation are discussed in Section 3.9.2.2. of this report.

3.12.3.4 Time-History Method

A time history analysis may be performed using either the modal superposition method or the direct integration method. The modal superposition method is described in SSAR Subsection 3.7.2.1.2. This approach involves the calculation and utilization of the natural frequencies, mode shapes, and appropriate damping factors of the particular system toward the solution of the equations of dynamic equilibrium. The orthogonality of the mode shapes is used to effect a coordinate transformation of the displacements,

velocities, and accelerations such that the response in each mode is independent of the response of the system in any other mode. Through this transformation, the problem becomes one of solving n independent differential equations rather than simultaneous differential equations. As long as the system is linear, the principle of superposition holds and the total response of the system oscillating simultaneously in n modes may be determined by direct addition of the responses of the individual modes.

The direct integration method is described in SSAR Section 3.7.3.1. This method involves the direct step-by-step numerical integration of the equations of motion and does not require the solution of an eigenvalue problem. The response in all modes is calculated simultaneously. The numerical integration time step, Δt , must be sufficiently small to accurately define the dynamic excitation and to render stability and convergence of the solution up to the highest frequency of significance. The integration time step is considered acceptable when smaller time steps introduce no more than a 10-percent error in the total dynamic response. For most of the commonly used integration methods, the maximum time step is limited to one-tenth of the smallest period of interest, which is generally the reciprocal of the cutoff frequency. In direct integration analysis, the damping is input in the form of α and β damping constants, which give the percentage of critical damping, λ as a function of the circular frequency, ω .

The total seismic response is predicted by combining the responses from the three orthogonal components (two horizontal and one vertical) of the earthquake. When separate time-history analyses are performed for each directional component, the combined response may be obtained by taking the SRSS of the maximum codirectional responses caused by each component. As an alternative, the combined response may be obtained by algebraically adding the codirectional responses from each analysis at each time step or the total response may be obtained directly by applying the three component motions simultaneously in one analysis. When either alternative method is used, the three component motions must be mutually statistically independent.

When the time-history method of analysis is used, the time-history data is broadened plus and minus 15 percent of Δt in order to account for modeling uncertainties. For loads such as safety-relief valve blowdown, tests have been performed that confirm the conservatism of the analytical results. Therefore, for these loads, the calculated force time histories are not broadened.

The staff reviewed the SSAR descriptions of the modal superposition and the direct integration time-history

Design of Structures, Components, Equipment, and Systems

analysis methods and found them to be in compliance with the applicable guidelines of SRP Section 3.9.2 and acceptable.

3.12.3.5 Inelastic Analysis Method

GE has not provided any information on the use of inelastic analysis methods for the ABWR piping analyses. If inelastic methods are to be used in any ABWR piping analyses, then the staff requires that the details of the inelastic method and its acceptance criteria, as well as the scope and extent of its application, be submitted to the staff for approval prior to its use by a COL applicant.

3.12.3.6 Small-Bore Piping Method

In the DFSER, the staff noted that GE had not provided any specific information about the method to be used for the structural design of small-bore piping systems and instrumentation lines in the ABWR plant. The staff requested that this information be included in the SSAR. This was DFSER Open Items 3.9.2.2-5 and 14.1.3.3.3.6-1. SSAR Section 3.7.3.8.1.9, "Design of Small Branch and Small Bore Piping," discusses the use of small-bore piping handbooks in lieu of performing piping analysis for piping 50.8 mm (2 in.) and less nominal pipe size, and small branch lines 50.8 mm (2 in.) and less nominal pipe size. It states that (a) the small-bore piping handbook must be currently accepted by the regulatory agency for use on equivalent piping at other nuclear power plants at the time of application; (b) when the handbook meets the purpose of the Design Report, it must meet all of the ASME requirements for a piping design report for piping and its supports; and (c) formal documentation exists showing that piping designed and installed in accordance with the handbook is conservative in comparison to results from a detailed stress analysis for all loads and load combinations, that it is not less reliable owing to loss of flexibility or excessive supports, and that it satisfies required clearances around sensitive components. The piping handbook methodology will not be applied when specific information is needed on pipe stresses, cumulative usage factors, accelerations, or break locations.

The staff reviewed the methodology described in SSAR Section 3.7.3.8.1.9, which states that the static and dynamic analysis methods defined in Section 3.7.3 of the SSAR will be used to provide the formal documentation showing that piping designed and installed to the small bore piping handbook is conservative in comparison to a detailed stress analysis. The staff finds this acceptable, and DFSER Open Items 3.9.2.2-5 and 14.1.3.3.3.6-1 are resolved.

3.12.3.7 Non-Seismic/Seismic Interaction (II/T)

All non-seismic Category I piping (or other systems and components) should be isolated from seismic Category I piping. This isolation may be achieved by designing a seismic constraint or barrier or by locating the two sufficiently apart to preclude any interaction. If it is impractical to isolate the seismic Category I piping system, the adjacent non-seismic Category I system should be evaluated to the same criteria as the seismic Category I system.

For non-seismic Category I piping systems attached to seismic Category I piping systems, the dynamic effects of the non-seismic Category I system should be considered in the analysis of the seismic Category I piping. In addition, the non-seismic Category I piping from the attachment point to the first anchor should be evaluated to ensure that, under all loading conditions, it will not cause a failure of the seismic Category I piping system. Section 3.7.3.13 in the SSAR contains criteria that are consistent with these staff positions and applicable portions of SRP 3.9.2 and RG 1.29, "Seismic Design Classification," Revision 3, and is therefore, acceptable.

3.12.3.8 Main Steamline and Bypass Line in the Turbine Building

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

For this reason, the staff's position is that the main steam piping beyond the second outermost isolation valve and up to the seismic interface restraint and the connecting branch lines up to the first normally closed valve should be classified as QG B (Safety Class 2) and seismic Category I. The MSL from the seismic interface restraint up to but not including the turbine stop valve (including branch lines to the first normally-closed valve) should be classified as QG B and inspected in accordance with the applicable portions of ASME Code, Section XI, but may be classified as non-seismic Category I if it has been analyzed, using a dynamic seismic analysis method to demonstrate its structural integrity under SSE loading conditions. However, all pertinent QA requirements of Appendix B to 10 CFR Part 50 are applicable to ensure that the quality of the piping material is commensurate with its importance to

Design of Structures, Components, Equipment, and Systems

safety during normal operational, transient, and accident conditions. To ensure integrity of the remainder of the alternative leakage path, the main steam bypass line, although it is not required to be classified as safety-related or as seismic Category I, the line from the first valve up to the condenser inlet, the main steam drain line from the first valve to the condenser, and the main steam piping between the turbine stop valve and the turbine inlet should be analyzed, using a dynamic seismic analysis, to demonstrate its structural integrity under SSE loading conditions. This was DSER Outstanding Issue 3 in SECY-91-153 and DFSEER Confirmatory Item 14.1.3.3.3.8-1.

SSAR Section 3.2.5.3, "Main Steamline Leakage Path," defines the piping and components that make up the main steam leakage path and provides their classifications and requirements for dynamic seismic analysis. It also states that a plant-specific walkdown will be conducted to confirm that the as-built main steam piping, bypass lines to the condenser, and the main condenser are not compromised by non-seismically designed systems, structures, and components. From its review, the staff review concludes that the SSAR adequately reflects these staff positions as described. On the basis of the evaluation as discussed and the evaluation reported in more detail in Section 3.2.1 of this report, DSER Outstanding Issue 3 and DFSEER Confirmatory Item 14.1.3.3.3.8-1 are resolved.

Lastly, the main steam piping, drain line, and bypass line in the turbine building should be protected from the collapse of any non-seismic Category I structure in the event of an SSE. As a confirmatory measure, a plant-specific walkdown should be performed before operation to assess the potential failures of non-seismically designed SSCs overhead, adjacent to, and attached to the alternative leakage path (i.e., the main steam piping, by-pass line, and the main condenser). In the DSER, the staff noted that this walkdown should be performed as a part of the ITAAC verification of non-seismic/seismic interaction. This was later identified in the DFSEER as Open Item 3.2.1-3 and Confirmatory Item 14.1.3.3.3.8-2. Upon further consideration, the staff found that the design detail and as-built and as-procured information for non-seismically designed SSCs in the turbine building as they affect the alternative leakage function of the main steam, bypass, and drain lines, and the main condenser are not required for design certification and the staff will review the spatial relationship between these SSCs and the main steam piping, bypass, and drain lines, and main condenser to ensure compliance with GDC 2. Subsequently, GE revised the SSAR and added SSAR Section 3.2.5.3 that contains a commitment to perform plant-specific walkdowns consistent with this staff position. This commitment to perform walkdowns is acceptable. Therefore, DFSEER Open Item 3.2.1-3 and DFSEER

Confirmatory Item 14.1.3.3.3.8-2 are withdrawn and resolved. The resolution of this issue is also discussed in Section 3.2.1 of this report.

3.12.3.9 Buried Piping

SSAR Section 3.7.3.12 originally outlined the criteria that will be used in the analysis of buried seismic Category I piping systems. These criteria conformed to the applicable guidelines in SRP Section 3.9.2. However, GE did not give any details on how the criteria are to be applied in the design of buried piping. In the DFSEER, the staff requested that the SSAR be revised to address, as a minimum, (1) the maximum bearing loads, (2) the categorization of seismic stresses in the Code evaluation, and (3) the allowable stress limits for the piping. This was DFSEER Open Items 3.9.2.2-7 and 14.1.3.3.3.9-1.

SSAR Section 3.7.3.12 states that all underground seismic Category I piping systems are installed in tunnels. The tunnels are analyzed as buried structures. The piping analysis is performed using one of the methods described in Section 3.7.3 of the SSAR. Because the SSAR states that the buried piping systems will not be in direct contact with the soil, the staff concludes that the information requested in the DFSEER is no longer applicable. Therefore, DFSEER Open Items 3.9.2.2-7 and 14.1.3.3.3.9-1 are withdrawn and closed.

3.12.3.10 ASME Code, Section III, Appendix N

The staff has not endorsed the use of ASME Code, Section III, Appendix N, which is a non-mandatory appendix that is still evolving and does not currently agree with some regulatory positions. Therefore, for the ABWR piping design, if the methodology in Appendix N is not consistent with regulatory positions discussed herein, the regulatory positions shall be used.

3.12.3.11 Conclusions

On the basis of the evaluations in Section 3.12.3, the staff concludes that the analysis methods to be used for all seismic Category I piping systems as well as non-seismic Category I piping systems that are important to safety are acceptable. The analysis methods utilize piping design practices that are commonly used in the industry and provide an adequate margin of safety to withstand the loadings as a result of normal operating, transient, and accident conditions.

3.12.4 Piping Modeling

GDC 2 requires that components important to safety should be designed to withstand effects of natural events including

Design of Structures, Components, Equipment, and Systems

earthquakes. 10 CFR Part 50, Appendix B requires that design quality should be controlled for ensuring structural and functional integrity of seismic Category I components. For determining design adequacy, each piping system is idealized as a mathematical model and dynamic analysis using computer programs is performed. Modeling techniques should be in conformance with generally recognized engineering practice and computer programs should be verified per one or more methods suggested in SRP Section 3.9.1. A piping benchmark program described in NUREG/CR-6049 is also provided by the NRC for aiding the verification process.

SSAR Sections 3.7.3.3 and 3.9.2.2 describe piping modeling techniques and SSAR Section 3.9.1.2 discusses quality control of computer programs and computer results.

3.12.4.1 Computer Codes

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

To review GE's computer verification program for ABWR piping modeling, the staff performed an independent confirmatory piping stress analysis of representative piping systems in the ABWR standard plant. The purpose of this analysis was to verify the adequacy of the computer program used by GE to generate the sample piping analyses that were audited by the staff on March 23 through 26, 1992, at GE's offices in San Jose, California. These were DFSER Open Items 3.9.1-2, 14.1.3.3.4.1-1 and 14.1.3.3.4.3-1. The results of the confirmatory analysis verify that this computer program is adequate with acceptable accuracy. The staff concludes that the computer program verification process for the ABWR is acceptable. Therefore, DFSER Open Items 3.9.1-2, 14.1.3.3.4.1-1, and 14.1.3.3.4.3-1 are resolved.

3.12.4.2 Dynamic Piping Model

For the dynamic analysis of seismic Category I piping, each system is idealized as a mathematical model consisting of lumped masses interconnected by elastic

members. The stiffness matrix for the piping system is determined, using the elastic properties of the pipe. This includes the effects of torsional, bending, shear, and axial deformations as well as change in stiffness as a result of curved members.

The staff reviewed the method for selecting the number of masses or degrees of freedom in the piping mathematical model to determine its dynamic response. GE's internal documents that were audited by the staff on March 23 through 26, 1992, showed pipe and fluid masses are lumped at nodes that are selected to coincide with the locations of large masses (e.g., valves, pumps, and tanks) and with locations of significant geometric changes (e.g., pipe elbows, reducers, and tees). Additional mass points are selected to ensure that the spacing between any two adjacent piping nodes and masses is no greater than an idealized value. This value corresponds to the length of a simply supported beam with a uniformly distributed mass whose undamped natural frequency is equal to the cutoff frequency. Since this approach, in effect, would capture all modes up to the cutoff frequency, the staff finds that the ABWR method for locating mass points is acceptable. In the DFSER, the staff requested that the SSAR be revised to reflect this approach as described (DFSER Confirmatory Items 3.9.2.2-2 and 14.1.3.3.4.2-1). SSAR Subsection 3.7.3.3.1.2, "Selection of Mass Points," provides more detailed mass point selection criteria for dynamic piping models. The staff reviewed these criteria and determined that the description in the SSAR reflects this approach as discussed and is, therefore, acceptable; therefore, DFSER Confirmatory Items 3.9.2.2-2 and 14.1.3.3.4.2-1 are resolved.

The effect of pipe support stiffness on the piping response must be considered in the analytical model. Supports must be modeled in accordance with the SSAR. If supports are not modeled as stated in the SSAR, justification will be provided to validate the stiffness values used in the piping model. The justification should include verification that the generic values are representative of the types of pipe supports used in the piping system. This alternative approach to use generic stiffness values and its bases should be submitted to the staff for review and approval before use. This was DFSER COL Action Item 14.1.3.3.4.2-1. SSAR Section 3.7.5.3, "Piping Analysis, Modeling of Supports," states that the COL applicant will provide the information requested in this action item. This is acceptable.

Additionally, because the amplified response spectra are generally specified at discrete building node points, any additional flexibility between these points and the pipe support (e.g., supplementary steel) also should be addressed. In the DFSER, the staff requested that the

SSAR be revised to incorporate this information. This was DFSE Open Item 3.9.2.2-2 and DFSE Confirmatory Item 14.1.3.3.4.2-2. SSAR Section 3.7.3.3.1.8, "Response Spectra Amplification at Support Attachment Points," states that the drywell equipment and pipe support structures (DEPSS) should meet the criteria given in SSAR Subsection 3.7.3.3.4. It further states that, if this criteria cannot be met, the COL applicant will generate the amplified response spectra at piping attachment points considering the DEPSS as part of the structure, using the dynamic analysis method described in SSAR Section 3.7.2, or will analyze the piping systems considering the DEPSS as part of the pipe support. The staff reviewed this clarification and determined that it is acceptable; therefore, DFSE Open Item 3.9.2.2-2 and DFSE Confirmatory Item 14.1.3.3.4.2-2 are resolved.

If piping terminates at non-rigid equipment (e.g., tanks, pumps, or heat exchangers), the analytical piping model should consider the flexibility and mass effects of this equipment. In the DFSE, the staff requested that the SSAR be revised to address how the flexibility and masses of equipment attached to the piping are to be modeled. This was DFSE Open Item 3.9.2.2-3 and DFSE Confirmatory Item 14.1.3.3.4.2-3. SSAR Subsection 3.7.3.3.1.6, "Modeling of Piping Supports," states that stiffnesses of supporting structures are included in the piping analysis model. Anchors at equipment such as tanks, pumps and heat exchangers are modeled with calculated stiffness properties. It also states that mass effects will be included for equipment that have a fundamental frequency of less than 60 Hz. A simplified model of the equipment will be included in the piping system model. The staff concludes that this issue is adequately addressed; therefore, DFSE Open Item 3.9.2.2-3 and DFSE Confirmatory Item 14.1.3.3.4.2-3 are resolved.

3.12.4.3 Piping Benchmark Program

To verify the adequacy of the computer program used by the COL applicant to complete the ABWR piping system design and analyses, the NRC staff established mathematical models of representative piping systems in the ABWR standardized plant that were used in a piping analysis benchmark program. The mathematical models are based on the dynamic piping model and on the piping stress analysis criteria described in Section 3.12.4.2 and Section 3.12.5, respectively, of this report. The benchmark program verifies the adequacy of linear-elastic, dynamic piping analysis methods using the enveloped response spectrum method, multiple response spectrum method, and time-history method of analyses.

The benchmark program essentially consists of constructing mathematical models of the ABWR feedwater piping system inside containment and an SRV discharge line inside the suppression pool wetwell area, using the COL applicant's computer program. The piping configuration for the piping models are described in NUREG/CR-6049, "Piping Benchmark Problems for the GE ABWR," and include piping dimensions, pipe sizes, materials, valve weights, support and anchor stiffnesses, and support locations. The piping input parameters for the benchmark analyses also are specified in the piping benchmark program and include damping values, loading definitions, and load combinations.

When the COL applicant's dynamic piping analyses are completed, the results of the analyses must be compared with the results of the benchmark problems provided in the piping benchmark program. The piping analysis results to be compared and evaluated include the system modal frequencies, the maximum pipe moments, the maximum support loads and equipment reactions, and the maximum pipe deflections. The acceptance criteria or range of acceptable values are specified in the piping benchmark program and must be satisfied. The COL applicant must document and submit any deviations from these values as well as the justification for such deviations to the NRC staff for review and approval before initiating final certified piping analyses. The piping benchmark program was DFSE COL Action Item 14.1.3.3.4.3-1. In SSAR Section 3.9.1.2 states that the COL applicant will provide this information as discussed. This is acceptable. The benchmark program provides assurance that the computer program used to complete the ABWR piping design and analyses produces results that are consistent with results considered acceptable to the NRC staff.

3.12.4.4 Decoupling Criteria

When analyzing piping systems, the size of the mathematical model might exceed the capacity of the computer program if large and small-bore piping are included. Thus, the small-bore branch lines are generally decoupled from the large-bore main piping. Originally, the SSAR did not provide any criteria for the decoupling of the piping systems in the analysis model. However, in a letter to the NRC dated February 24, 1992, GE provided the decoupling criteria in a GE document entitled, "ABWR SSAR Main Steam, Feedwater and SRVDL Piping Systems Design Criteria and Analysis Methods (draft)," Revision 0, dated February 1992. This document stated that if the ratio between pipe diameters of the branch line to main line is less than one-third, the branch line can be excluded from the piping model of the main line.

Design of Structures, Components, Equipment, and Systems

In the DFSER, the staff requested that GE provide the basis for this one-third ratio and revise the SSAR to include additional information on how mass and other effects are accounted for in the analysis (DFSER Open Items 3.9.2.2-4 and 14.1.3.3.4.4-1). SSAR Sections 3.7.3.3.1.3 and 3.7.3.8.1.9 provide more detailed decoupling criteria. The basic criterion was changed from a ratio of run to branch pipe diameter of 3 to 1, or more, to a moment of inertia ratio of 25 to 1, or more. In addition, small branch lines shall be designed with no concentrated masses, such as valves, in the first one-half span length from the main run pipe; and with sufficient flexibility to prevent restraint of movement in the main run pipe. Quantitative requirements for assessing the adequacy of the flexibility were provided. The staff reviewed the revised decoupling requirements and found them acceptable; therefore, DFSER Open Items 3.9.2.2-4 and 14.1.3.3.4.4-1 are resolved.

3.12.4.5 Conclusions

On the basis of these discussions and evaluation of SSAR Sections 3.7.3.3 and 3.9.2.2, the staff concludes that design control measures are acceptable to ensure quality of computer programs and design methods. The staff's conclusion is based on the following:

- (1) GE satisfies the requirements of GDC 2 by providing criteria for the seismic design and analysis of all seismic Category I piping and pipe supports using prescribed modeling techniques and design methods that are in conformance with generally recognized engineering practice.
- (2) GE meets Appendix B to 10 CFR Part 50 by demonstrating the applicability and validity of the computer programs for performing piping seismic analysis.
- (3) Computer programs to be used by the COL applicant to complete its analyses of the ABWR piping systems will be verified and validated using the NRC staff's piping benchmark program.

3.12.5 Pipe Stress Analysis Criteria

3.12.5.1 Seismic Input (Envelope vs. Site-Specific Spectra)

The ABWR standard plant is designed for an SSE ground motion defined by a RG 1.60 response spectrum anchored to a peak ground acceleration of 0.3g. Amplified building response spectra are generated for the ABWR standard plant to account for varying soil properties in the United States by enveloping 14 site conditions. GE has proposed

that the COL applicant use these enveloping amplified building response spectra provided in the SSAR to complete the design and analyses of the ABWR piping systems.

The staff recognizes that the enveloping amplified building response spectra for the ABWR plant contain conservatism that may be excessive for certain specific site conditions. If amplified building response spectra are generated using site-dependent properties, then the approach and method used must be submitted to the staff for review and approval as part of the COL application. The method used to generate the amplified building response spectra should be consistent with the method described in the SSAR Section 3.7.2 as approved by the staff. This was DFSER COL Action Item 14.1.3.3.5.1-1. Upon further consideration, the staff determined that a COL Action Item is not appropriate because this is only an option for the COL applicant. Therefore, DFSER COL Action Item 14.1.3.3.5.1-1 is deleted.

3.12.5.2 Design Transients

SSAR Table 3.9-1 lists the design transients for five plant operating conditions and the number of either plant operating events or cycles for each of the design transients that will be used in the design and fatigue analyses of the ASME Code Class 1 piping systems.

The operating conditions are

- ASME Service Level A - normal conditions
- ASME Service Level B - upset conditions — incidents of moderate frequency
- ASME Service Level C - emergency conditions — infrequent incidents
- ASME Service Level D - faulted conditions — low-probability postulated events
- testing conditions

The number of cycles in the original list of transients in SSAR Table 3.9-1 appeared to be based on a 40-year life. In the DFSER, the staff noted that for a design life of 60 years, the number of cycles for each transient should be increased by a factor of 1.5. GE was requested to revise the SSAR to reflect this factor. This was DFSER Open Items 3.9.1-1 and 14.1.3.3.5.2-1. The resolution of this item is discussed in Section 3.9.1 of this report.

The number of events or cycles resulting from each of the listed design transients that are applicable to other ASME

Code Class piping systems is to be documented by the COL applicant in its design specification and/or stress report for each component. This was DFSER COL Action Item 14.1.3.3.5.2-1. SSAR Section 3.9.7.2 states that a COL applicant will identify ASME Class 2 or 3 piping systems under severe cyclic loads and perform analysis to ensure integrity for 60 years. This is acceptable.

3.12.5.3 Loadings and Load Combinations

The staff reviewed the methodology used for load combinations and the selected values of allowable stress limits. GE provided the design criteria for all ASME Code, Class 1, 2, and 3 piping and piping supports, using the load combinations and stress limits given in SSAR Section 3.9.3.1. GE stated that the method used in the combination of dynamic responses of piping loadings shall be in accordance with NUREG-0484, Revision 1.

From its review, the staff concludes that appropriate combinations of normal, operating transients, and accident loadings are specified to provide a conservative design envelope for the design of piping systems. The load combinations are consistent with the guidelines provided in SRP Section 3.9.3 and are acceptable.

3.12.5.4 Damping Values

RG 1.61, "Damping Values for Seismic Design of Nuclear Power Plants," Revision 0, contains recommended values of damping to be used in the seismic analysis of SSCs. In addition, RG 1.84, Revision 25, conditionally endorses ASME Code Case N-411-1. The damping values used by GE are the same as those specified in either RG 1.61 or those specified in ASME Code Case N-411-1 as permitted by RG 1.84, which is acceptable.

The SSAR uses the damping values specified in ASME Code Case N-411 with the independent support motion (ISM) method of response spectrum analysis. The staff's position on the application of N-411 damping values to the ISM method of analysis is that it is acceptable when the ISM method is used in accordance with the information and recommendations in Sections 2.3 and 2.4 of NUREG-1061, Volume 4. In the DFSER, the staff requested GE to confirm that N-411 damping in an ISM analysis will be applied in accordance with the staff's position. This was DFSER Confirmatory Item 14.1.3.3.5.4-1. SSAR Subsection 3.7.3.8.1.10, "Multiple-Supported Equipment and Components with Distinct Inputs," states that when the ISM response spectrum method of analysis is used, ASME Code Case N-411-1 damping will not be used. On the basis of this new commitment, the staff concludes that GE will only use the recommended damping values given in RG 1.61 in an

ISM response spectrum analysis. This is acceptable. Therefore, DFSER Confirmatory Item 14.1.3.3.5.4-1 is resolved.

The staff's position on the use of N-411 damping values with ASME Code Case N-420 is that the two code cases may only be used in separate analyses as a further condition of RG 1.84, because the damping values established in Code Case N-411 might not be entirely appropriate for the damping characteristics of the linear energy absorbing supports. Therefore, the two Code Cases are not to be used in the same analysis. In the DFSER, GE was requested to confirm compliance with this staff position. This was DFSER Confirmatory Item 14.1.3.3.5.4-2. SSAR Subsection 3.7.3.8.1.7, "Damping Ratio," states that the ASME Code Case N-411-1 damping cannot be used for analyzing linear energy absorbing supports designed in accordance with ASME Code Case N-420. This is acceptable; therefore, DFSER Confirmatory Item 14.1.3.3.5.4-2 is resolved.

3.12.5.5 Combination of Modal Responses

For the response spectrum method of analysis, the modal responses are combined by the SRSS method. Closely spaced modes are combined using the criteria of RG 1.92, "Combining Modal Responses and Spatial Components in Seismic Response Analysis," Revision 1. The SSAR considers all modes with frequencies below 33 Hz in computing equipment and component response for seismic loadings, as stated in SSAR Sections 3.7.3.7.1 and 3.7.3.7.2. The staff concludes that this method is consistent with the applicable guidelines of SRP Section 3.9.2 and is acceptable.

3.12.5.6 High-Frequency Modes

For seismic analysis, consideration of high-frequency modes to preclude missing mass effects must be included. The staff's guidelines for this are provided in SRP Section 3.7.2, Appendix A.

For the analyses of vibratory loads (other than seismic) with significant high-frequency input (i.e., 33 to 100 Hz), the staff's positions are as follows:

- (1) GE should address the methodology for the combination of high-frequency modal results. The high-frequency modes must be combined in accordance with the guidelines provided in RG 1.92. Use of other combination methods, such as the algebraic modal combination method for combining high-frequency modes, will require further justification and staff approval before use.

Design of Structures, Components, Equipment, and Systems

- (2) GE should address non-linear analyses used to account for gaps between the pipe and its supports when subjected to vibratory loads with significant high-frequency. The description of and justification for such analyses must be submitted to the staff for review and approval before use.

In the DFSER, the staff requested that the SSAR be revised to reflect these two staff positions by stating that, if an alternative method is used, the details of its basis must be submitted to the staff for review and approval before use. This was DFSER Confirmatory Item 14.1.3.3.5.6-1. SSAR Subsection 3.7.3.7.3, "Methodologies Used to Account for High-Frequency Modes," provides a detailed procedure for calculating the responses associated with high-frequency modes above the cutoff frequency and combining them with the low-frequency modal responses. The staff reviewed this procedure and found that it follows the guidelines provided in SRP Section 3.7.2, Appendix A, and is acceptable.

GE also revised the SSAR to address the two issues related to analyses of vibratory loads with significant high-frequency input. SSAR Subsection 3.7.3.7, "Combination of Modal Responses," addresses the first issue. It clarifies the position that modal responses for modes below the cutoff frequency will be combined in accordance with methods that follow the guidelines of RG 1.92. The combination methods are applicable for seismic loads with 33 Hz cutoff frequency as well as for loads with higher frequency input, such as suppression pool dynamic loads that may have cutoff frequencies as high as 100 Hz. The responses associated with modes above the cutoff frequency are calculated and combined in accordance with the guidelines of SRP Section 3.7.2, Appendix A, as previously discussed. The staff reviewed this information and found that this revision adequately addressed the first issue. The second issue is addressed in the SSAR Section 3.7.3.3.4, "Analysis of Frame Type Supports," which states that nonlinear analysis methods to account for gaps between pipe and supports subjected to high frequency vibration loads, such as suppression pool loads will not be used. The staff reviewed this information and found it acceptable. The SSAR does not provide an analytical method to account for non-linear effects of excessively large gaps. Should such large gaps exist, it would mean a change to the commitments involving piping analysis methodology discussed in the SSAR, and it would result in an unreviewed safety question. Therefore, details of non-linear analysis should be submitted to the staff for approval prior to its implementation. On the basis of this evaluation, DFSER Confirmatory Item 14.1.3.3.5.6-1 is resolved.

3.12.5.7 Fatigue Evaluation for ASME Code Class 1 Piping

ASME Code, Section III requires that the cumulative damage from fatigue be evaluated for all ASME Code Class 1 piping. The cumulative fatigue usage factor should take into consideration all cyclic effects caused by the plant operating transients for a 60-year design life. However, recent test data indicates that the effects of the reactor environment could reduce the fatigue resistance of certain materials. A comparison of the test data with the Code requirements indicates that the margins in the ASME Code fatigue design curves might be less than originally intended. The DFSER reported that the staff was developing an interim position, which would be available at a later date, to account for the environmental effects in the fatigue design of the affected materials. At this time, the staff is assessing the potential generic implication of this issue on all operating plants. Depending on the severity of the issue, certain actions might be required to generically address this concern. This was inappropriately identified as Open Items 3.9.3.1-1 and 14.1.3.3.5.7-1 in the DFSER. DFSER Open Items 3.9.3.1-1 and 14.1.3.3.5.7-1 are subsumed by DFSER Open Item 14.1.3.3.5.7-2, which is discussed below.

Originally, for the ABWR, GE discussed with the staff its tentative procedure used for a foreign BWR plant design in progress. The information was provided to the staff during an audit held at the GE offices in San Jose, California, on March 23 through 26, 1992. The specified material for the ASME Code Class 1 piping in the ABWR is carbon steel. Using the GE position, additional fatigue evaluations would not be required when certain conditions are met, such as when the fluid temperature is below 245 °C (473 °F), the oxygen content is below 0.3 ppm, or the tensile stress hold time does not exceed 10 seconds. The exemption rules also extend to piping elbows and tees and valve bodies when these components are conservatively designed and analyzed, using the stress index method. Thus, only the circumferential girth butt welds in piping are considered to be critical by GE and are evaluated for environmental effects. The approach used by GE to account for the environmental effects on the girth butt welds is to modify the local peak stress through (1) the notch factor, (2) the mean stress factor, (3) the environmental correction factor, and (4) the butt-weld strength reduction factor. In the DFSER, the staff requested GE to include in its SSAR the proposed approach for accounting for the environmental effects in its fatigue analyses. This was DFSER Open Items 3.9.3.1-1 and 14.1.3.3.5.7-2.

Design of Structures, Components, Equipment, and Systems

SSAR Subsection 3.9.3.1.1.7, "Environmental Effects on Fatigue Evaluation of Carbon Steel Piping," commits to perform additional evaluations for environmental effects on the fatigue design of ASME Code, Section III, Class 1 carbon steel piping in accordance with GE document 408HA414. The SSAR describes the conditions for which these evaluations would be performed and the methodology for performing them. The staff found the approach consistent with the approach presented by GE at the March 1992 audit previously summarized. GE's procedure provided supplemental guidelines that enhance the design margin beyond the requirements of the ASME Code, Section III for fatigue evaluation. This is acceptable; and DFSER Open Items 3.9.3.1-1 and 14.1.3.3.5.7-2 are resolved.

3.12.5.8 Fatigue Evaluation of ASME Code Class 2 and 3 Piping

SSAR Section 3.9.3.1 states that the design life for the ABWR is 60 years. In response to an earlier staff request, SSAR Sections 3.9.3.1 and 3.9.7.2 stated that COL applicants will identify all ASME Code Class 2, 3, and QG D components that will be subjected to loadings that could result in thermal or dynamic fatigue and provide the analyses that are similar to those required by ASME Code, Section III, Subsection NB (ASME Class 1). This was DFSER COL Action Items 3.9.3.1-1 and 14.1.3.3.5.8-1. In the SSAR, GE modified these COL actions by further revising Sections 3.9.7.2 and 3.9.3.1 to state that COL applicants will identify ASME Code Class 2 or 3 or QG D components that are subjected to cyclic loadings, including operating vibration loads and thermal transient effects, of a magnitude and/or duration so severe that the 60-year design life cannot be assured by required Code calculations and, if similar designs have not already been evaluated, either provide an appropriate analysis to demonstrate the required design life or provide designs to mitigate the magnitude or duration of the cyclic loads. SSAR Section 3.9.3.1 provides criteria on the magnitude of severe thermal transients that should be evaluated for possible effect on plant life. These transients are temperature rate changes faster than 830 °C/hour (1494 °F/hour), when the total fluid temperature change is greater than 55 °C (100 °F). It also states specifically that COL applicants will perform ASME Code Class 1 fatigue analysis of the SRV discharge piping in the wetwell and the SRV quenchers. This is acceptable.

On the basis of current data, the staff believes that the margins built into the ASME fatigue design curves may not be sufficient to account for variations in the original fatigue test data because of various environmental effects. Therefore, consistent with the staff position discussed in Section 3.12.5.7 of this report, the staff's position for

ASME Code Class 2 and 3 piping for which a fatigue analysis is performed is that the environmental effects should be considered in the fatigue analysis. This was DFSER Open Item 14.1.3.3.5.8-1. SSAR Section 3.9.3.1 states that environmental effects will be considered in the fatigue analyses of the SRV discharge piping and SRV quenchers in accordance with the requirements for ASME Code, Section III, Class 1, carbon steel piping discussed in Section 3.9.3.1.1.7 of this report. This is acceptable, and DFSER Open Item 14.1.3.3.5.8-1 is resolved.

3.12.5.9 Thermal Oscillations in Piping Connected to the Reactor Coolant System

In accordance with NRC Bulletin 88-08, the staff requests that licensees and applicants review systems connected to the RCS to determine whether any sections of this piping that cannot be isolated can be subjected to temperature oscillations that could be induced by leaking valves.

In the design of the ABWR emergency core cooling systems, both the residual heat removal (RHR) system and high-pressure core flooders (HPCF) have piping that will be directly connected to the reactor pressure vessel (RPV). In the unisolable sections of RHR piping, leaking toward the RPV cannot occur because the pressure will always be higher on the reactor side during normal plant operation when the upstream pumps are not operating. In the HPCF system design, the only unisolable piping connected to the RPV will be the section of pipe between the reactor nozzle and the upstream isolation check valve. Cold water in this system will be upstream of the injection valve (gate valve) that is outside the primary containment. The region upstream of the injection valve will operate at a pressure lower than reactor pressure except when the HPCF safety function is required. Therefore, cold water will not flow to the unisolable pipe section and stratification will not be a problem in the HPCF system.

In the DFSER, the staff concluded that GE had adequately addressed the potential problems described in Bulletin 88-08. Subsequently, however, the staff noted that GE did not address the potential problem described in Supplement 3 to Bulletin 88-08, although Supplements 1 and 2 were addressed adequately. It involved the development of potential cyclic stratified flow and associated thermal striping that may occur because of possible leakage past the valve disk and out the valve stem packing gland. This flow stratification and striping may occur when the pressure on the upstream side of the valve is less than the RPV system pressure during normal operation. Supplement 3 reported an incident in which this phenomenon induced thermal fatigue, which led to a through-wall pipe crack in a foreign reactor. The staff reviewed the SSAR and requested that GE provide additional information to

Design of Structures, Components, Equipment, and Systems

address this concern. GE then identified the ABWR piping sections that could be susceptible to unacceptable thermal stresses, owing to this phenomenon. They include sections of the RHR system and the HPCF system. For the affected piping sections, GE will require that either (1) the gate valve in each of the unisolable piping sections be located at a distance equal or greater than 25 pipe diameters from the RPV nozzle or (2) stress analysis be performed to show that stresses and fatigue from potential stratification and thermal striping are acceptable per the ASME Code. The SSAR incorporated these new requirements by including an additional note on the P&ID's of the RHR system (SSAR Figure 5.4-10) and the HPCF system (SSAR Figure 6.3-7). This is acceptable; therefore, the staff concludes that GE has adequately addressed the potential problem described in Supplement 3 to Bulletin 88-08.

3.12.5.10 Thermal Stratification

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

During an audit conducted at GE's offices in San Jose, California, on March 23 through 26, 1992, the staff asked GE to explain and demonstrate how the thermal stratification phenomenon was considered in the ABWR piping design. In response, GE stated that thermal stratification will be considered as a normal design load in the ASME Code Class 1 stress and fatigue evaluation of the feedwater piping. The ABWR sample problem criteria document states that the feedwater line will be analyzed for two thermal stratification load cases: (1) thermal stratification in the piping at the RPV nozzle and (2) thermal stratification in the feedwater header piping. The loads will be included in the piping fatigue analysis and in the evaluations of the head fitting and RPV nozzles. The temperature differences and locations for the stratification loads were defined in the feedwater piping pressure/temperature cycle diagrams. This was DFSER COL Action Item 14.1.3.3.5.10-1. SSAR Section 3.9.3.1 states that the COL applicant shall design for thermal stratification in accordance with the requirements of that section. This is acceptable.

The staff reviewed and discussed the thermal stratification analysis methodology with the cognizant GE engineers. The staff found the analysis method acceptable with the exception of an apparent discrepancy in load application. GE defined the stratified temperature profile in the pipe cross section as a constant hot temperature in the top half and cold temperature in the bottom half, with a step change in temperature at the centerline. However, in the pipe stress analysis, a linear top-to-bottom temperature profile was applied. The linear temperature profile provides lower bending moments and stresses than the step change profile. GE was asked to justify (1) the adequacy of the piping analysis load input and (2) the omission of the high-cycle fatigue effects resulting from thermal striping and why it should not be considered in the analysis. The staff also asked GE to provide additional justification for their methodology, including test information to support their thermal stratification load definition. This was DFSER Open Items 3.9.3.1-2 and 14.1.3.3.5.10-1.

SSAR Section 3.9.3.1 states that thermal stratification is one of the specific operating conditions that is included in the loads and load combinations contained in the piping design specifications and design reports. Stratification can occur in the feedwater piping during plant startup and during hot standby conditions following scram. If evidence of stratification is detected in any other piping system during design or startup, it will be evaluated to determine whether it is significant in terms of stress and deflection. As a general guideline, if temperature differences between the top and bottom of the pipe are less than 27 °C (48.6 °F), it is assumed insignificant and not included in the design specification and design reports. The staff reviewed this information and found it acceptable.

In addition, SSAR Section 3.9.2.1.3, "Thermal Stratification in Feedwater Piping," describes a special test that will be performed as part of the startup program to monitor the conditions and effects of thermal stratification in sections of the feedwater piping where stratification is anticipated. The test program will measure temperatures around the pipe circumference, strains at points of high stress, and pipe displacements, which are due to bowing caused by stratification. GE committed to perform this test to address the staff concern over the adequacy of the linear temperature profile assumed in the thermal stratification analysis. From its review, the staff concludes that the test should provide appropriate data to evaluate the adequacy of the assumed temperature profile.

In a letter dated April 19, 1993, GE presented thermal striping evaluation for the feedwater header pipe. Calculations summarized in the letter report show that

Design of Structures, Components, Equipment, and Systems

stresses at the pipe wall caused by striping are well below the metal endurance limit and are, therefore, negligible. The staff concluded that GE had provided a technical approach to adequately address the issue of thermal stratification in the ABWR feedwater piping systems. GE has also included this information in the SSAR. On the basis of this evaluation, DFSER Open Items 3.9.3.1-2 and 14.1.3.3.5.10-1 are resolved.

3.12.5.11 Safety-Relief Valve Design, Installation, and Testing

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

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

In performing the hydraulic transient piping analyses associated with the SRV discharge, the SSAR assumes a minimum rise time of 20 msec. Rise times faster than this value could result in higher loads than analytically predicted. The assumed rise time is based on past SRV designs and existing test data. The COL applicant must retest the SRVs if it should purchase any SRV or install its SRV piping in a configuration that is not similar to those that have been tested. This was DFSER COL Action Items 3.9.3.2-1 and 14.1.3.3.5.11-1. In Amendment 33 to the SSAR, GE added Section 1A.3.7, "Testing of SRV and Discharge Piping," to state that the COL applicant will

confirm that any SRVs or discharge piping installed that is not similar to those that have been tested will be tested in accordance with SSAR Section 1A.2.9. The staff considers this acceptable. Therefore, COL Action Items 3.9.3.2-1 and 14.1.3.3.5.11-1 are resolved.

3.12.5.12 Functional Capability

Note 9 to the SSAR, Table 3.9-2, states that all ASME Code Class 1, 2, and 3 piping systems that are essential for safe shutdown under the postulated events listed in the table are designed to meet the recommendations in NUREG-1367, "Functional Capability of Piping Systems," dated November 1992. In no case shall the piping stress exceed the limits designated for Service Level D in the ASME Code, Section III. The Service Level D limits are $3.0 S_m$ (not to exceed $2.0 S_y$) for ASME Code Class 1 piping and $3.0 S_h$ (not to exceed $2.0 S_y$) for Class 2 and 3 piping. Dynamic testing conducted by EPRI, GE, and the NRC has established that these stress levels do not result in a loss of piping functional capability. Thus, the staff concludes the methodology and stress levels for ensuring the functional capability of piping systems acceptable.

3.12.5.13 Combination of Inertial and Seismic Motion Effects

Piping analyses must include the effects caused by the relative building movements at supports and anchors (seismic anchor motion) as well as the seismic inertial loads. This is necessary when piping is supported at multiple locations within a single structure or is attached to two separate structures.

The effects of relative displacements at support points must be evaluated by imposing the maximum support displacements in the most unfavorable combination. This can be performed, using a static analysis procedure. Relative displacements of equipment supports (e.g., pumps or tanks) must be included in the analysis along with the building support movements.

When required for certain evaluations, such as support design, the responses that are due to the inertia effect and relative displacement effect should be combined by the absolute sum method.

In lieu of this method, time histories of support excitations may be used, in which case both inertial and relative displacement effects are already included. Consideration of these effects and analyses was DFSER Open Item 14.1.3.3.5.13-1.

Section 3.7.3.8.1.8 of the SSAR describes the methodology for considering the effects caused by relative

Design of Structures, Components, Equipment, and Systems

building movements. The displacements that are obtained from the dynamic building analysis are applied to the piping anchors and restraints corresponding to the maximum differential displacements that could occur. Three analyses are performed: one for each of the two horizontal differential displacements and one for the vertical. The resulting stresses in the piping are treated as secondary stresses because they are self-limiting. The primary loads owing to inertia are combined with the secondary loads owing to the relative displacements by the SRSS method.

The SRSS combination of these responses deviates from the acceptance criteria of SRP Section 3.9.2, which specifies the more conservative absolute sum method. To justify this deviation, SSAR Section 3.7.3.8.1.8 states that the inertia and relative support displacement responses are dynamic in nature and their peak values are not expected to occur at the same time. Anchor movement effects are computed from static analyses in which the support movement effects are applied to produce the most conservative loads on the piping. In view of this, GE believes that an SRSS combination is appropriate. GE provided a description of the theoretical basis for this position based on an independent support-motion time-history analysis. The staff found the approach technically justifiable and consistent with the recommendations of NUREG-1061, Volume 4, for independent support motion analysis. Therefore, the staff concludes that the SRSS combination of responses owing to inertia and relative displacements is acceptable for the ABWR. On the basis of the above, DFSER Open Item 14.1.3.3.5.13-1 is resolved.

3.12.5.14 Cutoff Frequency for Hydrodynamic Loadings

SSAR Section 3.7 states that the cutoff frequency for dynamic analysis is 60 Hz for suppression pool hydrodynamic loads. For piping systems with a fundamental frequency greater than 20 Hz, the cutoff frequency is 100 Hz for suppression pool hydrodynamic loads. These cutoff frequencies were previously used in the hydrodynamic analyses for currently operating BWR plants. Because the hydrodynamic load methodology used for the ABWR is the same as that used for the operating BWR plants, the cutoff frequency is acceptable for the ABWR.

3.12.5.15 OBE as a Design Load

In SECY-93-087, the staff recommended eliminating the OBE from the design for advanced light-water reactors. The Commission approved the staff recommendations in its SRM dated July 21, 1993. For the ABWR, GE originally

proposed that the OBE be equal to one-third of the SSE. The DFSER noted that the staff was discussing details with GE for the necessary actions that will be required for eliminating the OBE from the design of SSCs in the ABWR and that the staff's evaluation of using a single-earthquake design based on only the SSE will be addressed in the FSER. This was DFSER Open Items 3.1-1 and 14.1.3.3.5.15-1.

In a letter to GE dated September 11, 1992, the staff transmitted a guidance document that identified the necessary changes to existing seismic design criteria that are acceptable for implementing the proposed rule change as it pertains to the design of safety-related SSCs in the GE ABWR. This document included specific supplemental criteria for fatigue, seismic anchor motion, and piping stress limits that should be applied when the OBE is eliminated. A detailed discussion of this document is in Section 3.1.1 of this report. For fatigue evaluation, two SSE events with 10 maximum stress cycles per event (or an equivalent number of fractional cycles) should be considered. The effects of SAM owing to the SSE should be considered in combination with the effects of other normal operational loadings that might occur concurrently. For Class 1 primary stress evaluation, seismic loads need not be evaluated for consideration of Level B Service Limits for Eq. (9). However, for satisfaction of primary plus secondary stress range limits in Eq. (10), the full SSE stress range or a reduced range corresponding to an equivalent number of fractional cycles must be included for Level B Service limits. These load sets should also be used for evaluating fatigue effects. In addition, the stress that is due to the larger of the full range of SSE anchor motion or the resultant range of thermal expansion plus half the SSE anchor motion range, must not exceed $6.0 S_m$. For Class 2 and 3 piping, seismic loads are not required for consideration of occasional loads in satisfying the Level B Service Limits for Eq. (9). Seismic anchor motion stresses are not required for consideration of secondary stresses in Eq. (10). However, stresses that are due to the combination of range of moments caused by thermal expansion and SSE anchor motions must not exceed $3.0 S_h$.

SSAR Section 3.9, defines a number of revised requirements associated with the elimination of the OBE. SSAR Section 3.7.3.2, "Determination of Number of Earthquake Cycles," states that the SSE is the only design earthquake considered for the ABWR. The fatigue evaluation of ASME components would take into consideration two SSE events with 10 peak stress cycles per event. Alternately, an equivalent number of fractional vibratory cycles may be used (but with an amplitude not less than one-third of the maximum SSE amplitude) when derived in accordance with Appendix D of IEEE Standard

Design of Structures, Components, Equipment, and Systems

344-1987. The staff found this commitment consistent with the NRC guidance document previously discussed above and the Commission-approved staff recommendations on the issue of OBE elimination as discussed in Section 3.1.1 of this report. This is acceptable.

SSAR Table 3.9-1, "Plant Events," was revised to replace the OBE cycles with 20 peak SSE cycles for evaluation of Service Level B limits. A footnote was added to require the effects of SAMs owing to SSE to be evaluated to ensure functionality during and following an SSE. The staff finds this commitment consistent with the NRC guidance document and the Commission-approved staff recommendations on the issue of OBE elimination as discussed in Section 3.1.1 of this report. This is acceptable.

SSAR Table 3.9-2, "Load Combinations and Acceptance Criteria," was revised to replace the normal operation and OBE load combination with a normal and SSE combination for fatigue evaluation of Class 1 components under Service Level B. Footnote 7 of this table defines the changes and additions to the stress limits of ASME Code, Section III, Subsections NB-3600 and NC/ND-3600. For Class 1 piping, the stress owing to the larger of the full range of SSE SAM moment or the resultant range of thermal plus one-half the SAM moment must not exceed $6.0 S_m$. SSE inertia and SAM loads must be included in NB-3600 Eqs. (10) and (11). For Class 2 and 3 piping, the stress that is due to the larger of the full range of SSE SAM moment or the resultant range of thermal plus one-half the SAM moment must not exceed $3.0 S_p$. SSE inertia and SAM loads must not be included in NC/ND-3600 Eqs. (9), (10), and (11). The staff finds these revised stress limits consistent with the requirements of the NRC guidance document. On the basis of this information and the more detailed discussion of this issue in Section 3.1.1 of this report, DFSER Open Items 3.1-1 and 14.1.3.3.5.15-1 are resolved.

3.12.5.16 Welded Attachments

For the analysis of local stresses at welded attachments to piping (e.g., lugs, trunnions, or stanchions), the SSAR presents several ASME Code Cases. Code Case N-318-3 is acceptable to the staff and is endorsed in RG 1.84. The staff noted that this Code Case is conditionally approved in RG 1.84 on the basis that the applicant specifies (1) the method of lug attachment, (2) the piping system involved, and (3) the location in the system where the case is to be applied. The staff concludes, however, that for the ABWR design certification, these conditions in RG 1.84 are not needed to reach a safety conclusion and, therefore, are not required.

Code Cases N-391 and N-392 are endorsed by the staff in RG 1.84 and are acceptable.

3.12.5.17 Modal Damping for Composite Structures

The staff reviewed the issue of modal damping for composite structures during its audit on March 23 through 26, 1992, at GE's offices in San Jose, California. At that time, the SSAR did not describe the application of modal damping for composite structures in the analysis of piping systems. However, a review of a GE internal document entitled, "Piping Systems Design Criteria and Analysis Methods," contained a table of damping values for various types of piping supports. The damping values for the piping supports (e.g., snubbers and struts) were higher than the damping values tabulated for the piping.

GE indicated that these values were presented because modal damping for composite structures could be used in a response spectrum analysis as an option. In the DFSER, the staff reported that if GE plans to use the modal damping for composite structures as an option for piping analysis, then a description and justification of the approach must be provided in the SSAR. This was DFSER Open Items 3.9.2.2-6 and 14.1.3.3.5.17-1.

SSAR Section 3.7.3.8.1.7, "Damping Ratio," states that strain-energy-weighted modal damping can also be used in the dynamic analysis of piping systems. Strain-energy weighing is used to obtain the modal damping coefficient owing to the contributions of damping from different elements of the piping system. The element damping values are specified in SSAR Table 3.7-1. The procedure for calculating strain-energy-weighted modal damping is given in SSAR Section 3.7.2.15. The staff reviewed the information in the SSAR and found it acceptable; therefore, DFSER Open Items 3.9.2.2-6 and 14.1.3.3.5.17-1 are resolved.

3.12.5.18 Minimum Temperature for Thermal Analyses

In the DFSER, the staff noted that GE had not provided any information that would establish a minimum temperature at which an explicit piping thermal expansion analysis would be required. Unless GE had provided this information in the SSAR, the staff would have required that thermal analyses be performed for all temperature conditions above ambient. These were DFSER Open Items 3.9.3.1-3 and 14.1.3.3.5.18-1.

SSAR Section 3.9.3.1, "Loading Combinations, Design Transients and Stress Limits," states that piping loads due to the thermal expansion of the piping and thermal anchor movements at supports are included in the piping load

Design of Structures, Components, Equipment, and Systems

combinations. All operating modes are evaluated, and the maximum moment ranges are included in the fatigue evaluation. Piping systems with maximum operating temperatures of less than or equal to 65.6 °C (150 °F) are not required to be analyzed for thermal expansion loading because, when below this temperature, thermal-induced stresses will be low and inconsequential to piping designs. The staff reviewed this information and concluded that GE had defined a reasonable and acceptable minimum temperature at which an explicit thermal analysis would be performed. On the basis of this evaluation, DFSE Open Items 3.9.3.1-3 and 14.1.3.3.5.18-1 are resolved.

3.12.5.19 Intersystem LOCA

In SECY-90-016, dated January 12, 1990, the NRC staff discussed the resolution of the intersystem LOCA issue for advanced light-water reactor plants by requiring that low-pressure piping systems that interface with the RCPB be designed to withstand full RCS pressure to the extent practicable. In its June 26, 1990, SRM, the Commission approved these staff recommendations provided that all elements of the low-pressure systems are considered. In the DFSE, the staff noted that GE had not yet submitted the details of the piping design for the full RCS pressure. This was DFSE Open Item 14.1.3.3.5.19-1.

SSAR Subsection 3.9.3.1, "Loading Combinations, Design Transients, and Stress Limits," requires that low-pressure piping systems that interface with the RC boundary be designed with a pipe wall thickness calculated for a pressure equal to 0.4 times the RCS pressure but not less than that of a schedule 40 pipe. On the basis of a staff review and discussions with GE and the more detailed evaluation in Section 3.9.3.1.1 of this report, the staff finds this requirement acceptable. Therefore, DFSE Open Item 14.1.3.3.5.19-1 is resolved.

3.12.5.20 Conclusions

On the basis of its review, the staff concludes that--

- (1) GE meets GDC 1 and 10 CFR Part 50, Appendix B with regard to piping systems being designed, fabricated, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed, and with appropriate quality control.
- (2) GE meets GDC 2 and 10 CFR Part 100, Appendix A with regard to design transients and resulting load combinations for piping and pipe supports to withstand the effects of earthquakes combined with the effects of normal or accident conditions.

- (3) GE meets GDC 4 with regard to piping systems important to safety being designed to accommodate the effects of and to be compatible with the environmental conditions of normal and accident conditions.
- (4) GE meets GDC 14 with regard to the reactor coolant pressure boundary of the primary piping systems being designed, fabricated, constructed, and tested to have an extremely low probability of abnormal leakage, of rapid propagating failure, and of gross rupture.
- (5) GE meets GDC 15 with regard to the reactor coolant piping systems being designed with specific design and service limits to assure sufficient margin that the design conditions are not exceeded.

3.12.6 Pipe Support Criteria

3.12.6.1 Applicable Codes

The staff reviewed the methodology used in the design of ASME Code Class 1, 2, and 3 component supports as described in SSAR Sections 3.9.3.4 and 3.9.3.5. The staff also assessed the design and structural integrity of three types of supports: plate and shell, linear, and component standard types. All ASME Code Class 1, 2, and 3 component supports for the ABWR standard plant will be constructed in accordance with ASME Code, Section III, Subsection NF. In addition, the SSAR states that the design is augmented by the application of Code Case N-476, Supplement 89.1, which governs the design of single-angle members. It further states that if eccentric loads or other torsional loads are not accommodated by designing the load to act through the shear center, analyses will be performed in accordance with such torsional analysis methods as "Torsional Analysis of Steel Members," AISC Publication T1142/83.

Although Code Case N-476 has not been endorsed by the staff in RG 1.84, the staff finds that it provides adequate design rules for the single-angle members. The staff finds that GE's proposed documents provide sufficient technical guidelines to perform a torsional analysis of steel members and are acceptable.

The staff finds Subsection NF acceptable for the design of piping supports. The staff has not endorsed the use of ANSI/AISC N-690, "Specification for the Design, Fabrication and Erection of Steel Safety-Related Structures for Nuclear Facilities," in lieu of Subsection NF (see Section 3.12.6.12 of this report for more information).

3.12.6.2 Jurisdictional Boundaries

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

3.12.6.3 Loads and Load Combinations

SSAR Section 3.9.3.4.1 states that the loading combinations for the design of piping supports correspond to those used for the design of the supported pipe. The staff's evaluation of the load combinations for the supported pipe is contained in Section 3.12.5.3 of this report. The stress limits for pipe supports are in accordance with the ASME Code, Section III, Subsection NF and Appendix F (1989 Edition). The supports are generally designed or qualified by the load rating method as described in NF-3260 or by the stress limits specified in NF-3231. This is acceptable.

3.12.6.4 Pipe Support Baseplate and Anchor Bolt Design

SSAR Section 3.9.3.4 of earlier amendments stated that the concrete anchor bolts, which would be used for pipe support base plates, will be designed to the applicable factors of safety defined in Office of Inspection and Enforcement (IE) Bulletin 79-02, Revision 1. Loading combinations for component supports are discussed in the previous section. In general, the factors of safety for anchor bolts are acceptable. However, in the DFSER, the staff noted that in SSAR Section 3.9.3.4, "Component Supports," GE had not discussed the use of specific types of anchor bolts to be used for pipe support base plates in the ABWR standard plant. For example, under-cut-type anchor bolts behave in a ductile manner, but the staff's position is that the safety factors in IE Bulletin 79-02, Revision 2, are still applicable unless justification for alternative safety factors is provided. Therefore, the COL applicant should justify the use of safety factors for anchor bolts other than those provided in IE Bulletin 79-02, Revision 2, and submit the justification to the staff for review and approval before their use. These were DFSER COL Action Items 3.9.3.3-1 and 14.1.3.3.6.4-1. In addition, the staff noted that irrespective of the type of concrete anchor bolt used for piping supports, the COL applicant should implement the action item in IE Bulletin 79-02 related to pipe support baseplate flexibility. This was DFSER COL Action Item 14.1.3.3.6.4-2.

SSAR Section 3.9.3.4 states that concrete anchor bolts (including under-cut-type anchor bolts) that are used for pipe support base plates will be designed to the applicable factors of safety that are defined in IE Bulletin 79-02, Revision 2. SSAR Section 3.9.3.4 further states that pipe support base plate flexibility will be accounted for in the calculation of concrete anchor bolt loads, in accordance with IE Bulletin 79-02, Revision 2. On the basis of these statements, the staff concludes that the ABWR pipe support baseplate and anchor bolt designs will be in accordance with applicable portions of IE Bulletin 79-02, Revision 2, and are acceptable. Loading combinations for component supports are discussed in the section above.

In addition, SSAR Section 3.9.3.4 states that the COL applicants shall provide justification for the use of safety factors for concrete anchor bolts other than those specified in IE Bulletin 79-02. This justification must be submitted to the NRC for review and approval prior to installing of the bolts. SSAR Section 3.9.3.4 also states that COL applicants shall account for pipe support base plate flexibility in accordance with IE Bulletin 79-02. This is acceptable.

3.12.6.5 Use of Energy Absorbers and Limit Stops

The DFSER noted that GE had not provided the specific analysis methods or procedures to be used for the ABWR pipe support design. The staff requested that GE address in the SSAR the use of seismic restraints other than snubbers and their modeling assumptions. This was part of DFSER Open Items 3.9.3.3-1 and 14.1.3.3.6-1.

SSAR Section 3.7.3.3.1.7, "Modeling of Special Engineered Pipe Supports," states that modifications to the normal linear-elastic piping analysis methodology used with conventional pipe supports are required to calculate the loads acting on the supports and on the piping components when special engineered supports are used. These supports include energy absorbers and limit stops, which are described in SSAR Section 3.9.3.4.1(G). The modifications are needed to account for greater damping of the energy absorbers and the non-linear behavior of the limit stops. If these special devices are used, the modeling and analytical methodology will be in accordance with methodology accepted by the regulatory agency at the time of design certification or at the time of COL application, per the discretion of the COL applicant. In addition, the information required by RG 1.84 related to the use of energy absorbers (ASME Code Case N-420) will be provided to the regulatory agency. This is acceptable. Although the staff has not yet endorsed the use of limit stops, the commitment in SSAR Section 3.7.3.3.1.7 to use design criteria in accordance with that accepted by the regulatory agency at the time of either design certification

Design of Structures, Components, Equipment, and Systems

or COL application, is also acceptable to the staff. On the basis of this evaluation, the applicable parts of DFSER Open Items 3.9.3.3-1 and 14.1.3.3.6-1 are resolved.

3.12.6.6 Use of Snubbers

The DFSER noted that GE had not provided the specific analysis methods or procedures to be used for the ABWR pipe support design. The staff requested that GE address in the SSAR the types of snubbers to be used in the ABWR standard plant and their characteristics. This was part of DFSER Open Items 3.9.3.3-1 and 14.1.3.3.6-1.

SSAR Section 3.9.3.4.1, "Piping," provides a description of the snubbers, design criteria and testing, installation, and examination requirements. It states that both mechanical and hydraulic type snubbers will be used. It describes how their operational and performance characteristics will be verified by test. From its review of the description, the staff concluded that the information provided is consistent with applicable portions of SRP Section 3.9.3 and is acceptable; therefore, the applicable parts of DFSER Open Items 3.9.3.3-1 and 14.1.3.3.6-1 are resolved.

3.12.6.7 Pipe Support Stiffnesses

The DFSER noted that GE had not provided the specific analysis methods or procedures to be used for the ABWR pipe support design. The staff requested that GE address in the SSAR the pipe support stiffness values and support deflection limits used in the piping analyses. This was part of DFSER Open Items 3.9.3.3-1 and 14.1.3.3.6-1.

SSAR Sections 3.7.3.3.1.6, "Modeling of Piping Supports," and 3.7.3.3.4, "Analysis of Frame Type Supports," describe the criteria for using equivalent stiffnesses for snubbers and struts, including their supporting structures, and for frame type supports. Methods for determining the stiffnesses of each support type were provided. Stiffnesses for frame type supports and for supporting structures must be included unless they can be shown to be rigid. GE provided a deflection limit criteria that must be met for supports that are modeled as rigid. The staff reviewed this information and found it acceptable. Therefore, DFSER Open Items 3.9.3.3-1 and 14.1.3.3.6-1 are resolved with regard to the issue of pipe support stiffness.

3.12.6.8 Seismic Self-Weight Excitation

The DFSER reported that GE had not provided the specific analysis methods or procedures to be used for the ABWR pipe support design. The staff requested that GE address in the SSAR the seismic excitation of the pipe supports

(especially large frame-type structures) in the design of the pipe support anchorage. This was part of DFSER Open Items 3.9.3.3-1 and 14.1.3.3.6-1.

SSAR Section 3.7.3.3.4 includes a description of the design loads to be considered in the analysis of frame-type supports. It explains that in addition to the loads transmitted from the piping to the support, the support internal loads caused by the weight, thermal, and inertia effects, which are due to the support structure itself, must be included in the support analysis. Therefore, the seismic self-weight excitation will be included, which is acceptable; therefore, DFSER Items 3.9.3.3-1 and 14.1.3.3.6-1 are resolved with regard to this issue.

3.12.6.9 Design of Supplementary Steel

SSAR Section 3.9.3.4 provides design criteria for the design of pipe supports, using supplementary steel. Supplementary steel for pipe supports are designed in accordance with ASME Code, Section III, Subsection NF. The use of Subsection NF is standard industry practice and has been proven to provide adequate design guidelines for the design of structural steel for use as pipe supports. This is acceptable.

3.12.6.10 Consideration of Friction Forces

The DFSER reported that GE had not provided the specific analysis methods or procedures to be used for the ABWR pipe support design. The staff requested that GE address in the SSAR the coefficient of friction to be used for considering friction forces between the pipe and the steel frames. This was part of DFSER Open Items 3.9.3.3-1 and 14.1.3.3.6-1.

SSAR Section 3.7.3.3.4, "Analysis of Frame Type Supports," includes a description of the design loads that must be considered in the analysis of frame-type supports. One of the design loads includes friction loads caused by a pipe sliding on the support. In calculating these loads, GE will use static coefficients of friction of 0.80 for steel on steel and 0.15 for lubricated plates. The staff finds this acceptable because the magnitude of these specified static coefficients is sufficiently conservative to define the friction force developed at different kinds of contact surfaces of a sliding pipe on its support; therefore, the applicable parts of DFSER Open Items 3.9.3.3-1 and 14.1.3.3.6-1 are resolved.

3.12.6.11 Pipe Support Gaps and Clearances

The DFSER noted that GE had not provided the specific analysis methods or procedures to be used for the ABWR pipe support design. The staff requested that GE address

the hot and cold gaps to be used between the pipe and the box-frame-type of support. This was part of DFSER Open Items 3.9.3.3-1 and 14.1.3.3.6-1.

SSAR Section 3.7.3.3.4 includes information on allowable gaps. It states that the total gap or diametral clearance between the pipe and frame support shall be between 1.6 mm (1/16 in.) and 4.76 mm (3/16 in.) when the pipe is in either the hot or cold condition. The staff finds this acceptable because a small gap will ensure validity of using the linear analysis methodologies presented in the SSAR for piping design; therefore, the applicable parts of DFSER Open Items 3.9.3.3-1 and 14.1.3.3.6-1 are resolved.

3.12.6.12 Instrumentation Line Support Criteria

In the DFSER the staff noted that GE had not provided any information on the design criteria for the structural design of instrumentation line supports. The staff requested that this information be included in the SSAR. This was DFSER Open Item 3.9.3.3-2 and part of DFSER Open Item 14.1.3.3.6-1.

SSAR Sections 3.7.3.8.1.9 and 3.9.3.4.1 state that supports for ASME Code, Section III instrumentation lines are analyzed in accordance with SSAR Subsection 3.7.3, "Seismic Subsystem Analysis," and designed in accordance with SSAR Section 3.9.3.4, "Component Supports." Thus, these instrumentation lines will be analyzed, using the same methodology that will be applied to small-bore piping, and will be designed in accordance with ASME Code, Section III, Subsection NF. The staff finds this acceptable because these criteria and this guidance are sufficient to ensure code compliance and seismic design adequacy of the instrumentation line supports; therefore, DFSER Open Item 3.9.3.3-2 and the applicable part of DFSER Open Item 14.1.3.3.6-1 are resolved.

The industry has taken the position that ANS/AISC N-690 is useful in the design of instrumentation sensing line supports and has recommended that the industry be allowed to use it. Its use would have the effect of reducing the QA recordkeeping requirements and Code stamping required by Section NF of ASME Code, Section III. The staff believes that ANS/AISC N-690 alone is not an acceptable standard for construction of ASME component supports. ASME Code, Section III, Subsection NF, should be used. However, the staff is currently participating in the ASME effort to incorporate N-690 into Subsection NF. Subsequent to a staff-endorsed version of NF incorporating N-690, Subsection NF will specify the rules acceptable to the staff for construction of ASME Code Class supports. In the DFSER, the staff stated that when this staff-approved version is available,

the COL applicant seeking its use may submit a request to the staff for approval on a plant-specific basis. This was DFSER COL Action Item 14.1.3.3.6.12-1. Upon further consideration, the staff determined that a COL action item would not be appropriate because, as discussed in Section 3.12.2.1 of this report, DFSER COL Action Item 14.1.3.3.2.1-1 provides a commitment for the COL applicant to ensure that the design will be consistent with ASME Code and addenda as endorsed in 10 CFR 50.55a in effect at the time of application. Therefore, DFSER COL Action Item 14.1.3.3.6.12-1 is deleted.

3.12.6.13 Pipe Deflection Limits

In the DFSER the staff noted that GE had not provided the specific analysis methods or procedures to be used for the ABWR pipe support design. The staff requested that GE include this information in the SSAR design criteria that will ensure that the maximum deflections of the piping at support locations for static and dynamic loadings are within an allowable limit to preclude failure of the pipe supports and hangers. This was part of DFSER Open Items 3.9.3.3-1 and 14.1.3.3.6-1.

SSAR Section 3.9.3.4.1, "Piping," includes design criteria to ensure that maximum deflections at pipe supports will be within allowable limits. The staff reviewed the deflection criteria and found them acceptable because they are compatible with design assumptions and standard engineering practice; therefore, the applicable portions of DFSER Open Items 3.9.3.3-1 and 14.1.3.3.6-1 are resolved.

3.12.6.14 Conclusions

On the basis of these discussions and the evaluation of SSAR Sections 3.9.3.3, 3.9.3.4, 3.9.3.5, 3.7.3.3, the staff concludes that supports of piping systems important to safety are designed to quality standards commensurate with their importance to safety. The staff's conclusion is based on the following:

- (1) GE satisfies the requirements of GDC 1 and 10 CFR 50.55a by specifying methods and procedures for the design and construction of safety-related pipe supports in conformance with general engineering practice, and
- (2) GE satisfies the requirements of GDC 2 and GDC 4 by designing and constructing safety-related pipe supports to withstand the effects of normal operation as well as postulated events such as LOCAs and dynamic effects resulting from the SSE.

Design of Structures, Components, Equipment, and Systems

3.12.7 High-Energy Line-Break Criteria

GDC 4 requires that SSCs important to safety be designed to be compatible with and to accommodate the effects of the environmental conditions resulting from normal operations, maintenance, testing, and postulated accidents, including LGCAs. It also requires that they be adequately protected against dynamic effects (including the effects of missiles, pipe whipping, and discharging fluids) that may result from equipment failures and from events and conditions outside the nuclear power plant.

In accordance with SRP Section 3.6.2, Revision 2, the staff reviewed GE's proposed criteria and methodology to postulate pipe breaks and leakage cracks and to analyze the effects of breaks in high-energy fluid systems on adjacent safety-related SSCs with regard to pipe whip and jet impingement loadings. In the DFSER, the staff noted that the COL applicant should use these criteria and methodology to postulate locations of pipe breaks and leakage cracks to ensure adequate protection against the dynamic effects of postulated ruptures of piping in the ABWR standard design. This was DFSER COL Action Item 14.1.3.3.7-1. SSAR Section 3.6.5.1, "Details of Pipe Break Analysis Results and Protection Methods," states that the COL applicant shall provide a summary of the dynamic analyses applicable to high-and moderate-energy piping systems in accordance with Section 3.6.2.5 of RG 1.70. This is acceptable.

In the DFSER, the staff also identified Open Items 3.6.2-2 and 14.1.3.3.7-1 regarding the edition of ANSI/ANS-58.2, "Design Basis for Protection of Light Water Nuclear Power Plants Against the Effects of Pipe Rupture," which is referenced in SSAR Section 3.6.2.2.1. Originally, there were inconsistencies between the criteria for evaluating the effects of fluid jets on essential SSCs specified in SSAR Section 3.6.2.3.1 and corresponding criteria specified in SRP 3.6.2 and ANSI/ANS-58.2, 1988 Edition. SSAR Table 1.8-21, "Industrial Codes and Standards Applicable to ABWR," specifies the current 1988 Edition of the ANSI/ANS-58.2 Standard. In addition, the criteria described in SSAR Section 3.6.2.3.1 are consistent with SRP Section 3.6.2 and ANSI/ANS-58.2, 1988 edition. This is acceptable; therefore, DFSER Open Items 3.6.2-2 and 14.1.3.3.7-1 are resolved.

By letter dated September 11, 1992, the staff provided GE with guidance for the use of a single-earthquake design for SSCs in the ABWR. This guidance proposes that the criteria for postulating pipe breaks in seismically designed, high-energy piping systems be based on normal and operational transients only, excluding earthquake loadings. SSAR Sections 3.6.1 and 3.6.2 and Tables 3.9.1 and 3.9.2 incorporate the staff guidance for the use of a single-

earthquake for the postulation of pipe break locations. On the basis of this incorporation and the staff's evaluation of this issue in Section 3.1.1.3 of this report, the staff concludes that the ABWR criteria are consistent with staff positions discussed in Section 3.1.1 of this report, which is acceptable.

The staff also told GE that relaxation of the current BTP MEB 3-1, Revision 2, June 1987, Section B.1.c.(1)(b) criteria to the previous BTP MEB 3-1, Revision 1, July 1981 criteria would be acceptable. These criteria are for the postulation of pipe breaks in ASME Code, Section III, Class 1 high-energy fluid-system piping in areas other than containment penetration. This staff position stated is based on the following rationale:

For ASME Code Class 1 piping, the NRC position for postulating pipe breaks is delineated in the Branch Technical Position MEB 3-1 of the SRP 3.6.2. Prior to issuance of Revision 2 of BTP MEB 3-1 in June 1987, breaks were postulated at intermediate locations between terminal ends of a pipe run if the maximum stress range as calculated by Code Eq. (10) $> 2.4 S_m$ and either by Eq. (12) or Eq. (13) $> 2.4 S_m$. These stated were implemented in many plants operating today.

In Revision 2 of BTP MEB 3-1, the same criteria are maintained for break exclusion in the containment penetration areas. However, for other areas, the criteria were revised requiring that breaks be postulated at any intermediate locations when only Eq. (10) exceeds $2.4 S_m$. The use of Eq. (12) and Eq. (13) was eliminated.

The staff reviewed the impact of BTP MEB 3-1, Revision 2, and found that the Revision 2 criteria are inconsistent in that they allow higher limits in the containment penetration areas than in other areas. It would appear that the break exclusion area should provide a margin greater than (or at least equal to) the margin for areas outside the break exclusion area. In addition, the Revision 2 criteria will result in a significant increase in the number of postulated pipe breaks, which may be counter-productive in terms of enhancing plant safety.

For these reasons, the staff concludes that the Revision 1 criteria related to allowing the use of Eq. (12) and Eq. (13) should be reinstated for the postulation of intermediate pipe breaks in ASME Code Class 1 piping systems. SSAR Section 3.6.2 includes the use of Revision 1 of BTP MEB 3-1 criteria in B.1.c.(1)(b). On the basis of this evaluation and a similar evaluation in Section 3.6.2 of this report, the staff concludes that these

criteria are an acceptable deviation of applicable criteria in SRP 3.6.2, Revision 2.

During the audit performed at GE in San Jose, California, on March 23 through 27, 1992, the staff discussed with GE the as-yet-developed procedures for the ABWR for postulated pipe break analyses. This was a part of DFSER Open Item 3.6.2-1. SSAR, Appendix 3L, "Evaluation of Postulated Ruptures in High Energy Pipes," defines a procedure for evaluating dynamic effects of fluid dynamic forces resulting from postulated ruptures in high-energy piping systems. The four major steps in the evaluations include: (1) the identification of rupture locations and rupture geometry, (2) the design and selection of pipe whip restraints, (3) the procedure for dynamic time-history analysis with simplified models, and (4) the procedure for dynamic time-history analysis, using detailed piping models. The staff's review concludes that the procedure in Appendix 3L is consistent with applicable guidelines in SRP Sections 3.6.2 and 3.9.2 and are acceptable; therefore, this part of DFSER Open Item 3.6.2-1 is resolved. The remainder of DFSER Open Item 3.6.2-1 is discussed in Section 3.6.2 of this report.

3.12.7.1 High-Energy Piping Systems

Pipe whip need only be considered for those high-energy piping systems having fluid reservoirs with sufficient capacity to develop a jet stream. The criteria for determining high- and moderate-energy lines in SRP Section 3.6.1, Branch Technical Position (BTP) ASB 3-1, are adequately defined in Section 3.6.2.1 of the SSAR. All high-energy systems are listed in SSAR Tables 3.6-3 and 3.6-4.

3.12.7.2 Pipe Break Criteria Within the Containment Penetration Areas

Breaks are not postulated in the ABWR design for those portions of high-energy piping between the isolation valves outside and inside the containment that are designed to meet ASME Code, Section III, Article NE-1120, and the additional design guidelines in SRP Section 3.6.2, including BTP MEB 3-1, Revision 2, June 1987. These guidelines recommend that an augmented inservice inspection program be implemented for those portions of piping within the break-exclusion region. In the DFSER, the staff noted that the COL applicant should perform a 100-percent volumetric examination of circumferential and longitudinal pipe welds in the break-exclusion region during each inspection interval as defined in Article IWA-2400, ASME Code, Section XI. This was DFSER COL Action Item 14.1.3.3.7.2-1. SSAR Section 3.6.5.3, "Inservice Inspection of Piping in Containment Penetration Areas," states that the COL

applicant shall perform an augmented inservice inspection program, as defined. This is acceptable.

3.12.7.3 Pipe Break Criteria Outside the Containment Penetration Areas

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

SRP Section 3.6.2 guidelines state that if a structure separates a high-energy line from an essential component, the separating structure should be designed to withstand the consequences of the pipe break in the high-energy line that produces the greatest effect at the structure, irrespective of the fact that the pipe break criteria in SRP Section 3.6.2 might not require such a break location to be postulated.

The ABWR structures are designed to withstand the dynamic effects of pipe breaks where the pipe rupture criteria require break locations to be postulated. In addition, for areas where physical separation of redundant trains is not practical, a high-energy line separation analysis (HELSEA) will be performed by the COL applicant to determine which high-energy lines meet the spatial separation requirements and which lines require further protection. For the HELSEA evaluation, which is discussed in Section 3.6.1.3.2.2 of the SSAR, no particular break points are evaluated. Breaks are postulated at any point in the piping system and any structure identified as necessary by the HELSEA evaluation are designed for worst-case loads. This was DFSER COL Action Item 14.1.3.3.7.3-1. SSAR Section 3.6.5.1, Item (8), states that the COL applicant will perform the HELSEA as described. This is acceptable.

Using this HELSEA evaluation, the staff finds that an adequate level of protection is provided to ensure that the safety-related function of components, systems, and equipment will not be adversely impacted by a postulated high-energy line break. Plant arrangement provides physical separation to the extent practical and the HELSEA evaluation ensures that no more than one redundant train can be damaged. If damage could occur to more than one division of a redundant safety-related system within 9.14 m (30 ft) of any high-energy piping, other protection devices such as barriers, shields, enclosures, deflectors, or pipe whip restraints are used. When necessary, the protection requirements are met through the use of walls, floors, columns, abutments, and foundations. Thus, the staff finds that the HELSEA criteria satisfy the intent of the SRP 3.6.2

Design of Structures, Components, Equipment, and Systems

guideline by ensuring that structures are adequately designed to withstand the consequences of a worst-case pipe break with no adverse impact on the safety-related function of systems, components, and equipment and are acceptable.

3.12.7.4 Conclusions

The staff concludes that the criteria for postulating pipe rupture and crack locations and the methodology for evaluating the subsequent dynamic effects resulting from these ruptures comply with SRP Section 3.6.2, meet GDC 4 and, are therefore, acceptable. The staff's conclusion is based on the evaluations in Section 3.12.3.7 of this report and the following:

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

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

The COL applicant is responsible for the arrangement of piping and restraints and the final design considerations for high- and moderate-energy fluid systems inside and outside the containment, including the RCPB. The COL applicant should use these staff-approved high-energy line break criteria and these guidelines to ensure that the SSCs important to safety that are in close proximity to the postulated pipe ruptures will be protected. Using these will ensure that the consequences of pipe ruptures will be adequately mitigated so that the reactor can be safely shut down and be maintained in a safe-shutdown condition in the event of a postulated rupture of a high- or moderate-energy piping system inside or outside the containment.

3.12.8 Leak-Before-Break Criteria

SSAR Section 3.6.3 and Appendix 3E provide a description of the evaluation procedures for an LBB methodology. Since no LBB analysis was submitted for

staff review and approval, the use of the LBB approach has not been pre-approved by the staff in the ABWR design certification phase. Hence, it is a design option for the COL applicant to consider in lieu of performing high-energy line break analyses as discussed in Section 3.12.7 of this report.

GDC 4 permits the application of the LBB methodology to piping systems. It states, in part, that "dynamic effects associated with postulated pipe ruptures in nuclear power units may be excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping." The analyses referred to in GDC 4 (52 Federal Register, 41288-41295, October 27, 1987) should be based on specific plant data, such as piping geometry, materials, piping loads, and pipe support locations. The staff must review the LBB analyses for specific piping designs before the applicant can exclude the dynamic effects from the design basis for the piping system.

The staff concludes that COL applicants seeking approval of the LBB approach for high-energy piping systems in the ABWR standardized plant must submit to the NRC staff an LBB plant-specific analysis in accordance with GDC 4. The staff recognizes that the LBB technology is continually evolving, therefore, the staff will review LBB requests for the ABWR plant on a case-by-case basis, using the staff's methodology and acceptance criteria in effect at the time of the submittal.

3.12.9 Generic Piping Design ITAAC

In Section 3.3 of the CDM, dated June 17, 1992, GE provided its ITAAC for piping design. Table 3.3 therein identified 12 certified design commitments (design elements) for the ABWR piping design and the corresponding ITAAC. In a meeting held at GE's offices in San Jose, California, on January 11 through 21, 1993, the staff and GE reached agreement on the resolution of comments from the industry/NUMARC on the generic piping design ITAAC (also referred to as the Piping DAC). The piping design description has been substantially expanded to include additional certified design commitments. The ITAAC design commitments have been reduced in number to consolidate those design commitments that are implicit in the ASME Code, Section III, requirements and to eliminate some design criteria that were deemed not appropriate for inclusion into ITAAC. GE gave the NRC additions and changes to the SSAR to support the piping DAC/ITAAC changes.

The staff's evaluation of the certified design commitments and ITAAC for piping design is given in the following sections.

3.12.9.1 Fatigue

GE provided a certified design commitment that the piping will be designed for a fatigue life of 60 years. The COL applicant will perform a fatigue analysis for ASME Code Class 1 piping systems in accordance with the applicable requirements of ASME Code, Section III. Section III rules will be followed for ASME Code Class 2 and 3 piping, using a stress range reduction factor of 1.0 for those piping systems expected to experience less than 7000 thermal cycles in their 60-year design life.

The COL applicant will be required to ensure that the fatigue analysis meets the ASME Code requirements for the 60-year design life. The acceptance criteria for the fatigue design of ASME Code Class 1 piping will be that the cumulative usage factor is less than 1.0 as specified in the ASME Code, Section III.

The staff finds that a fatigue analysis of safety-related piping is a necessary certified design commitment to ensure the integrity of the RCPB and the ability of the piping systems to perform their safety function for a 60-year design life. The design acceptance criterion for a cumulative usage factor to be less than 1.0 is consistent with current ASME Code requirements for fatigue evaluation as stated in Subparagraph NB-3222.4. The COL applicant will consider in its fatigue analysis the environmental effects as discussed in Section 3.12.5.7 of this report.

The inspection of the ASME Code certified stress report, including the fatigue analyses ensures that the ASME Code requirements for fatigue will be satisfied. However, in the DFSE, the staff reported that an additional certified design commitment is needed for any ASME Code Class 2 and 3 piping system that is expected to experience 7000 or more thermal stress cycles in its 60-year design life. For any such piping, the COL applicant should use a stress reduction factor of less than 1.0 as required by Subparagraph NC/ND-3611.2 of the ASME Code, Section III. In addition, if an ASME Code Class 1 fatigue evaluation is required for ASME Code Class 2 and 3 piping systems, as discussed in Section 3.12.5.8 of this report, a cumulative usage factor of 1.0 should be met and environmental effects should be considered. This was DFSE Open Item 14.1.3.3.9.1-1.

Section 3.9.3.1 of the SSAR provides information on loading combinations, design transients, and stress limits for ASME Code Class 1, 2, and 3 components and

supports. SSAR Section 3.9.3.1.19 states that the Class 1, 2, and 3 piping is constructed in accordance with the ASME Boiler and Pressure Vessel Code, Section III. For Class 1 piping, stresses are calculated on an elastic basis and evaluated in accordance with NB-3600 of the ASME Code, Section III. For Class 2 and 3 piping, stresses are calculated on an elastic basis and evaluated in accordance with NC/ND-3000 of the Code. From GE's commitment, the staff concluded that any Class 2 or 3 piping that is expected to experience 7000 or more thermal stress cycles must be designed to meet the allowable expansion stress range defined in NC/ND-3611.2(e) and based on the appropriate stress range reduction factor given in Table NC/ND-3611.2(e)-1 for the total number of full temperature cycles in its 60-year design life. The staff also concluded that for any piping system in which an ASME Code Class 1 fatigue evaluation will be performed, the evaluation must be performed in accordance with the requirements of NB-3653 and a cumulative usage factor of 1.0 must not be exceeded, considering the total number of stress cycles in its 60-year design life. In SSAR Section 3.9.3.1.1.7, GE commits to performing additional evaluations for environmental effects on fatigue design. The staff evaluation of this issue is discussed in Sections 3.12.5.7 and 3.12.5.8 of this report.

On the basis of these commitments, the staff concludes that there is reasonable assurance that the design of the ABWR piping systems will be adequately evaluated for fatigue effects.

3.12.9.2 Pipe-Mounted Equipment Allowable Loads

GE provided a certified design commitment that the loads imposed by the piping system on pipe-mounted equipment will meet the vendor allowable loads. The COL applicant will inspect the design documents and document that the as-designed interface loads meet the vendor's specified allowable loads. This was DFSE COL Action Item 14.1.3.3.9.2-1. SSAR Section 3.9.3.1.21, "Pipe-Mounted Equipment Allowable Loads," states that the COL applicant shall inspect the piping design reports and document that the pipe applied loads on attached equipment are less than the equipment vendor's specified allowable loads.

The staff finds it necessary to ensure that the calculated loads imposed by the piping on the equipment nozzles and other attachment interfaces are within the vendor's recommended allowable values. This verification will ensure that the equipment and supports will function as intended under normal operating, transient, and accident conditions.

Design of Structures, Components, Equipment, and Systems

On the basis of these commitments for a COL applicant to verify that the calculated piping loads are within the equipment and interface allowable loads, the staff concludes that there is reasonable assurance that the ABWR pipe-mounted equipment and piping attachment interfaces will adequately satisfy the vendor interface allowable limits to ensure that the equipment can perform its intended safety functions under normal, operating, transient, and accident loading conditions.

3.12.9.3 Piping Analysis Methods

GE provided a certified design commitment that would require that the analytical methods and load combinations be referenced or specified in a certified stress report. The COL applicant will use a suitable dynamic analysis method or an equivalent static load method in the analysis of the piping system.

The analytical methods to be used to complete the ABWR piping design will ensure the pressure integrity, structural integrity, and the functional capability of the piping system under normal operating and accident loading conditions and will use a suitable dynamic analysis or an equivalent static analysis method as approved by the staff. The analysis methods approved by the staff for the ABWR piping design are discussed in Section 3.12.3 of this report. The key analysis input parameters approved by the staff for the ABWR piping analysis are discussed in Section 3.12.5 of this report.

On the basis of these commitments for a COL applicant to use the above staff-approved analysis methods and input parameters for the ABWR piping analyses, the staff concludes that there is reasonable assurance that the ABWR analysis methods are adequate to ensure the pressure integrity, structural integrity, and functional capability of the piping.

3.12.9.4 High-Energy Line Break Analysis

GE provided a certified design commitment that would require an analysis demonstrating that safety-related systems, components, and structures are protected against the dynamic effects associated with the postulated rupture of high-energy piping systems.

The COL applicant will prepare a pipe rupture analysis report or an LBB analysis report to verify that the safety of the plant will not be adversely affected by the dynamic effects resulting from the postulated pipe breaks. For those impacted components needed to safely shut down the plant, the ASME Code requirements for faulted plant conditions and operability limits must be met. Pipe rupture mitigation devices (e.g., pipe whip restraints and

jet impingement shields) will be used to restrain the whipping pipe and deflect the blowdown loads. The COL applicant will inspect to verify the existence of a pipe break analysis report or LBB report as stated in SSAR Section 3.6.5.1. The COL applicant will also inspect the as-built high-energy pipe break mitigation features as discussed in SSAR Section 3.6.4.

A pipe rupture analysis will be completed by the COL applicant to demonstrate that safety-related SSCs will be protected against the dynamic effects of a postulated pipe break, using the methods described in Sections 3.6.2 and 3.12.7 of this report. As an alternative, the COL applicant may submit a request for staff approval to eliminate breaks, using an LBB approach as discussed in Section 3.12.8 of this report.

On the basis of these commitments for a COL applicant to perform high-energy line break analyses, using the staff-approved analysis methods, and to verify the results of the analyses, the staff concludes that there is reasonable assurance that the safety-related SSCs in the ABWR are adequately protected against the dynamic effects of postulated HELBs.

3.12.9.5 Functional Capability

GE provided a certified design commitment that all ASME Code Class 1, 2, and 3 piping systems essential for the safe shutdown of the plant must be designed to ensure that they will maintain sufficient dimensional stability to perform their required function under all loading conditions. In Section 3.12.5.12 of this report, the staff evaluated the stress limits GE specified to ensure the functional capability of safety-related piping systems. In no case will the piping stress exceed the primary stress limits designated for Service Level D in the ASME Code, Section III. The Service Level D limits are $3.0 S_m$ (not to exceed $2.0 S_y$) for ASME Code Class 1 piping and $3.0 S_h$ (not to exceed $2.0 S_y$) for Class 2 and 3 piping.

The staff finds that the limits specified by GE to ensure functional capability of piping as discussed in Section 3.12.5.12 of this report are acceptable. The use of Service Level D limits (not to exceed $2.0 S_y$) are consistent with the staff recommendations as documented in NUREG-1367, "Functional Capability of Piping Systems," for ensuring the functional capability of piping systems that were based on high-level dynamic tests sponsored by the EPRI and the NRC staff.

On the basis of these commitments for a COL applicant to limit piping stresses in the certified stress report to the design acceptance criteria discussed above, the staff concludes that there is reasonable assurance that the piping

is capable of performing its safety function under all normal operating, transient, and accident conditions.

3.12.9.6 Analytical Modeling of Piping

GE provided a certified design commitment to verify the piping analysis modeling technique for the computer code to be used by the COL applicant to complete its piping stress analyses. The piping analysis model must address the key parameters needed to ensure adequate static and dynamic characteristics of the piping system. The key parameters for the piping model are discussed in Section 3.12.4.2 of this report. The computer program and the modeling techniques must be evaluated, using the NRC benchmark program discussed in Section 3.12.4.3 of this report.

The COL applicant will verify the sufficiency of the computer code and modeling techniques in conjunction with the piping benchmark program.

On the basis of these commitments for a COL applicant to verify that the piping benchmark results are within the acceptable range of values specified in the benchmark program, the staff concludes that there is reasonable assurance that the computer code and analytical modeling techniques to be used to complete the ABWR piping design and analyses are adequate.

3.12.9.7 ASME Code Classification

GE provided a certified design commitment that the ABWR piping, its appurtenances, and its supports must satisfy the ASME Code class, seismic category, and QG requirements commensurate with their classification. The COL applicant will review the ASME Code-required design documents for installed components to verify the completion of a certified stress report and related Code-required documents.

On the basis of this commitment for a COL applicant to verify the existence of Code-required documents, the staff concludes there is reasonable assurance that the piping and its subcomponents will be adequately designed, fabricated, and examined in accordance with the applicable ASME Code requirements.

3.12.9.8 Fracture Toughness

GE provided a certified design commitment that the piping systems made of ferritic material must not be susceptible to brittle fracture. Only intrinsically tough grades of ferritic materials will be used. The COL applicant will perform fracture toughness tests in accordance with the requirements of the ASME Code, Section III.

On the basis of these requirements for a COL applicant to verify that ferritic materials satisfy the requirements of ASME Code, Section III, the staff concludes that there is reasonable assurance that the material for piping systems will be adequately specified to preclude brittle fracture under pressure loadings for the expected service conditions.

3.12.9.9 Cracking in Stainless Steel Piping

GE provided a certified design commitment that the fabrication process for piping systems made of austenitic stainless steel will be selected to minimize the possibility of cracking during their 60-year design life. Special chemical, fabrication, handling, welding, and examination requirements will be satisfied to minimize the potential for cracking. The guidelines in NUREG-0313, Revision 2, will be followed as stated in SSAR Section 5.2.3.4.

On the basis of these commitments for a COL applicant to use the material and processes that satisfy the ASME Code and special requirements, the staff concludes that there is reasonable assurance that the austenitic stainless steel piping systems will be adequately fabricated to minimize the potential for cracking during service.

3.12.9.10 As-Built Piping Verification

GE provided a certified design commitment that the as-built piping system must be reconciled with the certified piping design. The COL applicant will perform an inspection to verify the pipe routing configurations, as well as the location, size, and orientation of piping supports, valves, and equipment, and to identify deviations from the as-designed condition as described in SSAR Section 3.9.3.1.20. The piping configuration and component location, size, and orientation will be within the specified tolerances. Deviations (outside the tolerances) will be evaluated to ensure that the vendor-allowable loads and ASME Code, Section III, stress limits are satisfied.

The tolerances to be used for the reconciliation of the as-built installation of piping systems will be obtained from the EPRI report, "Guidelines for Piping System Reconciliation (NCIG-05, Revision 1)," NP-5639 dated May 1988. The staff's acceptance of this approach is documented in a letter from G. Arlotto (NRC) to W. Weber (Nuclear Construction Issues Group) dated February 3, 1988. The staff's endorsement of the EPRI report is subject to the restriction that (1) the acceptable as-built piping tolerances not be increased beyond those stated in NP-5639 and (2) they be limited to the piping systems analyzed using linear-elastic methods and qualified on the basis of the staff-approved design criteria specified in this report.

Design of Structures, Components, Equipment, and Systems

On the basis of these commitments for a COL applicant to verify that the installation tolerances are satisfied and that all deviations are reconciled using the staff-approved methods and design acceptance criteria discussed in this report, the staff concludes that there is reasonable assurance that the ABWR piping systems will be constructed in accordance with the design documents.

3.12.9.11 Pressure Integrity

GE provided a certified design commitment that the ASME Code Class 1, 2, and 3 piping will be designed to the requirements of ASME Code, Section III, as discussed in Section 3.12.2.1 of this report to retain its pressure integrity for its 60-year design life.

The COL applicant will inspect the ASME Code-required documents to verify that ASME Code, Section III, requirements are satisfied for ASME Code Class 1, 2, and 3 piping systems. The COL applicant will perform hydrostatic pressure tests of the ASME Code Class 1, 2, and 3 piping in accordance with the ASME Code requirements. In addition, the COL applicant will ensure that the low-pressure piping systems that interface with the RCS pressure are adequately designed to withstand full RCS pressure in the event of an intersystem LOCA as discussed in Section 3.12.5.19 of this report.

On the basis of these commitments for a COL applicant to perform these inspections and tests, the staff concludes that the ASME Code Class 1, 2, and 3 piping systems will be designed, constructed, and tested to ensure their pressure integrity in service under normal operating, testing, transient, and accident conditions.

3.12.9.12 Interferences

GE provided a certified design commitment that piping will be designed with adequate clearances to preclude interferences with nearby SSCs resulting from piping displacements.

The COL applicant will verify that the maximum calculated pipe deflections under normal operating, transient, and accident conditions do not exceed the minimum specified clearances between the piping and nearby SSCs as stated in SSAR Section 3.9.3.1.22.

On the basis of these commitments for a COL applicant to verify that maximum calculated pipe deflections are within the minimum specified clearances or do not impact nearby SSCs, the staff concludes that there is reasonable assurance that the piping deflections under normal operating, transient, and accident conditions will not cause interferences with nearby SSCs.

3.12.9.13 Erosion-Corrosion

GE provided a certified design commitment that piping systems will be designed to minimize the effects of erosion-corrosion. Erosion-corrosion will be controlled as discussed in SSAR Section 5.2.3.2.2.3. The EPRI-developed code CHECMATE will be used for two-phase environments to predict corrosion rates and to identify areas where design improvements to the material selection, hydrodynamic conditions, oxygen content, and temperature may be required to ensure adequate margin for extended piping performance.

On the basis of this certified design commitment for a COL applicant to evaluate the erosion-corrosion effects in piping systems, the staff concludes that there is reasonable assurance that the effects of erosion-corrosion are minimized in the piping design for the 60-year plant life.

3.12.9.14 Conclusions

In the DFSER, the staff requested that GE revise the CDM for piping design to ensure that all its certified design commitments in Table 3.3 of the CDM are included in the design description. This was DFSER Open Item 14.1.3.3.9.13-1. In a revision to Section 3.3, "Piping Design," of the Tier 1 CDM document, GE expanded the design description to include the certified design commitments. GE has also included this information in the final CDM. On the basis of this evaluation, DFSER Open Item 14.1.3.3.9.13-1 is resolved.

The staff concludes that the certified design commitments and ITAAC for piping design incorporate the staff-approved piping design criteria and analysis methods for ensuring that the piping systems will be adequately designed to perform their safety-related functions for all postulated combinations of normal operating, operating transient, and accident conditions, and provide reasonable assurance that the piping systems are built in conformance with the certified design.

3.12.10 Overall Conclusions

The staff concludes that GE has provided sufficient information in the SSAR for the staff to reach a safety determination in ABWR piping and pipe support design. The staff's conclusion is based on the following:

- (1) GE satisfies 10 CFR Part 50 requirements by identifying applicable codes and standards, design and analysis methods, design transients and load combinations, and design limits and service

Design of Structures, Components, Equipment, and Systems

conditions to ensure adequate design of all safety-related piping and pipe supports in ABWR for their safety functions.

- (2) GE satisfies 10 CFR Part 52 requirements by providing reasonable assurance that the piping systems will be designed and built in accordance with the certified design. The implementation of these preapproved methods and satisfaction of the acceptance criteria will be verified through the performance of the ITAAC by the COL applicant to ensure that the as-constructed piping systems are in conformance with the certified design for their safety functions.

- (3) GE satisfies 10 CFR Part 100, Appendix A, requirements by designing the safety-related piping systems, with reasonable assurance to withstand the dynamic effects of earthquakes with appropriate combination of other loads of normal operation and postulated events with adequate margin for ensuring their safety functions.

Any change to the commitments involving the piping analysis methodology discussed in Section 3.12 of this report would involve an unreviewed safety question and, therefore, requires NRC review and approval prior to implementation. Any requested change to these commitments must either be specifically described in the COL application or be submitted for license amendment after COL issuance.

4 REACTOR

4.1 General

The reactor assembly consists of the reactor pressure vessel, pressure containing appurtenances that include control rod drive (CRD) housings, in-core instrumentation housing, and the head vent and spray assembly. The reactor pressure vessel includes the reactor internal pump (RIP) casing and flow restrictors in each of the steam outlet nozzles and the shroud support and pump deck that form the partition between the RIP suction and discharge. The design and description of the reactor pressure vessel are discussed in Section 5.3 of this report.

The major reactor internal components are the core (fuel, channels, control blades, and instrumentation), the core support structure (including the shroud, top guide, and core plate), the shroud head and steam separator assembly, the steam dryer assembly, the feedwater spargers, and the core flooding spargers. Except for the Zircaloy in the reactor core, these reactor internals are stainless steel or other corrosion-resistant alloys. The fuel assemblies (including fuel rods and channel), control blades, shroud head and steam separator assembly, and steam dryers and in-core instrumentation dry tubes are removable when the reactor vessel is opened for refueling or maintenance.

A fuel and control rod design and core loading pattern typical of many currently operating boiling water reactors (BWRs) was used as the basis for the core design for the first cycle and for the system response analysis for standard safety analysis report (SSAR) Chapters 6 and 15. These elements of the core design meet criteria approved by the Nuclear Regulatory Commission (NRC) as presented in SSAR Appendices 4B, 4C, and 4D.

4.2 Fuel System Design

The staff reviewed the fuel system design in accordance with Standard Review Plan (SRP) Section 4.2, which includes the acceptance criteria of General Design Criteria (GDC) 10, 12, and 27, as discussed below and the reference to fuel designs approved by the NRC or to fuel that meets acceptance criteria approved by the NRC for GE Nuclear Energy (GE) fuel (NEDE-31152P, "GE Fuel Bundle Designs Evaluated with GESTAR - Mechanical Analysis Bases," SSAR Appendix 4B and SSAR Reference 4.2-2).

- (1) GDC 10 requires that acceptable fuel design limits be specified that are not to be exceeded during normal operation, including the effects of anticipated operational occurrences.
- (2) GDC 12 requires that power oscillations which could result in conditions exceeding specified

acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed.

- (3) GDC 27 requires that the reactivity control systems have a combined capability, in conjunction with poison addition by the emergency core cooling system, of reliably controlling reactivity changes under postulated accident conditions, with appropriate margin for stuck rods.

The fuel for the advanced boiling water reactor (ABWR) is similar in design, including most geometrical and material details, to the fuel commonly used for reloading most BWR cores. The fuel design bases, limits, analysis methodologies, and evaluations for the ABWR also are the same as those used and approved for initial loading and reloading of previous BWR cores. The ABWR fuel and core design provides the basis for the representative first cycle power distributions and rod patterns as presented in SSAR Appendix 4A and as used in the safety analyses of SSAR Chapters 6 and 15. The ABWR Cycle 1 core design, along with the fuel and control rod design (see below) are designated Tier 2 items and no inspections, tests, analysis, and acceptance criteria (ITAAC) are involved. However, pertinent elements of the fuel and control rod design criteria (SSAR Appendices 4B and 4C) are Tier 1 items and are included in the design descriptions. GE submitted the design description for the fuel and control rod. The adequacy and acceptability of these design descriptions are evaluated in Section 14.3 of this report.

Although the areas are Tier 2, the staff review has concluded that the fuel and control rod design criteria (of SSAR Appendices 4B and 4C), the first cycle fuel, control rod and core design, and the methods used to analyze these components may not be changed without prior NRC review and approval. The specific fuel, control rod, and core designs presented in SSAR Chapter 4 will constitute, based on this staff review and approval, an approved design that may be used for the COL first cycle core loading, without further NRC staff review. If any other design is requested for the first cycle, the COL applicant will be required to submit for staff review that specific fuel, control rod, and core design analysis and corresponding safety analysis described in SSAR Chapters 6 and 15. The review for the fuel, control rod, and core design will be based on fuel and control rod design criteria as described below.

Although it is not specifically referenced in the SSAR, GE customarily presents relevant generic information relating to core initial and reload cycle designs and analyses, including methodology related to fuel and control rod thermal-mechanical, nuclear, and thermal-hydraulic phenomena, in the licensing topical report NEDE-24011-P,

Reactor System

"General Electric Standard Application for Reactor Fuel" (GESTAR II) (proprietary). Currently approved GE BWR fuel designs are described in the GE report NEDE-31152P, "General Electric Fuel Bundle Designs Evaluated with GESTAR-Mechanical Analysis Bases" (proprietary). GE states in SSAR Section 4.2 that BP 8 x 8R fuel is used for the reference core design. The BP 8 x 8R is a standard fuel design frequently used in current BWRs for cycle reloads. An eight-by-eight array of prepressurized fuel pins and a barrier coating on the inner surface of the fuel cladding are used. A discussion and NRC staff approval of features of the fuel may be found in GESTAR II, Supplement for United States, Appendix C, and in Reference 4.2-2 of the ABWR SSAR. The fuel assembly descriptions are in Section AY of NEDE-31152P, Volume 3, Revision 3. These reviews covered all of the areas in SRP Section 4.2. Compliance of the fuel design with the fuel design criteria (discussed below) is presented in SSAR Appendix 4D, and the design is shown to meet all requirements. This fuel is acceptable for the ABWR.

GE provided the primary reference to acceptable fuel design for the ABWR in SSAR, Appendix 4B. This appendix provides a set of acceptance criteria to be satisfied by new fuel designs for the ABWR. These criteria were developed from preceding generic work in response to the NRC staff generic request that such acceptance criteria be established by each fuel vendor. With the NRC approval of the generic criteria, new fuel designs (or changes) satisfying the criteria would not require explicit staff review for current reactors.

In response to the NRC staff generic request, GE submitted Amendment 22 to GESTAR II, containing proposed fuel licensing acceptance criteria for current BWRs. These criteria include considerations of fuel thermal-mechanical, nuclear, and thermal-hydraulic aspects of design analyses. The NRC staff and the Committee to Review Generic Requirements reviewed and approved these criteria. The staff safety evaluation report (SER) accepting these criteria can be found in GESTAR II, Revision 10, Supplement for United States, Appendix C. However, the staff has noted deficiencies in the generic fuel licensing criteria. In its audit summary, "Audit Team Audit of GE II Fuel Design Compliance With NEDE-24011-P-A," dated March 25, 1992, the staff noted the lack of a burnup limit requirement in the fuel criteria at that time. A parallel fuel burnup problem area (DFSER Confirmatory Item 4.2-1) for the ABWR is discussed below.

For the ABWR, GE has not directly referenced GESTAR II or the fuel criteria amendment. Instead GE has provided SSAR Appendix 4B which provides a similar set of criteria for the ABWR. Since the Appendix 4B

criteria are essentially identical to the criteria approved by the staff for GESTAR II, they are generally acceptable. However, some additions or restrictions are necessary. The staff review of the Appendix 4B criteria and the staff audit of the generic criteria have indicated that the following restrictions are necessary for the long-term use of the criteria:

- NRC-approved analytical models and analysis procedures of General Criterion (1) to be used without further review must be limited to those referenced in GESTAR II, Revision 10, or previous revisions. Methods developed and approved in later GESTAR II revisions will not automatically apply to the ABWR and will have to be specifically reviewed and approved for ABWR use.
- Fuel burnup limits must be specified and justified on the basis of material properties versus exposure data for each fuel type used in the ABWR and may be extended only with NRC review and approval. This was identified as DFSER Confirmatory Item 4.2-1.

In response to the staff request for burnup limit and review statements in the fuel criteria, GE stated that such a criterion is unnecessary and not a safety issue. However, they proposed the following statement for the SSAR.

Burnup limits will be specified for each fuel type used in the ABWR. The current maximum exposure limit for any GE fuel design is 70 Giga watts days metric ton of uranium (GWd/MTU) (6,048 MJ/gU (3,214 * 10⁶ BTU/LbU) peak pellet exposure (~60 GWd/5,184 MJ/gU (2,755*10⁶ BTU/LbU or 60 GWd/MTU) rod average exposure). Any extension to this maximum exposure limit in excess of 10 GWd/MTU will be submitted to the NRC for review and approval based on the available supporting materials properties vs. exposure information and planned surveillance program. In no event will the GE fuel design maximum exposure limit required by the NRC be lower than the maximum of all exposure limits approved by the NRC for LWR fuel vendors.

The staff found that the proposed fuel burnup limit in this submittal is higher than that previously approved for GE (5,184 MJ/gU (2,755*10⁶ BTU/LbU or 60 GWd/MTU), peak pellet), that an unreviewed extension of 10 GWd/MTU is excessive, and that limits approved for other vendors do not necessarily apply to GE fuel without specific review for GE. The staff considers the burnup limit a safety question and has several fuel operating concerns at burnup levels above those currently approved for BWRs (about 5,184 MJ/gU (2,755*10⁶ BTU/LbU or

60 GWd/MTU) peak-pellet burnup). These concerns impact normal operation, off-normal transients, and accidents.

A brief summary of the concerns are:

- no prototypical LWR operating data above about 5,357 MJ/gU (2,847*10E6 BTU/LbU or 62 GWd/MTU)
- no fuel transient data above about 3,974 MJ/gU (2,112*10E6 BTU/LbU or 46 GWd/MTU)
- significant drop in cladding ductility observed at about 5,184 MJ/gU (2,755*10E6 BTU/LbU or 60 GWd/MTU)
- decrease in fuel thermal conductivity and changes in other physical properties
- changes in loss-of-coolant accident (LOCA) rod behavior at higher burnup levels
- fission gas release

Other issues that need to be addressed on a design specific basis for an extension in fuel burnup are the following:

- assembly and cladding corrosion
- fuel rod and assembly axial growth
- grid spacer spring relaxation

Since GE has provided a fuel burnup limit, the staff considers DFSEER Confirmatory Item 4.2-1 resolved. However, GE has been requested to augment its proposed fuel design criteria for the ABWR to include fuel burnup limits and to indicate that these limits may be extended only with NRC review and approval. In its later submittal on this subject, GE omitted previous objections that the staff has not imposed explicit burnup limits in the past. (They had appeared indirectly in the maximum average planar linear heat generation rate (MAPLHGR) TS maximum burnup listing, but this would effectively disappear in the core operating limit report.) However, they proposed a peak burnup limit of 6,048 MJ/gU (3,214*10E6 BTU/LbU or 70 GWd/MTU (peak pellet) which is greater than the limit previously approved by the staff (60), and they also proposed that extensions to approved values that would not need staff review and approval should be at least 10 GWd/MTU, which the staff considers to be highly excessive. This was an Open Item F4.2-1 for the ABWR review. In response to this open item, GE provided changes to the fuel licensing acceptance criteria Section 4B.3(2)(j), "Submittal

Supporting Accelerated ABWR Schedule-Response to Open Item F4.2-1," dated February 4, 1994, which now states that (1) fuel burnup limits will be specified for fuel used in the ABWR design, (2) the current limit for the ABWR fuel is 60 GWd/MTU rod average exposure, and (3) any extension of this limit will be submitted to the NRC for review and approval. These changes provide an acceptable resolution of the need for burnup restrictions indicated in the staff review. The 60 GWd/MTU limit is acceptable based on the staff review of high performance data for GE fuel during the NRC audit of the fuel design process for the GE 11 fuel referenced above. The data supporting the high burnup performance that were examined during the audit included GE 8 x 8 fuel of the type used in the ABWR reference core. This submittal resolved Open Item F4.2-1.

With approval of the ABWR fuel criteria, new ABWR fuel designs (or changes) satisfying the criteria would not require explicit staff review, other than that required by its use by a COL applicant for the first cycle core loading.

Similar to the presentation of the ABWR fuel design, GE has provided a specific design for the control rod. This design was used in the safety analyses of SSAR Chapters 6 and 15. GE also has proposed control rod design criteria, similar in concept to those for the fuel designs, to be used as a basis for the proposed control rods or future new design submittals. Just as for the fuel design, the specified control rod design used in the ABWR safety analyses will constitute, based on the staff review and approval, an approved design that may be used by the COL applicant for the first cycle without further staff review. If the COL applicant changes the design, the staff will require new submittals for review and approval.

The ABWR control rod design has, for the most part, the same geometrical and material design characteristics of those approved and used for current reactors for the first cycle and for replacement, including current improvements for corrosion-cracking control. The ABWR design differs significantly from current control rod designs only in that it does not provide a velocity limiter for the rod drop accident event. GE believes that this event is not credible because information on the blade-drive uncoupling signal will be provided in the new ABWR rod drive design. In any case, the velocity limiter has become much less significant with axial reactivity shaping Gadolinia burnable poison in the fuel than with the early fuel designs that required the velocity limiter. Because of the burnable poison distribution or, later in the cycle, burnup distribution, most of the available reactivity increase from the dropped rod is inserted in a short distance and appears in the transient without significant dependence on the velocity limiter action. NRC consultant analyses (Brookhaven National Laboratory (BNL) - NUREG-36891,

Reactor System

"Effects of Rod Worth and Drop Speed on the BWR Off-Center Rod Drop Accident," and ANS Transactions, November 1985) have shown little sensitivity to drop velocity. Thus removal of the velocity limiter is acceptable.

In SSAR Appendix 4C, GE submitted the set of control rod licensing acceptance criteria for the ABWR. These criteria describe the safety-related functional performance requirements for the control rods. The staff reviewed these criteria in Appendix 4C and proposed additions and modifications necessary to provide acceptable criteria. The changes which were submitted in the GE response were (1) removal of a statement from the general criteria indicating that a control rod design meeting the criteria did not require specific NRC review and (2) indicating that surveillance programs are to be implemented when changes in design features could impact the control function. Also added to the bases for the criteria were (1) inclusion of irradiation effects to the stress and strain limits, (2) further details of inspection of lead depletion rods with new design features, and (3) inclusion of crudding, crevices, and stress corrosion effects upon control rod material. The GE responses were responsive and satisfactory, and these criteria, as revised, were acceptable. This was DFSER Confirmatory Item 4.2-2. GE incorporated the appropriate changes have been in SSAR Amendment 31. The staff finds it to be acceptable. Therefore, DFSER Confirmatory Item 4.2.2 is resolved.

Current GE control rod designs that have been reviewed and approved by the NRC are suitably adaptable for the ABWR (when modified by the elimination of the velocity limiter). GE has provided a description of the reference control rod blade design in SSAR Section 4.2. GE has provided an evaluation of the design via a comparison with the Appendix 4C criteria. GE has revised SSAR Section 4.2.3.2.2.1 to indicate they have completed the evaluation of the control rod design, based on the criteria of SSAR Appendix 4C. GE has stated that the control rod evaluations described in Section 4.C.3 have been completed for the reference control rod design, and the criteria are satisfied. The control rod evaluation, performed by GE using proprietary Japanese data, was not submitted for staff review but is available for audit. It was not necessary to audit this data because the reference control rod blade design is similar to previously approved GE control rod designs. As discussed above, the staff finds GE's conclusions to be reasonable. The control rod design and evaluation were Open Item 14 in the draft safety evaluation report (DSER), and DFSER Confirmatory Item 4.2-3, respectively. Both items are now resolved.

The staff concludes that with approval of the specific fuel design type and of the fuel licensing acceptance criteria of SSAR Section 4B, as indicated above, and with the staff acceptance of the control rod design criteria in SSAR Section 4C, the ABWR core design approach meets the requirements of SRP Section 4.2 and is acceptable.

4.3 Nuclear Design

The staff reviewed the nuclear design in accordance with SRP Section 4.3 which includes the relevant requirements of the GDC related to the reactor core and reactivity control systems. The relevant requirements are as follows:

- (1) GDC 10 requires that acceptable fuel design limits be specified that are not to be exceeded during normal operation, including the effects of anticipated operational occurrences.
- (2) GDC 11 requires that in the power operating range, the prompt inherent nuclear feedback characteristics tend to compensate for a rapid increase in reactivity.
- (3) GDC 12 requires that power oscillations which could result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed.
- (4) GDC 20 requires automatic initiation of the reactivity control systems to assure that acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and to assure automatic operation of systems and components important to safety under accident conditions.
- (5) GDC 25 requires that no single malfunction of the reactivity control system (this does not include rod ejection or dropout) causes violation of the acceptable fuel design limits.
- (6) GDC 26 requires that two independent reactivity control systems of different design be provided, and that each system have the capability to control the rate of reactivity changes resulting from planned, normal power changes. One of the systems must be capable of reliably controlling anticipated operational occurrences. In addition, one of the systems must be capable of holding the reactor core subcritical under cold conditions.
- (7) GDC 27 requires that the reactivity control systems have a combined capability, in conjunction with

poison addition by the emergency core cooling system, of reliably controlling reactivity changes under postulated accident conditions, with appropriate margin for stuck rods.

- (8) GDC 28 requires that the effects of postulated reactivity accidents neither result in damage to the reactor coolant pressure boundary greater than limited local yielding, nor cause sufficient damage to impair significantly the capability to cool the core.

In SSAR Section 4.3, GE describes how the ABWR meets GDC 10, 11, 12, 20, 25, 26, 27, and 28 and the other requirements of SRP Section 4.3 by direct reference to the fuel licensing acceptance criteria in SSAR Appendix 4B. Appendix 4B contains proposed criteria on fuel design and on related neutronic and thermal-hydraulic aspects of the fuel design. As discussed in Section 4.2 of this report, the staff approved these criteria in its review of the ABWR fuel system design.

As discussed in Section 4.2 of this report, GE used a specific core design for the first cycle and for the ABWR system response analysis and provided specific designs for first core fuel assembly enrichment patterns based on the core design. GE has also developed, with staff approval, a slightly revised fuel assembly enrichment pattern with neutronic parameters falling within the bounds of the reference design but with decreased local power peaking. This is to be used for the first cycle as required by the staff review of the fuel misorientation event, as discussed in Section 15.3(2) of this report. In SSAR Appendix 4A, GE describes use of an example control rod pattern throughout a cycle and the resulting power distributions for the cycle. The core operating limits on process variables determined from the reference core safety evaluation are incorporated in the core associated technical specifications (TS). SSAR Chapters 6 and 15 provide the analyses to satisfy the acceptance criteria for all design-basis transients and accidents initiated from worst-case steady-state operating conditions within TS operating limits for the design life of the core.

The core design and control rod pattern operations are generally similar to current BWR designs. They only differ in relatively small ways, in details of the geometry and operating limits, which result in small differences in neutronic parameters and characteristics. These differences are generally in a direction of a more conservative core neutronic design and operation than, for example, a BWR/6. They are designed with slightly lower average power densities. The ABWR control rod pitch will be slightly greater than current designs and this will result in a six percent larger core water to UO_2 volume ratio (a

moderation ratio increase), which in turn will result in a smaller (absolute magnitude) void reactivity coefficient throughout the range of power operating conditions. Both the lower average power density and smaller void reactivity coefficient tend to improve core thermal hydraulic stability and pressurization transient response. The moderation ratio increase tends to make the end-of-cycle low power (low temperature) moderator temperature reactivity coefficient less negative and possibly slightly positive. But the increase is not sufficient to cause an operational problem during startup or shutdown or a reactivity insertion problem, even assuming the maximum potential integrated positive reactivity is available for rapid insertion into the core.

In the operations examples, the ABWR control rods are withdrawn in patterns that are generally similar to current reactor withdrawal patterns using the banked position withdrawal sequence (BPWS). The primary difference is that, throughout the withdrawal patterns, multiple rods will be withdrawn simultaneously. Rather than pulling rods individually to provide the step banked patterns of BPWS, the rods in a group will be withdrawn simultaneously by the electric-drive motor systems. Similar to current BPWS patterns, about 1/8 of the 205 control rods in each of the 4 rod groups will be withdrawn to a 50-percent rod density checkerboard pattern (to about hot critical conditions). Because of the rod distribution in the patterns, the total group reactivity worths will not be very large (about 2 percent delta-K each for groups 2, 3, and 4 withdrawal when criticality might be expected). Operational control with this reactivity magnitude, and its accompanying differential reactivity worth, will be straightforward and not significantly different from current operation. Groups 3 and 4 (covering cold to hot critical) will be operated in a jog mode to avoid any approach to period scram levels. Beyond 50-percent rod density, the groups will be divided into groups of four or eight rods that will be moved simultaneously, with patterns similar to BPWS and operation similar to BWR/6s. The simultaneous withdrawal for the first 50-percent groups will reduce the maximum reactivity worth, which could be associated with a (postulated) rod drop accident to insignificant levels. In the power range, the ABWR examples use a "control cell core" strategy, which, combined with axial zoning of fuel enrichment and burnable poison, will result in very little movement of the control rods over most of the cycle in the normal power operation range. These various control rod operational characteristics are either very similar to current approved operation or are improvements. The operation and characteristics are acceptable.

Design criteria for the core neutronics for the ABWR are included in the fuel design criteria in SSAR Appendix 4B. The nuclear criteria are 4B.4, 1 through 8. These criteria

Reactor System

provide for fuel and moderator reactivity coefficients, shutdown margin, and fuel storage reactivity. These criteria are in GESTAR Amendment 22 and are discussed in Section 4.2 of this report in connection with the fuel design. They are acceptable.

The staff concludes that on the basis of (1) meeting the relevant fuel licensing acceptance criteria of SSAR Appendix 4B, (2) the general similarity to current operating GE BWR cores, and (3) the additional conservatism relative to current BWRs in the area of power density and less negative void coefficient, the ABWR nuclear design meets the requirements of SRP Section 4.3 and is acceptable.

4.4 Thermal Hydraulic Design

The NRC staff reviewed the thermal-hydraulic design of the ABWR reactor core in accordance with SRP Section 4.4. SSAR Section 4.4 describes how the ABWR meets GDC 10 and the other requirements of SRP Section 4.4 by direct reference to the fuel licensing acceptance criteria in SSAR Appendix 4B. As discussed in Section 4.2 of this report, SSAR Appendix 4B contains criteria on fuel design and related neutronic and thermal-hydraulic aspects of the fuel design. The staff approved these criteria after its review of the ABWR fuel system design.

GE provided a specific first cycle core design for the ABWR and used this design for system response analysis. As discussed in Section 4.2 of this report, the fuel used in this design is the same as currently approved fuel used in existing BWRs and the core fuel arrangement is similar to current fuel loadings. Alternate fuel loading designs for the first or subsequent cycles will have to conform to the fuel criteria of SSAR Appendix 4B, or be specifically reviewed and approved by the staff; for the first cycle any changes will have to be reviewed and approved by the NRC, along with any changes necessary for the analyses of SSAR Chapters 6 and 15. These criteria include the requirements for providing new thermal-hydraulic data such as critical power ratio correlations and limits on fuel stability characteristics.

The reference core design is generally similar to current BWR designs. It only differs in relatively small ways in details of the geometry and operating limits, and which result in small differences in neutronic and thermal-hydraulic parameters and characteristics. These differences generally reflect a more conservative core neutronic and thermal-hydraulic design and operation than, for example, a BWR/6. The core has a with slightly lower average power density. The fuel bundle and overall core

parameters for this design fall within the range of applicability of existing critical power correlations.

The only significant new design feature of the recirculation flow system is the use of the reactor internal pumps (RIPs). This system is discussed in Section 5.4.1 of this report. This change from current GE BWR design does not produce a significantly different power/flow operation map for the ABWR than that produced by current GE BWRs, although map parameters are slightly changed in some areas. GE provided a power/flow map, corresponding to the core design (discussed in SER Sections 4.2 and 4.3 of this report), to be used in systems response analyses. Each COL applicant should provide a plant-specific power/flow map at the time of application and for core reloads. This was DFSEER COL Action Item 4.4-1. GE has also included this action item in the SSAR and the staff finds it to be acceptable. Flow lines and control rod lines of the power/flow map generally will be within the range of operation currently permitted for BWR/6s, although natural circulation occurs at a slightly lower flow. The ABWR is expected to operate with a minimum pump speed of 30 percent of nominal full-flow speed, providing a minimum (all pumps operating) flow of about 40 percent. The ABWR design replaces the usual cavitation restriction region with a steam separation limit region to provide for acceptable moisture carryover. There is an interlock to reduce RIP speed to prevent operation in this restricted region. The maximum expected flow for the RIP system is about 115 percent of normal full flow. For the submitted design power/flow map, GE did not consider operating above the 102-percent power control rod line. These power/flow parameters generally are within expected bounds and present an acceptable region for normal operation. As indicated in SSAR Section 4.4.3.2, for normal operation at least 9 of the 10 RIPs are required to be operating. GE has provided a power/flow map for both 9 and 10 RIP operation.

In addition to the above boundaries of the power/flow map, there is a restricted region in the low-flow/high-power areas of the map that is intended to eliminate possible problems with thermal-hydraulic stability. (This is Region III on the map, generally considered to be approximately above the 80-percent control rod line and below the 40-percent flow line.) Operation will not be permitted in this region. Automatic startup logic will be programmed to block rod withdrawal in this region during low power and startup operation and to insert rods to withdraw from the region if entered inadvertently from higher power/flow regions.

In addition to this automatic control, the ABWR has several design features intended to improve the stability

status of the reactor that are not present in current reactors. These features are (1) smaller inlet orifices (loss coefficient doubled) to increase single-phase pressure drop, (2) wider control rod pitch to increase flow area and increase the moderator/fuel ratio and reduce the void reactivity coefficient, (3) more steam separators to reduce the two-phase pressure drop, (4) multiple RIPs on multiple power supplies along with minimum pump speed logic to reduce the likelihood of significant flow loss, and (5) regional local power range monitor (LPRM) and average power range monitor neutron flux time histories available for operator display to detect oscillations.

The control rod insertion to withdraw from Region III, if entered, will be provided by the selected control rod run-in system. This system is described in SSAR Section 7.7.1.2. Multiple control rods will be automatically inserted (simultaneously) to move to a stable region (e.g., from the 100-percent control rod line to below 25-percent power if two or more RIPs trip and flow is below the trip set point). Set points for power and flow will be adjustable and may be changed if stability analyses require it.

The NRC, in parallel with work by the BWR Owners' Group (BWROG) stability sub-committee, has for some time been reviewing generic questions relating to BWR stability. This review is still in progress. Questions concerning the need and the methods for improved stability control so as not to exceed core thermal-hydraulic safety limits and to understand and control, if necessary, adverse effects of power oscillation during anticipated transient without scram (ATWS) events are essentially complete. The ABWR design features indicated above are desirable for improving inherent stability and the system for control rod blocking and insertion to provide exclusion from Region III are consistent with some of the proposed long term solutions (LTS) for current BWRs. The BWROG has developed several (LTS that have been approved by the NRC for application to operating reactors. The region exclusion system for the ABWR is similar to the BWROG LTS Option IA. In response to a staff question requesting review of the recent LTS work and the possibility of its application to the ABWR, GE has stated that in addition to the region exclusion system, the BWROG Option III system, the oscillation power range monitor, which is based on the detection of oscillation signals by the LPRMs, will be implemented in the ABWR design (see SSAR Section 7.6.1.1.2.2). This LTS methodology has been accepted by the staff (letter from A. Thadani, NRC, to L. England, BWROG, "Acceptance for Referencing of Topical Reports NEDO-31960 and NEDO-31960 Supplement 1, 'BWR Owners Group Long-Term Stability Solution Licensing Methodology'" and is considered to be acceptable for the ABWR.

In the DFSE, the staff stated that the information presented by GE and the staff review of the subject of instability during ATWS transients was not complete. This was DFSE Open Item 4.4-1. GE subsequently submitted a report describing their analyses of oscillations during limiting ATWS transients susceptible to thermal-hydraulic instability and possibly large oscillations (Ref: GE ATWS Stability study, February 19, 1993, GE transmitted by letter dated February 22, 1993). GE performed calculations using the TPACG code, which has been used for stability studies for operating BWRs and has been audited by the staff. The ABWR calculations were reviewed by the staff and its consultants, and the stability characteristics of the ABWR have been explored using the Oak Ridge National Laboratory LAPUR code, commonly used by the staff for stability analyses. The LAPUR calculations showed, in accord with GE findings, that the ABWR is more stable than most current BWRs and will have smaller power oscillations, if any, at comparable operating conditions. The GE TRACG calculations appear to bound the worst expected ATWS conditions for instability and do not result in exceeding ATWS fuel failure limit criteria. The ABWR mitigating actions are similar to the BWROG proposed actions for current BWRs, and in SSAR Section 4.4.3.7 states that the ABWR emergency procedure guidelines will incorporate any changes recommended by the BWROG committee on thermal-hydraulic stability. There is automatic feedwater reduction and boron injection via the standby liquid control system. The feedwater runback and lowering of reactor level below the feedwater spargers are very effective in reducing oscillations magnitudes. The TRACG modelling and input for these calculations were found to be acceptable. The staff concludes that the issue of instability induced large power oscillations during ATWS has been properly addressed for the ABWR. GE has also included this information in the SSAR and the staff find it to be acceptable. On the basis of this evaluation, this item is resolved.

The ABWR design, as initially presented, did not include a loose-parts monitoring system (LPMS). However, in response to the staff position that an LPMS is required, GE submitted an LPMS general description, including a design basis, system description, system operation, safety evaluation, test, and inspection and application. This system is designed in conformance with Regulatory Guide (RG) 1.133, "Loose-Parts Detection Program for the Primary System of Light-Water-Cooled Reactors," Revision 1. The system includes sensors (accelerometers located at natural loose parts collection regions, e.g., steam outlet nozzle, feedwater inlet nozzle, control drive housings), signal conditioning, signal analysis, alarms, and calibration. The sensitivity is such that a sensor will be able to detect a metallic part between 0.1-14 kg (0.25 to

Reactor System

30 lbs) with a kinetic energy of 0.7 joule (0.5 ft-lb) on an inside surface within 1 m (3 ft) of a sensor. There will be provisions for online channel checks and functional tests and offline calibration. The system is designed to meet the seismic and environmental operability recommendations of RG 1.133, Revision 1. GE has provided an ITAAC for the LPMS as part of its Tier 1 certified design material submittal. It provides for design commitment for detector locations and sensitivity with appropriate accompanying inspections, tests, and criteria. In the DFSER, the staff stated that, as DFSER Open Item 4.4-2, Certified Design Commitment 1 in the LPMS ITAAC should be expanded to explicitly state that the LPMS design is consistent with the requirements of RG 1.133, Revision 1. This has been done; therefore, DFSER Open Item 4.4-2 is resolved. GE has submitted the Design Description, the ITAAC for the LPMS. The adequacy at acceptability of the ABWR Design Description and ITAAC are evaluated in Section 14.3 of this report. On the basis of this evaluation, this open item is resolved.

Core flow patterns are expected to be uniform at the core inlet during normal operations as a result of flow distributions from the downcomer through the RIPs, into the lower plenum, up through the orifices of the lower core plate, and into the fuel assemblies. TS will require for normal operation that at least 9 of the 10 RIPs are operating. Operation with fewer than 9 RIPs operating will require supporting analyses and justification by the COL applicant. This was DFSER TS Item 4.4-1. GE asserts that, with the allowed number of RIPs inoperable, pump operation will be close to normal and bounded by one recirculation loop operation in current jet pump BWRs (with, in effect, half the pumps out) for which there are no restrictions other than similar-type LOCA power-density restrictions. The restriction to no fewer than 9 RIPs operating is a recent modification of SSAR Section 4.4 and of the TS, and therefore, DFSER TS Item 4.4-1 is resolved. The restriction and specification are acceptable. The staff requested GE to provide existing flow test information. This was DFSER Confirmatory Item 4.4-1. GE has provided references in SSAR Section 4.4 to such information on current reactors relevant to the above restricted modes of operation. GE has also included this information in the SSAR and the staff finds it to be acceptable. Therefore, DFSER Confirmatory Item 4.4-1 is resolved.

4.5 Reactor Materials

4.5.1 Control Rod Drive System Structural Materials

The acceptance criteria used as the bases for the staff's evaluation of control rod drive (CRD) structural materials are SRP Section 4.5.1. The CRD structural materials are

acceptable if they meet the relevant requirements of: GDC 1 as it relates to structures, systems, and components important to safety being designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed; GDC 14 as it relates to the reactor pressure boundary being designed, fabricated, erected and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure and of gross rupture; GDC 26 as it relates to the control rod being capable of reliably controlling reactivity changes so that specified acceptance fuel design limits are not exceeded; and Section 50.55a, of Title 10 of the Code of Federal Regulations, Part 50 as it relates to structures, systems, and components shall be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety functions to be performed.

The SSAR states that the properties of the materials selected for the ABWR CRD mechanism will be equivalent to those given in Appendix I to Section III of the American Society of Mechanical Engineers (ASME) Code; Parts A, B, and C of Section II of the ASME Code; and RG 1.85. No cold-worked austenitic stainless steels except those with controlled hardness or strain are employed in the CRD system. All materials used in this system will be selected for their compatibility with the reactor coolant as described in Articles NB-2160 and NB-3120 of the ASME Code. The materials selected as identified in SSAR Section 4.5.1 will be resistant to stress corrosion in a BWR environment. The controls imposed on the austenitic stainless steel of the CRD mechanism conform to the recommendations of RGs 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal," (Revision 3) and 1.44, "Control of the Use of Sensitized Stainless Steel," (Revision 0).

All materials selected for application in CRD mechanism components will conform with the ASME Code Section III or RG 1.85. Fabrication and heat treatment practices performed in accordance with the Code and regulatory guide provide added assurance that stress corrosion cracking will not occur during the design life of the components. Both martensitic and precipitation-hardening stainless steels will be given tempering or aging treatments in accordance with staff positions. Cleaning and cleanliness control will be in accordance with American National Standards Institute Standard N 45.2.1-1973, "Cleaning of Fluid Systems and Associated Components During Construction Phase of Nuclear Power Plants," and RG 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants," Revision 0.

The information in the SSAR meets the criteria in SRP Section 4.5.1. The staff concludes that the structural

materials for the CRD mechanism conform to the staff's regulatory guidance to ensure that the requirements of GDC 1, 14, and 26 of Appendix A to 10 CFR Part 50 and the requirements of 10 CFR 50.55a are satisfied. This is acceptable.

4.5.2 Reactor Internal Materials

The acceptance criteria used as the bases for the staff's evaluation of reactor internal materials are SRP Section 4.5.2. The reactor internals are acceptable if the design, fabrication, and testing of the materials used in the reactor internal and core support structures meet the code and standards commensurate with the safety function to be performed so that the relevant requirements of GDC 1 and Section 50.55a of Title 10 of the Code of Federal Regulations, Part 50 are met.

The SSAR states that the requirements of GDC 1 and 10 CFR 50.55a will be met with regard to ensuring that the design, fabrication, and testing of the materials used in the reactor internal and core support structures are of high quality standards and adequate for structural integrity. The controls imposed on components constructed of austenitic stainless steel will satisfy the recommendations of RGs 1.31 and 1.44.

The materials to be used for the construction of components of the reactor internals and core support structures were identified in SSAR Section 4.5.2 by specification and found to be in conformance with the requirements of NG-2000 of Section III and Parts A, B, and C of Section II of the ASME Code. Extensive tests and satisfactory performance have shown that the specified materials are compatible with the BWR environment. The controls imposed on the reactor coolant chemistry satisfy the chemistry limits specified in EPRI report NP-4947, "BWR Hydrogen Water Chemistry Guidelines 1987 Revision," December 1988. This will provide reasonable assurance that the reactor internal and the core support structures will be adequately protected during operation from conditions that could lead to stress corrosion, including irradiation-assisted stress corrosion cracking and loss of component structural integrity.

The materials selection, fabrication practices, examination and testing procedures, and control practices provide reasonable assurance that the materials used for the reactor internal and core support structures will be maintained in a metallurgical condition that will preclude inservice deterioration. Conformance with the requirements of the ASME Code constitutes an acceptable basis for meeting, in part, the requirements of GDC 1 and 10 CFR 50.55a.

The staff finds that the information in the SSAR related to reactor internal materials meets the criteria of SRP Section 4.5.2 and is, therefore, acceptable.

4.6 Functional Design of Fine Motion Control Rod Drive System

The staff reviewed the fine motion control rod drive (FMCRD) system in accordance with SRP Section 4.6. The staff performed an audit review of each of the areas listed in the "Areas of Review" portion of the SRP section according to the guidelines provided in the "Review Procedures" section of the SRP section. Conformance with the acceptance criteria formed the basis for the staff's evaluation of the CRD system with respect to the applicable regulations of 10 CFR Part 50.

The ABWR incorporates electric-hydraulic FMCRDs, which provide electric fine rod motion during normal operation and hydraulic pressure for scram insertion. Fine motion during normal operation is provided by a ball nut and spindle arrangement driven by the electric stepper motor. In response to a scram signal, the control rods are inserted hydraulically via the stored energy in the scram accumulator similar to the current operating BWR CRDs.

A scram signal is also given simultaneously to insert the FMCRDs electrically via the FMCRD motor drive. This diversity, hydraulic and electric methods of scrambling provides a high degree of assurance of rod insertion on demand.

The FMCRD and recirculation flow control system (RFCS) are designed to control reactivity during power operation. Reactivity will be controlled in the event of fast transients by automatic rod insertion. During ATWS conditions, the internal recirculation pumps will be tripped automatically. In the event the reactor cannot be shut down with the control rods, the operator can actuate the standby liquid control system (if not automatically started) that pumps a solution of sodium pentaborate into the primary system. The evaluation of the functional design of the standby liquid control system is addressed in Section 9.3.5 of this report. This evaluation resolved Open Item 15 from the DSER (SECY-91-153). Compliance with the ATWS rule is discussed in Section 15.5 of this report.

Reactivity in the core will be controlled by the FMCRD by moving control rods interspersed throughout the core. These rods will control the reactor's overall power level and will provide the principal means of quickly and safely shutting down the reactor.

Reactor System

GE submitted a proprietary failure modes and effects analysis for the FMCRD system. The single-failure analysis of the FMCRD and hydraulic control unit (HCU) components indicates that the system design is satisfactory. This resolved Open Item 17 from the DSER (SECY-91-153). A supply pump (with a spare pump on standby) will provide the HCUs with water from the condensate treatment system and/or condensate storage tank to supply CRD purge water and to supply the purge water to the RIPs and reactor water cleanup pumps. The supply pump also will provide water to a scram accumulator in each HCU to maintain the desired water inventory. When necessary, the accumulator will force water into the drive system to scram the control rods connected to that HCU; the volume of water in the scram accumulator will be sufficient to scram two rods. A single failure in an HCU would result in the failure of two rods only. The failed rods would not be adjacent; they would be sufficiently separated so that the reactivity effect would essentially be the same as for the failure of one rod in current BWRs. Therefore, adequate shutdown margin exists with a single HCU failure even though the HCU is shared by two drives.

The FMCRD is designed to permit periodic functional testing during power operation with the capability to independently test individual scram channels and motion of individual control rods. The FMCRD is also designed so that failure of all electrical power or instrument air will cause the control rods to scram, thereby protecting the reactor. This satisfies the protection system failure mode requirements of GDC 23.

Preoperational tests of the CRD hydraulic system will be conducted to verify the capability of the system. Startup tests will be conducted over the range of temperatures and pressures from shutdown to operating conditions to determine compliance with applicable TS. Each rod that is partially or fully withdrawn during operation will be exercised one notch at least once each week. After each refueling shutdown, control rods will be tested for compliance with scram time criteria from the fully withdrawn position.

The FMCRD is designed to control reactivity under normal operating conditions and during anticipated operational occurrences. This capability is demonstrated by the safety analyses discussed in SSAR Chapter 15 (including effects of stuck rods). This CRD system also will be capable of holding the core subcritical under cold shutdown conditions. The RFCS will be capable of accommodating reactivity changes during normal operating conditions. The standby liquid control system will be capable of bringing the reactor subcritical under cold shutdown conditions in the event the control rods cannot be inserted.

These protection and reactivity control systems, taken together, satisfy the requirements of GDC 26, 27, and 29 pertaining to reactivity control system redundancy and capability, combined reactivity control system capability, and protections against anticipated operational occurrences.

The control rod system design incorporates appropriate limits on the potential amount and rate of reactivity increase. Control rod withdrawal sequences and patterns will be selected to achieve optimum core performance and low individual rod worths. The rod control and information system (RCIS) will reduce the chances of withdrawal other than by the preselected rod withdrawal pattern. The RCIS function will assist the operator with an effective backup control rod monitoring routine that enforces adherence to established control rod procedures for startup, shutdown, and low-power-level operations.

A malfunction in the FMCRD could result in a reactivity change. GE demonstrated in SSAR Chapter 15 that the FMCRD system will limit these postulated transients to within acceptable fuel design limits, as required by GDC 25.

The control rod mechanical design incorporates a brake system and ball check valve, which will reduce the chances of rapid rod ejection. This engineered safeguard will protect against a high reactivity insertion rate from a potential control rod ejection. Normal rod movement and the rod withdrawal rate will be limited through the FMCRD.

GE adopted an internal CRD housing support to replace the support structure of beams, hanger rods, grids, and support bars used in current BWR designs. This system will use the outer tube of the drive to provide support. This tube will be welded to the drive middle flange and will attach by a bayonet lock to the guide tube base. The guide tube, supported by the housing extension, will prevent downward movement of the drive in the event of housing failure. The CRD housing support is designed to prevent ejection of a CRD and attached control rod.

The FMCRD is designed to detect separation of the control rod from the drive mechanism. Two redundant and separate Class 1E switches will be provided to detect the separation of either the control rod from the hollow piston or the hollow piston from the ball nut. Actuation of either of these switches will cause an immediate rod block and will initiate an alarm in the control room, thereby reducing the chances of a rod drop accident. The ABWR control rod design does not include a velocity limiter because of the design features described. The design features of the reactivity control system will limit the potential amount

and rate of reactivity increase to ensure that GDC 28 is satisfied for postulated reactivity accidents.

The safety concerns associated with a pipe break, given in NUREG-0803, "Safety Concerns Associated With a Pipe Break in the BWR Scram System," are not applicable for the ABWR. The ABWR design does not include scram discharge volume piping. The water displaced by the CRD during the scram will be routed to the reactor pressure vessel.

Nitrogen charge pressure for the CRD accumulator is substantially increased in the ABWR design. The initial N₂ charge pressure will be 1,4817 kPa (2,134 psig) compared with a maximum of 10,446 kPa (1,500 psig) for the operating BWRs. GE's calculations have verified that even with initial N₂ charge pressure of 12,859 kPa (1,850 psig), there is sufficient differential pressure to drive the CRD fully with a reactor pressure of 9,584 kPa (1,375 psig). (The peak calculated reactor pressure for a main steam isolation valve closure event during ATWS is less than 9,584 kPa (1,375 psig)). The calculations have shown that at 4.53 seconds, when the CRD is fully inserted, the accumulator N₂ charge pressure is calculated as 10,625 kPa (1,526 psig) with reactor pressure at 9,584 kPa (1,375 psig). There is a remaining differential pressure of 166 kPa (151 psid), demonstrating that accumulator pressure is more than adequate for the rod insertion.

GE submitted the LaSalle test report on FMCRD in-plant test program by letter dated October 12, 1989. This report is proprietary. The in-plant test shows that the basic design of the FMCRD is acceptable. There is significant operating experience with the FMCRDs. In Europe, 2,700 drives are in service with over 15,000 drive years of experience. Thus, FMCRDs in BWRs, are a proven technology. Open Item 16 in the DSER (SECY-91-153) was resolved in the DFSER.

GE submitted the design description and the ITAAC for the FMCRD system. This was DFSER Open Item 4.6-1. GE has provided a revised set of design descriptions and ITAAC. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Section 14.3 of this report. On the basis of this evaluation, this open item is resolved.

The staff concludes that the functional design of the reactivity control system conforms to the requirements of GDC 23, 25, 26, 27, 28, and 29 with regard to demonstrating the ability to reliably control reactivity changes under normal operation, anticipated operational occurrences, and accident conditions, including single failures. The design of the reactivity control system conforms to the applicable acceptance criteria of SRP Section 4.6, and is acceptable.

5 REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

5.1 Introduction

The reactor coolant systems (RCS) and connected subsystems are evaluated in the following sections.

5.2 Integrity of Reactor Coolant Pressure Boundary

5.2.1 Compliance With Code and Code Cases

The staff reviewed the measures used to provide and maintain the integrity of the reactor coolant pressure boundary (RCPB) and other pressure-retaining components and their supports that are important to safety for the design lifetime of the plant.

5.2.1.1 Compliance With 10 CFR 50.55a

According to 10 CFR 50.55a, components important to safety are subject to the following:

- (1) RCPB components must meet the requirements for Class 1 (Quality Group (QG) A) components specified in ASME Code, Section III, except for those components that meet the exclusion requirements of 10 CFR 50.55a(c)(2).
- (2) Components classified as QG B and C must meet the requirements for Class 2 and 3 components, respectively, specified in ASME Code, Section III.

SSAR Table 3.2-1 classifies the pressure-retaining components of the RCPB as ASME Code, Section III, Class 1 components. These Class 1 components are designated QG A in conformance with Regulatory Guide (RG) 1.26, "Quality Groups Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," Revision 3. The staff reviewed the QG A RCPB components in accordance with SRP Section 5.2.1.1, as discussed below.

In addition to the QG A components of the RCPB, certain lines that will perform a safety function and that meet the exclusion requirements of 10 CFR 50.55a(c)(2) are classified as QG B in accordance with Position C.1 of RG 1.26, Revision 3, and will be constructed as ASME Code, Section III, Class 2 components. The staff's review of these components and other pressure-retaining components that will be constructed to ASME Code, Section III, Class 2 and Class 3 specifications, is discussed in Section 3.2.2 of this report.

SRP Section 5.2.1.1 recommends that safety analysis reports for both construction permits and operating licenses

contain a table identifying the ASME component code, code edition, and applicable addenda for all ASME Code, Section III, Class 1 and 2 pressure vessel components, piping, pumps, and valves in the RCPB. SSAR Section 5.2.1.1 states that the ASME Code edition, applicable addenda, and component dates will be in accordance with 10 CFR 50.55a and that ASME Code, Section III, will be used for the design of ASME Code Class 1, 2, and 3 pressure retaining components and their supports. The specific edition and addenda are given in SSAR Tables 1.8-21 and 3.2-3. The ASME Code is considered Tier 1 information; however, the specific edition and addenda are considered Tier 2 information partly because of the continually evolving design and construction practices (including inspection and examination techniques) of the Code. Fixing a specific edition and addenda during the design certification stage might result in inconsistencies between design and construction practices during the detailed design and construction stages. The ASME Code involves a consensus process to reflect the evolving design and construction practices of the industry. Although reference to a specific edition of the Code for the design of ASME Code Class components and their supports is suitable to reach a safety finding during the design certification stage (as reflected in the discussion of the reactor pressure vessel (RPV) design in Section 5.2.4 of this report), it is necessary that the construction practices and examination methods of an updated Code that would be effective at the COL stage be consistent with the design practices established at the design certification stage.

To avoid this potential inconsistency for the advanced boiling water reactor (ABWR) pressure-retaining components and their supports, it is appropriate that the ASME Code be specified as Tier 1 information and the specific edition and addenda as Tier 2 information so that the COL applicant has the option to revise or supplement the referenced Code edition with portions of the later Code editions and addenda and still ensure consistency between the design and construction practices. In this manner, consistency with the latest design, construction, and examination practices also is ensured. However, the staff finds that there might be a need to fix certain design parameters from a specific Code edition or addenda during its design certification review particularly when that information is important for establishing a significant aspect of the design or is used by the staff to reach its final safety determination. Such considerations, if necessary, are reflected in the various sections of this report.

Therefore, all ASME Code, Class 1, 2, and 3 pressure-retaining components and their supports shall be designed in accordance with the requirements of ASME Code, Section III, using the specific edition and addenda

Reactor Coolant System and Connected Systems

given in the SSAR. The COL applicant should ensure that the design is consistent with the construction practices (including inspection and examination methods) of the ASME Code edition and addenda in effect at the time of the COL application, as endorsed in 10 CFR 50.55a. The COL applicant should identify in its application the portions of the later code editions and addenda for NRC staff review and approval. This was DFSE COL Action

Item 14.1.3.3.2.1-1. The resolution of this item is discussed in Section 3.12.2.1 of this report.

Since the above position is not totally consistent with ASME Code, Section III, Subsection NCA-1140, "Use of Code Editions, Addenda, and Cases," the following comparison is provided:

NCA-1140 Rules

NCA-1140(a)(1) - Under the rules of this Section, the owner or his designee shall establish the Code edition and addenda to be included in the Design Specifications. All items of a nuclear power plant may be constructed to a single Code edition and addenda, or each item may be constructed to individually specified Code editions and addenda.

NCA-1140(a)(2) - In no case shall the Code edition and addenda dates established in the Design Specifications be earlier than 3 years prior to the date that the nuclear power plant construction permit application is docketed.

NCA-1140(b) - Code editions and addenda later than those established by (a) above may be used by mutual consent of the owner or his designee and certificate holder. For Division 2 design and construction, the consent of the designer shall also be obtained. Specific provisions within an edition or addenda later than those established in the design specifications may be used, provided that all related requirements are met.

NCA-1140(c) - Code Cases are permissible and may be used beginning with the date of approval by the ASME Council (and the American Concrete Institute for Division 2 design and construction). Only Code Cases that are specifically identified as being applicable to this Section may be used.

Design Certification Position

(1) The vendor must specify the Code edition and addenda in the SSAR during design certification. The COL applicant may update the Code edition or addenda (or portions thereof) referenced in the SSAR using a process similar to that specified in 10 CFR 50.59 without prior NRC approval unless the proposed change involves a change to the certified design or an unreviewed safety question. The COL applicant may update the referenced edition or addenda for all items of a nuclear power plant or a specific item if the construction practices (including fabrication, inspection, and examination methods) are not compatible with the design requirements. All changes to the referenced edition and addenda must be documented and maintained by the COL applicant and must be available for audit.

(2) The specific Code edition and addenda are required to be established during design certification except for Section XI requirements related to inservice inspection (ISI), inservice testing, and system pressure tests, which must meet the requirements of 10 CFR 50.55a(f) and (g). Further discussions and any exceptions to 10 CFR 50.55a(f) and (g) are noted in Sections 5.2.4 and 6.6 of this report.

(3) This is acceptable subject to the conditions noted in Position (1) above.

(4) As discussed in Section 5.2.1.2 of this report, only those Code Cases identified in RG 1.84, Revision 24, or 1.85, Revision 24, as specified in the SSAR may be used. The COL applicant may submit for staff review and approval future Code Cases that are endorsed in future revisions of RGs 1.84 and 1.85 with its COL application, provided these cases do not involve a change to the certified design or an unreviewed safety question.

Reactor Coolant System and Connected Systems

NCA-1140 Rules

NCA-1140(d) - Code Cases may be used by mutual consent of the owner or his designee, and the certificate holder on or after the date permitted by (c) above. For Division 2 design and construction, the consent of the designer shall also be obtained.

NCA-1140(e) - Existing materials previously produced and certified in accordance with Code editions and addenda earlier than the one specified for construction of an item may be used, provided all of the following requirements are satisfied.

- (i) The material (NCA-1220) meets the applicable requirements of a material specification permitted by paragraph 2121 of the applicable subsection of the Section III edition and addenda specified for construction.
- (ii) The material meets all the requirements of Article 2000 of the applicable Subsection of the Section III edition and addenda specified for construction.
- (iii) The material was produced under the provisions of a quality system program that had been accepted by the society or qualified by a party other than the society (NCA-3820), in accordance with the requirements of the latest Section III edition and addenda issued at the time the material was produced. Material exempted from portions of the provisions of NCA-3800 by paragraph 2610 of the applicable Subsection of Section III may be used, provided the requirements of (i) and (ii) above are met.

NCA-1140(f) - Code editions, addenda (including the use of specific provisions of editions addenda permitted by (b) and (e) above), and Cases used shall be reviewed by the owner or his designee for acceptability to the regulatory and enforcement authorities having jurisdiction at the nuclear power plant site.

Design Certification Position

(5) Code Cases to be used in the design of the standard plant must be identified by the vendor in the SSAR during design certification.

(6) Does not apply to design certification.

(7) The vendor must specify the Code edition and addenda in the SSAR during design certification. Use of later editions and addenda (or portions thereof) and Code Cases not approved during design certification must be reviewed by the COL applicant or its designee for acceptability to the regulatory and enforcement authorities having jurisdiction at the nuclear power plant site.

Reactor Coolant System and Connected Systems

5.2.1.1.1 Conclusion

The staff concludes that the construction of all ASME Code, Class 1, 2, and 3 components and their supports will conform to the appropriate ASME Code editions and addenda and the Commission's regulations and that component quality will be commensurate with the importance of the safety function of all such components and their supports. This constitutes an acceptable basis for satisfying GDC 1 and is acceptable.

5.2.1.2 Applicable Code Cases

SSAR Table 5.2-1 identifies specific ASME Code Cases that will be applied in the construction of pressure-retaining ASME Code, Section III, Class 1, 2, and 3 components. The staff's review of this table is based on the guidelines in RG 1.84, "Design and Fabrication Code Case Acceptability - ASME Section III, Division 1," and RG 1.85, "Materials Code Case Acceptability - ASME Section III, Division 1." All ASME Code Cases that have been either conditionally or unconditionally endorsed by the staff are discussed in one of these RGs, as applicable. Table 5.2-1 of the SSAR lists 15 code cases that will be used in the design of the ABWR. All of these code cases have been endorsed by the staff and are included in one of the above RGs.

In SSAR Sections 3.9.3.4 and 3.9.3.5, Code Case N-476, "Class 1, 2, 3, and MC Linear Component Supports - Design Criteria for Single Angle Members, Section III, Division 1, Subsection NF," is referenced as augmenting ASME Subsection NF rules for the design of component supports. As stated in Section 3.9.3.3 of this report, the staff finds this code case acceptable. Therefore, it will be an acceptable addition to SSAR Table 5.2-1.

The only acceptable ASME Code Cases that may be used for the design of ASME Code, Class 1, 2, and 3 piping systems in the ABWR standard plant are those either conditionally or unconditionally approved in RGs 1.84 and 1.85 in effect at the time of design certification. However, the COL applicant may submit, with its COL application, future code cases that are endorsed in RGs 1.84 and 1.85 at the time of the application provided they do not alter the staff's safety findings on the ABWR certified design. In addition, the COL applicant should submit those Code Cases that are applicable to RG 1.147, "Inservice Inspection Code Case Acceptability - ASME Section XI, Division 1," which is in effect at the time of the COL application.

The staff concludes that all of the Code Cases in SSAR Table 5.2-1 either meet the guidelines of RG 1.84 or 1.85

or have been reviewed and endorsed by the staff and are acceptable for use on the ABWR design. Compliance with the requirements of these Code Cases will result in a component quality that is commensurate with the importance of the safety functions of these components, constitutes the basis for satisfying GDC 1, and is acceptable.

5.2.2 Overpressure Protection

The staff evaluated overpressure protection in the ABWR in accordance with SRP Section 5.2.2, which states that the acceptance criteria also are based on GDC 31 as it relates to the fracture behavior of the RCPB. This review area is addressed in Section 5.3.1 of this report. SRP Section 5.2.2 also states that overpressure protection during low-temperature operation need not be considered for BWRs, since BWRs never operate in water solid conditions. Hence, overpressure protection during low-temperature conditions is not addressed for the ABWR.

The RCPB is designed with a pressure relief system to:

- prevent the pressure in the RCPB from rising beyond 110 percent of the design value
- provide automatic depressurization if small breaks in the nuclear system should occur together with failure of the high-pressure core flooders (HPCF) and reactor core isolation cooling (RCIC) system. (This depressurization will allow operation of the low-pressure flooders systems to protect the fuel barrier.)

To be acceptable, the pressure relief system must permit verification of its operability and must withstand adverse combinations of loadings and forces resulting from normal, upset, emergency, and faulted conditions.

Overpressure protection in the ABWR will be provided using 18 safety/relief valves (SRVs); of which 8 are part of the automatic depressurization system (ADS). The 18 SRVs are divided into six nominal pressure set point groups and mounted on the four main steamlines (MSLs) between the reactor vessel and the first isolation valve inside the drywell. The SRVs will discharge through piping to the suppression pool. The design of the ABWR pressure relief system is similar to that for BWR 4, 5, and 6 plants.

The SRVs are classified as QG A and seismic Category I, as shown in SSAR Table 3.2. The SRVs are designed to meet RGs 1.26, Revision 3, and 1.29, "Seismic Design Classification," Revision 3.

The nominal pressure set points of the SRVs will be distributed in six valve groups with a minimum set point of 7.9 MPa (1,149.2 psig) and a maximum of 8.2 MPa (1,189.0 psig) in the safety mode of operation. The nominal pressure set points of the SRVs in the relief mode of operation will be at a minimum of 7.5 MPa (1,089.5 psig) and maximum of 7.9 MPa (1,139.2 psig). The SRVs can also be operated in the relief mode by remote-manual controls from the main control room. Four SRVs can also be operated from the remote shutdown panel. The effects of flow-induced SRV discharge line backpressure on the performance of the SRV are addressed by sizing the line to ensure that the steady-state backpressure does not exceed 40 percent of the SRV inlet pressure. These sizing criteria control the effective backpressure buildup and maintain the required force balance needed to keep the SRV open and to permit proper blowdown.

In the DFSER, the staff noted that before the valves are installed, the SRV manufacturer will test them hydrostatically according to ASME Code, Section III requirements. During startup testing, opening response time and set pressure tests will be conducted to verify that design and performance requirements have been met. This was DFSER COL Action Item 5.2.2-1. Since these tests are being addressed in Chapter 14 of the SSAR, they need not be specified as a COL action item. Therefore, COL Action Item 5.2.2-1 is resolved.

GDC 15 defines the basis for overpressurization protection in a nuclear reactor. It requires that the RCPB design conditions not be exceeded during any condition of normal operation, including anticipated operational occurrences. To satisfy this criterion, the overpressurization protection system for the ABWR is designed in compliance with ASME Code, Section III, which requires that the maximum pressure reached during the most severe pressure transient be less than 110 percent of the design pressure. For the ABWR, that pressure limit is 9.5 MPa (1,375.3 psig). GE analyzed the series of transients that would be expected to require SRV actuation to prevent overpressurization. The analysis was performed using the computer-simulation model ODYNA. ODYNA is the ABWR version of ODYN incorporating changes unique to the ABWR such as reactor internal pumps (RIPs). ODYN is described in GE Topical Report NEDO-24154. The staff reviewed ODYN and found it acceptable as documented in "Safety Evaluation for the General Electric Topical Report Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors, NEDO-24154 and NEDO-24154-P Volumes I, II, III, June 1980." The staff performed an audit of ODYNA and found the changes to be acceptable. The acceptability of ODYNA for ABWR transient analysis is discussed in Section 15.1 of this

report. The analyses show that the maximum pressure will remain below the 9.5-MPa (1,375.3-psig) limit. For the most severe transient (i.e., closure of all main steam isolation valves (MSIVs) with a high neutron flux scram), the maximum vessel bottom pressure is calculated to be 8.8 MPa (1274.4 psig) when all 18 SRVs are assumed to operate in the safety mode. The analysis assumed that the plant was operating at 102.7 percent of rated steam flow of 7.85×10^6 kg/hr (about 17×10^6 lb/hr) and a vessel dome pressure of 7.2 MPa (1,040 psig). This is consistent with SRP Section 5.2.2 and is acceptable.

GE based the sizing of the SRVs on the initiation of a reactor scram by the high-neutron flux scram, which is the second safety-grade scram signal from the reactor protection system following MSIV closure. The staff believes that the qualification and redundancy of reactor protection system equipment, coupled with the fact that the reactor vessel pressure is limited to less than 110 percent of design pressure, provides adequate assurance that the reactor vessel integrity will be maintained for the limiting transient event.

The staff evaluation of TMI-2 Action Items (NUREG-0737 requirements that are incorporated into 10 CFR 50.34(f)(1)(vi), (1)(v), (2)(vi), 2(x), and (2)(xi)), as related to SRVs, is discussed in Chapter 20 of this report.

GE performed the overpressure protection analysis for a core loading pattern, which is described in Chapter 4 of the SSAR.

GE originally submitted the design description and the inspection, test, analysis, and acceptance criteria (ITAAC) for SRVs. At the time the DFSER was issued, the ITAAC review was in progress. Therefore, this was DFSER Open Item 5.2.2-1. GE has since submitted revised design description and ITAAC in its certified design material (CDM). The adequacy and acceptability of the CDM are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

The staff concludes that the pressure relief system, in conjunction with the reactor protection system, will provide adequate protection against overpressurization of the RCPB and that the overpressurization system is acceptable and meets the relevant requirements of GDC 15.

5.2.3 Reactor Coolant Pressure Boundary Materials

The staff reviewed RPCB materials in accordance with SRP Section 5.2.3. The RPCB materials are acceptable if they meet the requirements of (1) GDC 1 and 30 as related

Reactor Coolant System and Connected Systems

to quality standards for design, fabrication, erection and testing; (2) GDC 4 as related to compatibility of components with environmental conditions; (3) GDC 14 and 31 as related to extremely low probability of rapidly propagating fracture and gross rupture of the RCPB; (4) Appendix B to 10 CFR Part 50 as related to onsite material cleaning control; (5) Appendix G to 10 CFR Part 50 as related to material testing and acceptance criteria for fracture toughness of the RCPB; and (6) 10 CFR 50.55a as related to quality standards and fracture toughness.

In the DSER (SECY-91-153), the staff noted that the materials used for the construction of RCPB components had been identified by specification and were in conformance with Section III of the ASME Code and NUREG-0313. However, the staff requested that GE use Revision 2 (not Revision 1 as proposed by GE) of NUREG-0313. GE revised the SSAR (Amendment 14) to reference NUREG-0313, Revision 2. GE's compliance with the provisions of the Code for material specifications and conformance with NUREG-0313, Revision 2, satisfy the quality standards of GDC 1 and 30 and 10 CFR 50.55a.

In the DFSE, the staff noted that the construction materials for the RCPB had been identified in SSAR Table 5.2-4 and were compatible with the primary coolant water, which will be chemically controlled in accordance with appropriate technical specifications (TS). This was DFSE TS Item 5.2.3-1.

GE addressed DFSE TS Item 5.2.3-1 in its April 16, 1993, submittal, "Response to TS Items in ABWR DRAFT FSER," which stated that the reactor coolant chemistry limits had been removed from the improved TS (NUREG-1433 and -1434) and thus would not be included in the ABWR TS. The reactor coolant chemistry limits are to be controlled by appropriate administrative controls outside of TS. This is acceptable. Therefore, this item is resolved.

The RCP materials of construction identified in the SSAR that will be exposed to the reactor coolant have been identified and are compatible with the primary coolant water, which will be chemically controlled to maintain adequate water purity. This compatibility has been proven by extensive testing and satisfactory performance. This includes conformance with the recommendations of RG 1.44, "Control of Sensitized Stainless Steel," Revision 0, and with the staff guidelines of NUREG-0313, Revision 2. General corrosion of all materials, except unclad carbon and low-alloy steels will be negligible. For unclad carbon and low-alloy steel, GE has provided conservative corrosion allowances for all exposed surfaces in

accordance with ASME Code, Section III. This compatibility with the reactor coolant and compliance with the ASME Code satisfy the requirements of GDC 4 as related to the compatibility of components environmental conditions.

The main source of radiation buildup in operating plants is cobalt-60, which is formed by neutron activation of cobalt-59. GE reduced the cobalt content in alloys to be used in high-fluence areas such as fuel assemblies and control rods. It replaced cobalt-base alloys, used for pins and rollers in control rods, with non-cobalt alloys. This will reduce occupational exposure from cobalt-60 during operation and maintenance of plant components.

The materials to be used for the construction of the RCPB are compatible with the thermal insulation used in these areas and conform to the recommendations of RG 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steels," Revision 0. They satisfy the requirements of GDC 14 and 31 as they relate to the prevention of RCPB failure.

The ferritic steel tubular products and the tubular products fabricated from austenitic stainless steel that GE proposes to use will be examined nondestructively in accordance with the provisions of ASME Code, Section III. This is acceptable and satisfies the quality standards of GDC 1 and 30 and 10 CFR 50.55a.

The fracture toughness tests, required by the ASME Code and augmented by Appendix G to 10 CFR Part 50, provide reasonable assurance that adequate safety margins against nonductile behavior or rapidly propagating fracture can be established for all pressure-retaining components of the RCPB. The use of Appendix G to ASME Code, Section III, and the results of fracture toughness tests performed in accordance with the ASME Code and NRC regulations in establishing safe operating procedures provide adequate safety margins during operations, testing, maintenance, and postulated accident conditions. This satisfies the requirements of GDC 14 and 31 and 10 CFR 50.55a regarding the prevention of RCPB fracture and of gross rupture. The use of low-alloy steel is restricted to the RPV. The controls imposed on preheat temperatures for welding of ferritic steels conform to the requirements of ASME Code, Section III, and provide reasonable assurance that components made from ferritic steels will not crack during fabrication. These controls also minimize the possibility of subsequent cracking due to the retention of residual stresses in the weldment and satisfy the quality standards requirements of GDC 1 and 30 and 10 CFR 50.55a.

The controls imposed on electroslag welding of ferritic steels are not necessary because electroslag welding will not be used for RCPB components.

The controls imposed on the welding of RCPB materials under conditions of limited accessibility are in accordance with the recommendations of RG 1.71, "Welder Qualification for Areas of Limited Accessibility," Revision 0, and provide assurance that proper requalification of welders will be required in accordance with the welding conditions. These controls also satisfy the quality standards requirements of GDC 1 and 30 and 10 CFR 50.55a. The controls imposed on stainless steel weld cladding are not necessary because the use of low-alloy steels is restricted to the RPV.

The controls to avoid stress corrosion cracking in RCPB components constructed of austenitic stainless steels conform to the recommendations of RGs 1.44 and 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Plants," and NUREG-0313, Revision 2. Cold work of austenitic stainless steel is controlled by applying limits on hardness, bend radii, and surface finish on ground surfaces. The controls to be followed during material selection, fabrication, examination, and protection, in accordance with these recommendations, in order to prevent excessive yield strength, sensitization, and contamination provide reasonable assurance that the RCPB components of austenitic stainless steels will be in a metallurgical condition that minimizes susceptibility to stress corrosion cracking during service. These controls meet the requirements of GDC 4 pertaining to the compatibility of components with environmental conditions and those of GDC 14 pertaining to the prevention of leakage and failure of the RCPB.

Since hydrogen water chemistry will be used, a hydrogen injection of less than 1 part per million (ppm) in the feedwater will also be used to minimize intergranular stress corrosion cracking (IGSCC) of the reactor internals in the ABWR. To suppress IGSCC, the reactor coolant conductivity will be maintained below 0.3 microsiemen/cm and sufficient hydrogen will be added to the feedwater to reduce the electrochemical potential below -0.23 volt (standard hydrogen electrode). These controls will further ensure that the materials exposed to the reactor coolant will not be subject to IGSCC. This is acceptable.

The staff concludes that the RCPB materials are acceptable because of the above reasons and because they meet the requirements of GDC 1, 4, 14, 30, and 31 of Appendix A to 10 CFR Part 50; of Appendices B and G to 10 CFR Part 50; and of 10 CFR 50.55a.

5.2.4 Inservice Inspection and Testing of the Reactor Coolant Pressure Boundary

SSAR Section 5.2.4 and Table 5.2-8 describe certain commitments and plans for the preservice inspection (PSI) and inservice inspection (ISI) programs. GE discussed basic inspection concepts and general ASME Code provisions because it had reasoned that the requirements of the NRC regulations might be controlled by the date of order of each specific component subject to examination. The staff review was performed in accordance with SRP Section 5.2.4, except as discussed below.

Throughout the service life of the plant, the COL applicant will have the overall responsibility for the ISI of the RCPB although other organizations also will contribute to the examination activity (e.g., the reactor vendor, the architect-engineer and the inspection agency). The staff and GE representatives discussed the issue of ISI requirements during meetings on March 25 and 26, 1992, as documented in an NRC meeting summary dated April 28, 1992. It was determined that GE is responsible for designing the RPV for accessibility to perform PSI and ISI. For all ASME Code, Class 1, 2, and 3 components, the development of the PSI and ISI programs is the responsibility of the COL applicant.

Pursuant to 10 CFR 50.55a(g)(3), "Components which are classified as ASME Code Class 1 shall be designed and be provided with access to enable the performance of inservice examination of such components and shall meet the preservice examination requirements set forth in Section XI of editions of the ASME Boiler and Pressure Vessel Code and Addenda applied to the construction of the particular component." The applicable construction code should be determined by Paragraph NCA-1140 of ASME Code, Section III, but the construction code that is selected must be incorporated by reference in 10 CFR 50.55a(b).

Throughout the service life of a nuclear power facility, components (including supports) that are classified as ASME Code Class 1 must meet the requirements in ASME Code, Section XI, that become effective after the editions specified in 10 CFR 50.55a(g)(2) and are incorporated by reference in 10 CFR 50.55a(b), to the extent practical within the limitations of design, geometry, and materials of construction of the components. The inservice examination of components conducted during the initial 10-year interval must conform to the requirements of the latest edition and addenda of ASME Code, Section XI, incorporated by reference in 10 CFR 50.55a(b) on the date 12 months before the date of issuance of the operating license. The regulations also require that the ISI program be updated for each subsequent 10-year interval to comply

Reactor Coolant System and Connected Systems

with the ASME Code incorporated by reference in 10 CFR 50.55a.

Compliance with the requirements of the regulations for the PSI and ISI must be based on commitments for or by the COL applicant. The applicable ASME Code or Codes for both design and ISIs are not known at the time of design certification. In addition, the regulation permits the COL applicant the option to change to a newer edition of ASME Code, Section XI, by component, during the construction of the plant.

The staff review of the information in the SSAR emphasized design access and the use of an effective nondestructive examination (NDE) methodology. Since the PSI requirements are established and known at the time each component is ordered, 10 CFR 50.55a(g) does not have provisions for "relief requests" for impractical examination requirements. ASME Code, Section XI has provisions to use certain shop and field examinations in lieu of the onsite preservice examination. Therefore, the NRC staff concluded in the DFSER that the COL applicant must incorporate plans for NDE during construction in order to meet all access requirements of the regulations. This was DFSER COL Action Item 5.2.4-1.

GE responded to this COL item by including additional information in SSAR Section 5.2.6.2, "Plant-Specific ISI/PSI." This section states that the COL applicant will submit the complete plant-specific ISI/PSI program to the NRC including references to the edition and addenda of ASME Code, Section XI, that will be used for selecting components subject to examination, a description of the components exempt from examination by the applicable code, and isometric drawings used for the examination. This is acceptable. Further, SSAR Section 5.2.4.2 states that all items within the Class 1 boundary are designed to provide access for the examinations required by ASME Code, Section XI, IWB-2500. This is also acceptable.

ASME Code, Section XI, states that the PSI should be conducted with equipment and techniques equivalent to those that are expected to be used for subsequent ISIs. Ultrasonic testing of RCPB components will improve in the near future, as indicated by ASME Code, Section XI, Appendix VII, "Qualification of Nondestructive Examination Personnel for Ultrasonic Examination," and Appendix VIII, "Performance Demonstration for Ultrasonic Examination Systems." The NRC has referenced in 10 CFR 50.55a(b) the ASME Code, Section XI edition that includes the published Appendix VII. In addition, the NRC staff has established a technical contact to coordinate the implementation of Appendix VIII. Therefore, the staff concluded in the

DFSER that GE should include in its SSAR PSI program provisions that ultrasonic testing be performed in accordance with Appendices VII and VIII pursuant to 10 CFR 50.55a(g)(3). This was DFSER Open Item 5.2.4-1. However, on further review the staff determined that this open item could be resolved by the action taken by the COL applicant.

GE responded to this open item by including additional information in SSAR Section 5.2.6.2. This section states that the COL applicant will submit, for staff review, the complete plant-specific ISI/PSI program, including references to the edition and addenda of ASME Code, Section XI, that will be used for selecting components subject to examination, a description of the components exempt from examination by the applicable code, and isometric drawings used for the examination. This is acceptable. Therefore, DFSER Open Item 5.2.4-1 is resolved.

The requirements for the initial ISI program will be determined by the ASME Code in effect 1 year before issuance of the COL. The COL applicant is required to meet the ISI requirements "to the extent practical within the limitations of design, geometry and materials of construction of the components." The regulations have provisions for staff evaluation, on written request, of new ASME Code requirements determined by a licensee to be impractical for its facility.

In the DSER (SECY-91-355), the staff found that the SSAR information pertaining to compliance with 10 CFR 50.55a(g), design access, PSI requirements, and proposed methodology for ISI was unacceptable. This was DSER Outstanding Issue 1. In SSAR Section 5.2.4, GE addressed the topics discussed in SRP Section 5.2.4. GE described the access provisions for examining the major components of the RCPB including the reactor vessel, closure head, RPV studs, nuts and washers, reactor vessel support skirt, piping, pumps, valves, and component supports. GE stated that all items within the Class 1 boundary are designed to provide access for the examinations required by ASME Code, Section XI, IWB-2500. This is acceptable and resolved DSER Outstanding Issue 1.

SSAR Table 5.2-1 lists the applicable ASME Code Cases for the major RCPB components. In a letter dated March 11, 1992, GE stated that the Code addenda requirements for the ABWR plants will comply with 10 CFR 50.55a, except for the RPV, because as of that date the 1989 Edition of the ASME Code had not been referenced by the regulation. The 1989 Edition was subsequently referenced by 10 CFR 50.55a. As a result of a meeting on March 25 and 26, 1992, it was determined that the COL applicant

would be responsible for the development of the PSI and ISI programs for all ASME Code, Class 1, 2, and 3 components. Since the applicable ASME Code edition for the PSI and ISI programs at the time of the COL application cannot be determined at the time of design certification, pursuant to 10 CFR 50.55a(g), the development of the PSI and ISI programs for ASME Code, Class 1, 2, and 3 components for the ABWR was DFSEER COL Action Item 5.2.4-2. In the DSER (SECY-91-153), the staff noted that its reviews of the PSI and ISI programs were in progress. These were DSER Outstanding Issues 20 and 23. In response to the above items, GE included additional information in the SSAR to state that the COL applicant will submit the complete plant-specific ISI/PSI program to address a number of concerns including those discussed above. This is acceptable. The staff concludes that the SSAR enables the COL applicant to meet the requirements of the NRC regulations defined in 10 CFR 50.55a, except for the RPV, which is evaluated below as an individual component pursuant to 10 CFR Part 52. Therefore, DSER Outstanding Issues 20 and 23 are resolved.

GE committed to design the RPV so that PSI based on the requirements of ASME Code, Section XI, 1989 Edition, can be performed. This edition is specified in the RPV ITAAC, Section 2.1.1, "Reactor Pressure Vessel System." The RPV shell welds are designed for 100 percent accessibility for both PSI and ISI. The RPV nozzle-to-shell welds will be 100 percent accessible for PSI but might have limited areas that will not be accessible from the outer surface for inservice volumetric examination using current examination techniques. The staff will review the ISI program in accordance with the ASME Code edition in effect and the ISI techniques available at the time of the COL application.

The staff reviewed GE's use of the 1989 Edition of ASME Code, Section XI, and evaluated the extent of examination, design access, methodology for NDE, and personnel qualifications. Subarticle IWB-2200 "Preservice Examination" of the 1989 Edition states that examinations required for Class 1 components shall be completed before initial plant startup. In addition, these PSIs should be extended to include essentially 100 percent of the pressure-retaining welds in all Class 1 components except in those "components exempt from examination" as defined by IWB-1220(a), (b), or (c). GE's use of the 1989 Edition of ASME Code for design of the RPV should not significantly change the extent of examination required by 10 CFR 50.55a(g).

GE committed to design the RPV with essentially 100 percent of the shell welds accessible for both PSI and ISI. The RPV nozzle-to-shell welds, a different examin-

ation category defined by ASME Code, Section XI, will be 100 percent accessible for PSI. The staff assumes that PSI of the nozzle-to-shell regions will be accomplished by shop and field inspections during construction as permitted by the Code. Although the RPV nozzle-to-shell welds might have limited areas that will not be accessible from the outer surface for inservice volumetric examination using current examination techniques, the actual Code edition for ISI, as defined by 10 CFR 50.55a(g), cannot be determined until the construction of the ABWR plant is essentially complete. Therefore, the staff will review all examination limitations with the COL application. This is acceptable because the concepts described above are equivalent or superior to examinations performed on existing operating BWRs. However, the staff concluded in the DFSEER that the COL applicant must monitor the construction of structural supports around the RPV and the installation of auxiliary equipment to ensure that the design access that will be provided is not adversely affected. This was DFSEER COL Action Item 5.2.4-3.

GE responded to this item in SSAR Section 5.2.6.2, which states that the COL applicant will submit the complete plant-specific ISI/PSI program to the NRC including references to the edition and addenda of ASME Code, Section XI, that will be used for the selection of components subject to examination, a description of the components exempt from examination by the applicable code, and isometric drawings used for the examination. This is acceptable. GE also stated that the COL applicant will submit plans for preservice examination of the RPV welds to address the degree of compliance with RG 1.150, "Ultrasonic Testing of Reactor Vessel Welds During Preservice and Inservice Examinations," Revision 1. Further, SSAR Section 5.2.4.2 states that all items within the Class 1 boundary are designed to provide access for the examinations required by ASME Code, Section XI, IWB-2500. This is also acceptable.

The SSAR Section 5.2.4.3.2.1 states that ultrasonic testing of the reactor vessel welds will be performed in accordance with RG 1.150 and that personnel performing ultrasonic examinations shall be qualified in accordance with ASME Code, Section XI, Appendix VII. Ultrasonic examination systems will be qualified in accordance with an industry-accepted program for implementing ASME Code, Section XI, Appendix VIII. GE does not consider the supplemental examinations recommended in GE service information letters (SILs) and rapid communication service information letters (RICSILs) for previous BWR designs applicable to the ABWR. In the ABWR design either the components addressed by the SIL or RICSIL (e.g., jet pumps) have been eliminated or the need for the examination no longer exists because of the elimination of crevices and the use of materials resistant to the known

Reactor Coolant System and Connected Systems

degradation mechanisms (e.g., IGSCC) on which the SIL and RICSIL examinations were based. GE's proposed methodology for NDE of the reactor vessel and personnel qualifications are acceptable because the concepts described above have been successfully used in existing operating BWRs.

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

As a vendor-specific issue, GE based its design of the RPV on the 1989 Edition of the ASME Code, which is a national standard referenced by 10 CFR 50.55a(b). The staff reviewed the design of the RPV using the 1989 Edition of the ASME Code to determine if the design is technically acceptable. In the DFSEER, the staff concluded that GE's proposal related to the examination of the RPV will be technically acceptable, if it is properly described in the SSAR and the Tier 1 document. GE was asked to revise the SSAR to indicate that it had based its design of the RPV on the 1989 Edition of the ASME Code and that PSI will be performed in accordance with that same code edition, rather than the code in effect at the time of COL application (as required by 10 CFR 50.55(g)). In addition, a discussion of the PSI and the 1989 Edition of the Code will be added to the RPV Tier 1 document design information. This was DFSEER Confirmatory Item 5.2.4-1.

GE responded to this item in SSAR Section 5.2.4 by stating that the design for performing PSI on the reactor vessel shall be based on the requirements of ASME Code, Section XI, 1989 Edition. For the required preservice examination, the reactor vessel shall meet the acceptance standards of Section XI, IWB-3510. The RPV shell welds are designed for 100-percent accessibility for both PSI and ISI. The RPV nozzle-to-shell welds will be 100-percent accessible for PSI but might have limited areas that will not be accessible from the outer surface for performing inservice examination techniques. The staff reviewed revised SSAR Section 5.2.4 and found it acceptable. After a discussion with the applicant, the staff decided that PSI requirements will include ultrasonic examination in addition to the radiographic examination required by Section III of the Code. These requirements are included in the RPV Tier 1 design description. However, the specific edition of the ASME Code (1989) is considered

Tier 2 information as previously discussed in Section 5.2.1.1 of this report. This item is resolved.

5.2.5 Reactor Coolant Pressure Boundary Leakage Detection

The staff reviewed the RCPB leakage detection systems in accordance with SRP Section 5.2.5. Staff acceptance of the RCPB leakage detection systems is based on the design meeting the requirements of GDC 2 as it relates to the ability of systems to maintain and perform their safety functions following an earthquake and meeting the requirements of GDC 30 as it relates to the detection, identification, and monitoring of the source of reactor coolant leakage. Conformance with GDC 2 is based on the design meeting the guidelines of RG 1.29, Positions C.1 and C.2 while conformance with GDC 30 is based on the design meeting the guidelines of RG 1.45, Positions C.1 through C.9.

A limited amount of leakage is to be expected from components forming the RCPB. Leakage is classified into two types, identified and unidentified. Components such as valve stem packing, pump shaft seals, and flanges will not be completely leaktight. Because some leakage is expected from them, it will be collected and monitored. This leakage from the selected components (1) is considered identified leakage and (2) its total flow rate will be monitored separately from unidentified leakage (which may be symptomatic of an unexpected failure of the RCPB). Items (1) and (2) above are requirements of Position C.1 of RG 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," Revision 0. The sensitivity of detection methods (3.79 L/min (1 gpm)) and leakage limits (ranging from 3.79 L/min (1 gpm) to 19 L/min (5 gpm)) for unidentified leakage will be selected to detect and correct potential through-wall flaws (cracks) in the RCPB before such cracks can grow sufficiently to threaten the safety of the plant.

Reactor coolant leakage for the ABWR involves leakage within the drywell, leakage external to the drywell (i.e., in the equipment areas in the reactor building, the main steam tunnel, and the turbine building) and intersystem leakage. These are discussed below.

(1) Leakage Within the Drywell

Identified Leakage: Within the drywell, the drywell equipment drain sump will collect leakage from the reactor vessel head flange inner seal, stem inner packing for large remote power-operated valves, and other known leakage sources. The sump is designed with two pumps, timers, and

AIM

Association for Information and Image Management

1100 Wayne Avenue, Suite 1100
Silver Spring, Maryland 20910
301/587-8202

Centimeter

Inches

MANUFACTURED TO AIM STANDARDS
BY APPLIED IMAGE, INC.

3 of 10

instrumentation for control room monitoring. The sump instrumentation and timers will monitor identified leakage by measuring the rate of change in sump level and the sump's fill-up and pump-out times. The ABWR design provides control room indication and alarm capabilities. The sump level monitoring instrument and the fill-up and/or pump-out timer will activate an alarm in the control room when the total leak rate reaches 95 L/min (25 gpm). Different parameters will monitor leakage from individual sources. The leakage from the head flange inner seal will be monitored by means of a seal drain line pressure instrument. Safety/ relief valve (SRV) and valve stem leakage will be monitored by temperature sensors provided on each SRV discharge line. These monitors will continuously indicate and/or record leakage in the control room. In addition, the monitors will trip and activate an alarm in the control room if leakage from the monitored components is detected. The stems for large power-operated valves will be equipped with drain lines to the sump. A remote-operated solenoid valve installed in each drain line can be used during plant operation, in conjunction with additional sump instrumentation, to identify leaking valve packing and to isolate leakage flow from the valve stem's inner seal.

Unidentified Leakage: Within the drywell, the drywell floor drain sump will collect unidentified leakage from sources such as control rod drives, valve flanges, closed cooling water for reactor services (e.g., cooling of RIP motor), condensate from the drywell atmosphere coolers, and other leakage not collected in the drywell equipment drain sump. The sump is designed with two pumps, timers, and instrumentation for control room monitoring. Sump fill-up and pump-out times will be monitored. The instrumentation will activate an alarm in the control room when preset limits are reached. The rate of change in sump level will be indicated continuously in the control room. Other primary detection methods for small unidentified leaks will include increases in condensate flow rate from the drywell air coolers and increases in the radioactivity count level for noble gases, iodines, and particulates in the drywell atmosphere. The flow rate for condensate, which results from condensation of thermally hot leakage inside the drywell, will be monitored by flow instrumentation provided on the common drain line for the condensate from all the coolers. The radioactivity count levels will be monitored by drywell radiation monitors. These variables will be continuously indicated and/or recorded in the control room. The

monitors will activate alarms in the control room when their preset limits are reached. The sensitivity and response time for all these primary detection systems is 3.79 L/min (1 gpm) or its equivalent in less than 1 hour, thus satisfying Positions C.2 and C.5 of RG 1.45, Revision 0. Secondary detection methods will include monitoring the drywell temperature and pressure for gross unidentified leakage. These variables will be recorded in the control room and will activate control room alarms when they are high. Excessive leakage inside the drywell (e.g., during a loss-of-coolant accident (LOCA)) will be detected by high drywell pressure, low reactor vessel pressure, or high steamline flow (for breaks downstream of the flow elements). The instrumentation for these monitored variables will trip, activate alarms and isolate the appropriate valves when their predetermined limits are exceeded. Position C.3 of RG 1.45, Revision 0, states that at least three methods of leak detection should be used in the design. Position C.7 of RG 1.45, Revision 0, states that indicators and alarms for each leakage detection system should be provided in the main control room. Also, it states that procedures for correcting various indications to a common leakage equivalent should be available to reactor operators. The leakage detection methods described above meet Position C.3 of RG 1.45, Revision 0, and are acceptable. The instrumentation and alarms discussed in the SSAR meet the indicator and alarm provisions of Position C.7 of RG 1.45, Revision 0 and are acceptable. The applicant referencing the ABWR design will provide the procedures to convert leakage indicators to a common leakage equivalent.

The total leakage rate from the RCPB consists of all leakage, identified and unidentified, that flows into the drywell floor drain and equipment drain sumps. GE specifies an identified leakage rate limit of 95 L/min (25 gpm) and an unidentified leakage rate limit of 3.79 L/min (1 gpm) from the RCPB. GE also states that the total leakage rate from both categories from the RCPB is limited to 95 L/min (25 gpm).

Position C.9 of RG 1.45, Revision 0, requires that the TS include limiting conditions for identified and unidentified leakage and address the availability of various instruments to ensure adequate coverage. The technical specification limits on RCS leakage are 3.79 L/min (1 gpm) for unidentified leakage and 98.4 L/min (26 gpm) for total leakage. Since the above limits are either equal to or less than

Reactor Coolant System and Connected Systems

current BWR standard TS limits for RCPB leakage limits, the system meets Position C.9 of RG 1.45, Revision 0, and is acceptable. This was DFSER TS Item 5.2.5-1. GE has provided the applicable TS. Therefore, DFSER TS Item 5.2.5-1 is resolved.

(2) Leakage External to the Drywell

The areas designated as external to the drywell that are monitored for reactor coolant leakage include the equipment areas in the reactor building, the main steam tunnel, and the turbine building. The process piping for each monitored system is located in separate compartments or rooms, where feasible, to facilitate the detection of the leak through area temperature indications. Each leakage detection system has the capability to detect leak rates that are less than the established limits for identified and unidentified leakage from the RCPB. Many detection and monitoring systems will be used to detect leakage external to the drywell.

- Equipment Areas in the Reactor Building

Leakage from unknown or unidentified sources, including shutdown cooling system piping, reactor water cleanup system (CUW) piping, process instrumentation piping, and control rod drive hydraulic control unit (HCU) piping is collected in several reactor building floor drain sumps. Identified leakage from known sources will be collected in reactor building equipment drain sumps.

Pumps, timers, and instrumentation used for processing and monitoring the reactor building equipment and floor drain sumps will be the same as those provided for the drywell equipment and floor drain sumps. Sump levels and sump fill-up and pump-out times will be monitored. Alarms will be initiated in the control room when the applicable set points are reached.

In addition to the above parameters (i.e., sump levels and sump fill-up and pump-out times), other parameters also will be used for detecting leakage. Equipment rooms that house the RCIC, RHR, and CUW components will be monitored by dual-element thermocouples that sense high ambient temperature. These sensors will indicate possible reactor coolant leaks in these areas. The high-temperature condition

will be indicated, alarmed, and recorded in the control room. In some cases, a high-temperature condition will provide isolation signals to close appropriate valves. These monitors will be suitable to detect leakage of 95 L/min (25 gpm) or less into the monitored areas. In addition to area temperature monitoring, leakage external to the drywell originating from specific systems can be detected by monitoring other parameters such as low steam pressure and high steam flow in the RCIC system and main steamlines (MSLs), high RCIC exhaust line diaphragm pressure, and high differential flow between suction and discharge lines of the CUW system. These detection systems will activate alarms in the control room and/or initiate closure of applicable valves when the preset limits for the corresponding parameters are reached.

- Main Steam Tunnel

Leakage within the main steam tunnel will be detected by monitoring area temperatures and radiation. MSL area temperature will be monitored by thermocouples located in the area of the main steam and RCIC pipelines. All temperature elements will be located and shielded so that they will be sensitive to ambient air temperature rather than the radiated heat from hot equipment. The monitors will alarm in the control room and provide isolation signals to the MSL and the MSL drain line isolation valves as well as the CUW isolation valves. If the leakage is from the feedwater system, the system will be isolated manually. These monitors will be suitable to detect leakage of 95 L/min (25 gpm) or less into the monitored areas. MSL radiation will be monitored by the process radiation monitoring system (PRMS). The PRMS trip functions will include isolation of the MSIVs and reactor scram. The PRMS will initiate control room readouts and alarms.

- Turbine Building

Reactor coolant leakage in the turbine building will be detected by monitoring MSL pressure, main condenser vacuum, and turbine building ambient temperature in areas traversed by the MSL. These monitors will alarm and indicate in the control room and, in some cases, will also provide signals for isolating the MSIV and MSL drain lines.

Large leaks external to the drywell will be detected by monitoring MSL flow rate, reactor vessel level, RCIC steamline flow rate, and low pressure on the RCIC steamline or the RCIC turbine. Abnormal conditions detected by any of the above monitors will alarm in the control room, and isolation of appropriate system(s) will be initiated.

(3) Intersystem Leakage

Position C.4 of RG 1.45, Revision 0, states that provisions should be made in the design to monitor systems connected to the RCPB for signs of intersystem leakage.

The ABWR design provides for monitoring intersystem leakages (i.e., leakages from the reactor coolant system (RCS) into other connected systems). Detection of leakage into systems connected to the RCS is necessary because intersystem leakage could result in damage to those systems. The systems of concern are:

- HPCF system
- RCIC system
- RHR shutdown cooling (SDC) system
- RHR low-pressure core flooders (LPFL)
- secondary side of RHR heat exchangers
- secondary side of RIP heat exchangers
- secondary side of CUW heat exchangers
- secondary side of fuel pool cooling (FPC) heat exchangers

In response to request for additional information (RAI) Question (Q)430.2c, GE submitted information regarding the detection of intersystem leakage for systems connected directly to the RCS. In the DFSEER, the staff stated that this information should be incorporated into SSAR Section 5.2.5. This was identified as Confirmatory Item 5.2.5-1. GE added a reference to the response to RAI Q430.2c in the text of SSAR Section 5.2.5.2.2(11). Since the response is presented in SSAR Chapter 20 and is adequately referenced in Section 5.2.5, Confirmatory Item 5.2.5-1 is resolved. GE's response to the RAI is summarized below.

The ABWR high-pressure safety injection system consists of two HPCF divisions (Divisions B and C) and the RCIC system. There is no connection between the RCS and the inlet suction of HPCF divisions B and C. Both divisions will draw their suction flow from the condensate storage pool or

suppression pool, and not from the RCS. The discharge lines of the HPCF divisions connect to the RCS through normally closed discharge check valves and injection valves. Substantial (potential) leakage through the closed discharge check valves and closed injection valves into either of the discharge lines would result in pressurization of both the discharge line and the HPCF pump suction line, leading to a control room alarm indicating high HPCF B (or C) suction pressure. Significant pressurization of the suction piping would result in a discharge to the suppression pool via a pressure relief valve.

There is no connection between the RCS and the suction or discharge of the RCIC system. The system will draw its suction flow from the condensate storage pool or suppression pool, and not from the RCS. The discharge line of the RCIC system connects to feedwater line B through normally closed discharge check and injection valves and is not considered to be connected to the reactor system pressure boundary. Therefore, the provisions of SRP Section 5.2.5 are not applicable to the RCIC system.

Each of the three suction lines for the shutdown cooling mode of the RHR system connect to the RCS through a shutdown cooling line suction valve and through both an inboard and an outboard containment isolation valve. Substantial (potential) leakage through both of the closed containment isolation valves into any of the suction lines would be detected by pressure sensors located between the outboard containment isolation valve and the key-locked closed RHR shutdown cooling mode suction valve. High pressure in this section of piping would result in a control room alarm. Significant pressurization of this section of piping would result in a discharge to the suppression pool via a pressure relief valve.

In the LPFL mode of RHR, suction flow will be drawn from the suppression pool through normally opened RHR pump suction valves, and not from the RCS.

The RHR discharge lines will be used by all three RHR divisions to return flow to the reactor for both the shutdown cooling mode or LPFL mode of operation. The discharge lines normally will be filled with water by the RHR discharge line fill pumps. Only RHR divisions B and C will discharge directly into the RPV; division A will discharge into feedwater line A. Therefore,

Reactor Coolant System and Connected Systems

intersystem leakage is only postulated to occur in RHR divisions B and C. The discharge lines of RHR connect to the RPV through normally closed discharge check valves and injection valves. Substantial (potential) leakage through the closed discharge check valves and closed injection valves into either of the discharge lines would result in pressurization of the discharge line, leading to a control room alarm. Significant pressurization of the discharge piping would result in a discharge to the suppression pool via a pressure relief valve.

Radiation monitors will detect reactor coolant leakage into the reactor building cooling water (RCW) system, which will provide cooling water to the RHR heat exchangers, the RIP heat exchangers, the CUW nonregenerative heat exchangers (NRHXs), and the fuel pool pooling and cleanup (FPC) heat exchangers. At least two process radiation monitoring channels will monitor leakage into each of the two common cooling water headers that will receive RCW return flow from the above heat exchangers. Each channel will alarm on high-radiation conditions that indicate process leakage into the CUW system.

The staff concludes that intersystem leakage applicable to the ABWR, such as that into the RHR, HPCF, and RCIC systems, would be highly unlikely because it would have to occur through closed check valves and/or closed containment isolation valves. All potential intersystem leakage from the RCPB will be into closed systems, normally filled with water. Therefore, indicators for abnormal water levels or flows in the affected areas will not be used for monitoring intersystem leakage. For these systems, high-pressure alarms will be used. Thus, the ABWR design satisfies the requirements of Position C.4 of RG 1.45, Revision 0, and is acceptable.

Position C.8 of RG 1.45, Revision 0, states that leakage detection systems should be equipped with provisions to permit testing for operability and calibration during plant operation.

The ABWR leakage detection systems are equipped with features to permit operability testing and calibration during plant operation using the following methods:

- simulation of input signals into trip units
- comparison of methods (e.g., comparison of airborne particulate monitoring or air cooler condensate flow with sump fill-up rate)
- comparison of channels when more than one channel is used for any one detection method (e.g., area temperature monitoring)

These methods meet Position C.8 of RG 1.45, Revision 0, and are acceptable.

However, in the DFSER, the staff noted that the COL applicant should provide information regarding the sensitivity of detection methods and leakage limits for unidentified leakage and procedures and graphs for converting various indicators into a common leakage equivalent to meet the procedures portion of Position C.7 of RG 1.45. This was DFSER COL Action Item 5.2.5-1. GE has included this action in SSAR Section 5.2.5.9. This is acceptable.

GE originally submitted the design description and the ITAAC relating to RCPB leakage detection. At the time the DFSER was issued, the ITAAC review was in progress. Therefore, this was DFSER Open Item 5.2.5-1. GE has since provided revised design description and ITAAC. The adequacy and acceptability of the design description and the ITAAC are evaluated in Chapter 14.3 of this report. Therefore, DFSER Open Item 5.2.5-1 is resolved.

The leakage detection systems proposed for the ABWR design meet RG 1.45, Revision 0, Positions C.1 through C.5, C.8, and C.9. RG 1.29, Revision 3, states in Positions C.1 and C.2 that systems required for monitoring systems important to safety should be designed to meet seismic Category I standards and to withstand the effects of a safe shutdown earthquake (SSE) without failure. RG 1.45, Revision 0, indicates in Position C.6 that the airborne particulate monitoring system should function when subjected to an SSE, and the leakage detection system should be capable of performing its functions following seismic events not requiring shutdown. The ABWR leakage detection systems are designed to remain functional following seismic events. Specifically, the drywell airborne particulate radioactivity monitoring system is designed to seismic Category I standards. Thus, the requirements of GDC 2, "Design Basis for Protection Against Natural Phenomena," and the guidelines of RG 1.29, Revision 3, Positions C.1 and C.2, and of RG 1.45, Revision 0, Position C.6, are satisfied.

Indicators and alarms for each leakage detection system are provided in the control room. Level and flow from floor and equipment drain sumps, both internal and external to the drywell, will be monitored for leakage. Condensate flows from the drywell air cooler also will be monitored and will alarm in the control room on high flow.

The staff concludes that the leakage detection systems provided to detect leakage from components of the RCPB furnish reasonable assurance that structural degradation that may develop in pressure-retaining components will be detected on a timely basis. Thus, corrective action can be taken before such degradation becomes sufficiently severe to jeopardize the safety of the system, or before the leakage increases to a level beyond the capability of the makeup system to replenish the loss. On the basis of the information in the SSAR and the evaluation above, the staff concludes that the systems are in conformance with the guidelines of RG 1.29, Revision 3, Positions C.1 and C.2, and RG 1.45, Revision 0, Positions C.1 through C.9, and satisfy the requirements of GDC 2 and 30. Therefore, the RCPB leakage detection systems meet the acceptance criteria of SRP Section 5.2.5 and are acceptable.

5.3 Reactor Vessel

5.3.1 Reactor Vessel Materials

The materials specified for construction of the reactor vessel and its appurtenances have been identified by specification on SSAR Table 5.2-4 and found to be in conformance with Section III of the ASME Code. GE has identified special requirements with regard to the control of residual elements that the staff considers acceptable. Compliance with the provisions of the Code for material specifications satisfies GDC 1 and 30 and 10 CFR 50.55a.

Ordinary processes will be used for the manufacture, fabrication, welding, and NDE of the reactor vessel and its appurtenances. Fabrication processes and NDEs will be performed in accordance with the requirements specified in the ASME Code. Compliance with these ASME Code provisions meets GDC 1 and 30 and 10 CFR 50.55a.

In the DSER (SECY-91-153), the staff noted that GE should discuss the onsite cleaning and cleanliness controls for austenitic stainless steel. This was DSER Outstanding Issue 21. As discussed in SSAR Section 5.3.1, the controls to be used during all stages of welding to prevent contamination and sensitization that could cause stress corrosion cracking in austenitic stainless steel conform with the recommendations of RGs 1.37, Revision 0, and 1.44, Revision 0, and NUREG-0313, Revision 2. These controls will provide reasonable assurance (1) that austenitic stainless steel components will be properly cleaned on site, thus satisfying Appendix B to 10 CFR Part 50, and (2) that welded components will not be contaminated or excessively sensitized before or during the welding process, thus satisfying GDC 1, 4, and 30 and 10 CFR 50.55a. This is acceptable and resolved DSER Outstanding Issue 21.

When components of austenitic stainless steels are welded, ASME Code controls will be supplemented by conformance with the recommendations of RG 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal," Revision 3, and NUREG-0313, Revision 2. These controls will provide reasonable assurance that the welds will not contain microcracks and will be resistant to IGSCC and satisfy GDC 1, 14, and 30 and 10 CFR 50.55a as related to the integrity of the RCPB.

When components of low-alloy steels are welded, ASME Code controls will be supplemented by conformance with the recommendations of RG 1.50, "Control of Preheat Temperature for Welding of Low-Alloy Steel," Revision 0. Low-alloy steel components either will be held for an extended time at preheat temperatures to ensure the removal of hydrogen, or preheat will be maintained until postweld treatment. This will provide reasonable assurance that components made from low-alloy steels will not crack during fabrication and will minimize the potential for subsequent cracking. Adherence to the recommendations of RG 1.43, "Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components," Revision 0, is not necessary because the RPV specifications require that all low-alloy steel be produced in accordance with fine grain practice. This will provide reasonable assurance that underclad cracking will not occur during the weld cladding process. These controls satisfy GDC 1 and 30 and 10 CFR 50.55a.

Integrity of the reactor vessel studs and fasteners is ensured by conformance with the recommendations of RG 1.65, "Materials and Inspections for Reactor Vessel Closure Studs," Revision 0, which satisfies GDC 1, 30, and 31 and 10 CFR 50.55a and the requirements of Appendix G to 10 CFR Part 50.

The staff reviewed the fracture toughness of the reactor vessel materials, the RCPB materials, and the materials surveillance program for the reactor vessel beltline region according to SRP Section 5.3.1.

GDC 31 require that the RCPB be designed with sufficient margin to ensure that, when stressed under operating, maintenance, and test conditions, the boundary behaves in a nonbrittle manner and the probability of rapidly propagating fracture is minimized. GDC 32 require that the RCPB be designed to permit an appropriate material surveillance program for the RPV. The fracture toughness requirements for the ferritic materials of the RCPB are defined in Appendices G and H to 10 CFR Part 50.

The edition and addenda of the ASME Code that are applicable to the design and fabrication of the reactor vessel and RCPB components are specified in 10 CFR

Reactor Coolant System and Connected Systems

50.55a. The ASME Code edition and addenda that are applicable to the ABWR are specified in SSAR Tables 1.8-21 and 3.2-3.

In the ABWR SSAR, GE states that the reactor vessel will comply with the fracture toughness requirements of Appendix G to 10 CFR Part 50. However, the staff concluded in the DFSER that to confirm this conclusion, the COL applicant must provide fracture toughness data based on the limiting reactor vessel materials. This was DFSER COL Action Item 5.3.1-1.

GE responded to this item in SSAR Section 5.3.4 by stating that the COL applicant will provide fracture toughness data based on the limiting reactor vessel materials. This is acceptable.

Appendix G, "Protection Against Nonductile Failures," to Section III of the ASME Code is used, together with the fracture toughness test results required by Appendices G and H to 10 CFR Part 50, to calculate the pressure-temperature limits for the ABWR reactor vessel. The fracture toughness tests required by the ASME Code and Appendix G to 10 CFR Part 50 provide reasonable assurance that adequate safety margins against the possibility of nonductile behavior or rapidly propagating fracture can be established for all pressure-retaining components of the RCPB. These methods will provide adequate safety margins during operating, testing, maintenance, and anticipated transient conditions. Compliance with these code provisions and NRC regulations is acceptable and satisfies GDC 31.

The materials surveillance program will be used to monitor changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region resulting from exposure to neutron irradiation and the thermal environment, as required by GDC 32. The ABWR surveillance program, which must comply with Appendix H to 10 CFR Part 50 and American Society for Testing and Materials (ASTM) E-185-82, "Standard Recommended Practices for Surveillance Tests for Nuclear Reactor Vessels," requires that fracture toughness data be obtained from material specimens that are representative of the limiting base weld and heat-affected-zone materials in the beltline region. These data will permit the determination of the conditions under which the vessel can be operated with adequate margins of safety against fracture throughout its service life.

In the SSAR, GE states that the reactor vessel materials surveillance program will comply with the requirements of Appendix H to 10 CFR Part 50 and ASTM E-185-82. However, the staff concluded in the DFSER that to

confirm this conclusion, the COL applicant must provide details of its surveillance program including the specific materials in each surveillance capsule, the capsule lead factors, the withdrawal schedule for each capsule, the neutron fluence to be received by each capsule at the time of its withdrawal, and the vessel end-of-life peak neutron fluence. This was DFSER COL Action Item 5.3.1-2.

GE responded to this item in SSAR Section 5.3.4 by stating that the COL applicant will submit the following information to the NRC: the specific materials in each surveillance capsule, the withdrawal schedule for each surveillance capsule, the neutron fluence to be received by each capsule at the time of its withdrawal, and the vessel end-of-life peak neutron fluence. This is acceptable.

The materials surveillance program will generate information on the effects of irradiation on material properties so that changes in the fracture toughness of the material in the ABWR reactor vessel beltline region can be properly assessed and adequate safety margins against the possibility of vessel failure can be provided. The surveillance program must generate sufficient information to permit the determination of conditions under which the reactor vessel will be operated with an adequate margin against rapidly propagating fracture throughout its service lifetime. On the basis of a 40-year design life, ASTM E-185-82 recommends that three materials surveillance capsules be installed in the reactor vessel beltline. However, in the DFSER, the staff noted that for the ABWR, the design life is expected to be increased to 60 years. Accordingly, GE needed to reassess the number of materials surveillance capsules to be provided to account for the additional 20-year increase in the expected life of the vessel. The staff noted that GE should revise its SSAR to address changes to the materials surveillance program, including the number and location of the vessel surveillance capsules, for the ABWR reactor vessel to account for the 60-year design life of the plant. This was DFSER Open Item 5.3.1-1.

GE addressed this open item in the SSAR. Specifically, GE committed to provide four surveillance capsules instead of the three previously proposed capsules. GE also required that the capsules be placed to produce a lead factor of approximately 1.2 to 1.5. Furthermore, GE specified that the weld metal specimens be made from the same heat of weld wire and lot of flux and that the same welding practice as that for the beltline weld be used. GE also calculated that the predicted end-of-license adjusted reference temperature of the reactor vessel will be less than 38 °C (100 °F). The staff finds this response acceptable because these requirements will ensure that the surveillance program will generate sufficient information

to determine the conditions under which the reactor vessel will be operated throughout its 60-year service lifetime. Therefore, this open item is resolved.

Compliance with Appendix H to 10 CFR Part 50 and ASTM E-185-82 ensures that the surveillance program will be capable of monitoring radiation-induced changes in the fracture toughness of the reactor vessel material and will satisfy the requirements of GDC 32 regarding the design of an appropriate materials surveillance program for the reactor pressure vessel.

5.3.2 Pressure-Temperature Limits

The acceptance criteria for reviewing the pressure-temperature limits are given in SRP Section 5.3.2 and Appendices G and H to 10 CFR Part 50. Appendices G and H describe the operating conditions that require pressure-temperature limits and provide the bases for these limits, specifically requiring that pressure-temperature limits provide safety margins at least as great as those recommended in ASME Code, Section III, Appendix G. Appendix G to 10 CFR Part 50 requires additional safety margins whenever the reactor core is critical (except for low-level physics tests) for the materials in the closure flange and beltline regions.

The staff reviewed the pressure-temperature limits that will be imposed on the RCPB during the following operations and tests to ensure that there will be adequate safety margins against nonductile behavior or rapidly propagating failure of ferritic components as required by GDC 31:

- preservice hydrostatic tests
- inservice leak and hydrostatic tests
- heatup and cooldown operations
- core operation - criticality

Appendices G and H to 10 CFR Part 50 require the applicant to predict the amount of increase in reference temperature, RT_{NDT} , resulting from neutron irradiation. The increase in RT_{NDT} resulting from neutron irradiation is then added to the initial RT_{NDT} and the margin to establish the adjusted reference temperature. The staff's recommended method for calculating the increase in RT_{NDT} resulting from neutron irradiation is contained in RG 1.99, "Radiation Embrittlement of Reactor Vessel Materials," Revision 2. The relationships in the guide were derived by statistical analysis of 216 material data points that were reported from the testing of irradiated materials.

These materials were contained in surveillance capsules and had been irradiated inside U.S. commercial nuclear reactor vessels. As more surveillance data become

available, this guide may need additional revision, since the relationship between the increase in RT_{NDT} and neutron fluence is empirically derived from analysis of material surveillance data on U.S. commercial nuclear reactor vessels.

The pressure-temperature curves in SSAR Figure 5.3-1 are system hydrotest limits with fuel in the vessel, non-nuclear heating limits, and nuclear (core critical) limits. In the DFSE, the staff noted that these three limits were different from the suggested limits on preservice hydrostatic tests, inservice leak and hydrostatic tests, heatup and cooldown operations, and core operation in SRP Section 5.3.2. The pressure-temperature curves in the SSAR were generic and were not valid for any specific effective full-power years. The staff found that the general shapes of the curves are acceptable, but the curves should be used as a reference only. For approval of any COL, COL applicants must submit plant-specific calculations of RT_{NDT} , stress intensity factors, and pressure-temperature curves similar to those in RG 1.99 and SRP Section 5.3.2. This was DFSE COL Action Item 5.3.2-1.

GE responded to this item in the SSAR, Section 5.3.4, by stating that the COL applicant will submit plant-specific calculations of RT_{NDT} , stress intensity factors, and pressure-temperature curves similar to those in RG 1.99 and SRP Section 5.3.2. This is acceptable.

In response to the staff's questions, GE submitted a calculation of the RT_{NDT} shift for the vessel plate and weld metal. The calculation followed RG 1.99, closely. The calculated RT_{NDT} was low (-22 °C (-8 °F) for the weld metal and -8.4 °C (17 °F) for the plate) because of the low neutron fluence and the low copper and nickel contents used in the calculation.

GE predicted the neutron fluence at end of life to be 6×10^{17} n/cm², which is low in comparison to the existing BWR design. In the DFSE, the staff noted that GE should submit additional information to show how the fluence value of 6×10^{17} n/cm² was predicted. This was DFSE Open Item 5.3.2-1.

GE responded to this item in SSAR Section 5.3.2.1.5. It stated that fast neutron fluence for the ABWR vessel was evaluated using the Oak Ridge National Laboratory code on the CRAY X-MP Super Computer based on an eighth core symmetry fixed source model. The neutron source was based on a three-dimensional nodal fuel model of ABWR for an integrated equilibrium core with a 26-group neutron spectrum. The results were found to be reasonable in comparison to the BWR/6 calculations performed earlier by the Department of Energy (DOE). In evaluating the relative fluence, the power level and shroud-

Reactor Coolant System and Connected Systems

to-vessel water thickness were taken into account. The incorporation of integral pumps increased the annulus between the shroud and the vessel walls for the ABWR. This led to an order of magnitude reduction in expected fluence. Therefore, this item is resolved.

The staff concludes that the pressure-temperature limits imposed on the RCS for operating and testing conditions to ensure adequate safety margins against nonductile or rapidly propagating failure will be in conformance with the fracture toughness criteria of Appendix G to 10 CFR Part 50 and Section III, including Appendix G, of the ASME Code. The use of operating limits, based on the criteria in SRP Section 5.3.2, provides reasonable assurance that nonductile or rapidly propagating failure will not occur and constitutes an acceptable basis for satisfying the requirements of 10 CFR 50.55a and GDC 1, 14, 31, and 32.

5.3.3 Reactor Vessel Integrity

Although the staff reviewed most areas separately in accordance with its review plans, reactor vessel integrity is of such importance that a special summary review of all factors relating to it was warranted. The staff reviewed the fracture toughness of the ferritic materials for the reactor vessel and the RCPB, the pressure-temperature limits for operation of the reactor vessel, and the materials surveillance program for the reactor vessel beltline. The acceptance criteria and references that are the bases for this evaluation are given in SRP Section 5.3.3.

The staff reviewed the information in each area to ensure that no inconsistencies existed that would reduce the certainty of vessel integrity. The areas reviewed and the sections of this report in which they are discussed are given below.

- RCPB materials design (Section 5.2.3)
- ISI and testing of RCPB (Section 5.2.4)
- reactor vessel materials fabrication methods (Section 5.3.1)
- pressure-temperature limits and operating conditions (Section 5.3.2)

The staff concludes that the structural integrity of the reactor vessel is acceptable and meets the requirements of GDC 1, 4, 14, 30, 31, and 32 of Appendix A to 10 CFR Part 50; Appendices B, G, and H to 10 CFR Part 50; and 10 CFR 50.55a.

The basis for this conclusion is that the design, materials, fabrication, inspection, and quality assurance requirements for the ABWR plant satisfy the NRC regulations and

regulatory guides and the rules of the ASME Code, Section III. The stringent fracture toughness requirements of the regulations and the ASME Code, Section III will be met, including requirements for surveillance of vessel materials properties throughout service life, in accordance with Appendix H to 10 CFR Part 50. Also, operating limitations on temperature and pressure will be established for the plant in accordance with Appendix G, "Protection Against Nonductile Failure," to ASME Code Section III and Appendix G to 10 CFR Part 50.

The integrity of the reactor vessel is assured because the vessel

- (1) will be designed and fabricated to the high standards of quality required by ASME Code and by any pertinent Code Cases;
- (2) will be made from material of controlled and demonstrated high quality;
- (3) will be subjected to extensive preservice inspection and testing to provide assurance that the vessel will not fail because of material or fabrications deficiencies;
- (4) will operate under conditions and procedures and protective devices that provide assurance that the vessel design conditions will not be exceeded during normal reactor operation, maintenance, testing, and anticipated transients;
- (5) will be subjected to periodic inspection to demonstrate that the high initial quality of the reactor vessel has not deteriorated significantly under service conditions;
- (6) will be subjected to surveillance to account for neutron irradiation damage so that the operating limitation may be adjusted.

5.4 Components and Subsystem Design

5.4.1 Reactor Recirculation System

This system is not addressed in the SRP. However, since the ABWR reactor recirculation system is unique, as compared to the current BWR designs, an evaluation of the system is provided.

A significant change in the ABWR from current BWR designs is the elimination of the external loops and the incorporation of RIPs for reactor coolant recirculation. The containment volume is reduced as a consequence of

using RIPs instead of external pumps because the external large pipes of the reactor recirculation system are eliminated. (Rupture of large-bore external pipes in the lower part of the reactor vessel is eliminated as the design-basis accident). There is no pipe larger than approximately 5 cm (2 in.) below the core; thus, the probability of fuel uncover is reduced during a loss-of-coolant (LOCA). This improves plant safety performance. The use of RIPs requires a vessel with a larger diameter, approximately 706 cm (278 in.), to allow pump impeller removal. However, this results in reduced neutron flux at the vessel beltline and reduced reactor pressure rates during transients. Full-power operation will be possible with one RIP out of service, improving plant availability. Maintenance of RIPs will be easier than that of current recirculation pumps with less radiation exposure to plant personnel.

The reactor recirculation system consists of 10 RIPs with their shafts, impellers, and diffusers internal to the reactor vessel and will be removed from above for service. The RIPs are mounted vertically onto and through the pump nozzles which are arranged in an equally spaced ring pattern on the bottom head of the RPV. The RIPs are single-stage, vertical pumps driven by variable-speed induction motors. The pump speed will be changeable by varying the voltage and frequency output of the individual pump motor's electrical power supply. The RIPs will provide recirculation flow from the downcomers through the lower plenum and up through the lower grid, the reactor core, and steam separators. The flow rate will range from minimum flow, established by the pump characteristics, to above the maximum flow required to obtain rated reactor power.

The RIP motors are variable-speed, four-pole, ac induction wet-motor type. The operating speed of the pump motor will depend on the variable voltage and the variable frequency output of the adjustable speed drives. The RIP motors will be cooled by circulating water in the motor cavity through the shell side of the recirculation motor heat exchanger. Hot water in the shell side of the heat exchanger will be cooled by the reactor building cooling water system. There is one heat exchanger per motor. A clean purge flow will be provided by the control rod drive system to prevent reactor water from entering the motor cavity region, thereby preventing any impurity buildup. Also, anti-reverse rotation devices will be installed on the motor shaft to prevent possible damage of the motor caused by reverse pump flow. For service, the motors will be removed from below the RPV.

The staff discusses acceptability of the flow coastdown characteristics to maintain fuel thermal margins during abnormal operational transients in Chapter 15 of this report. The RIP inertia is significantly less than the pump

inertia for operating BWRs (i.e., about 0.7-second inertia time constant for the ABWR compared with about 3 to 5 seconds for operating BWRs). The ABWR design also uses motor generator (MG) sets as pass-through energy devices on 6 of the 10 RIPs. The MG sets will be powered from two separate electrical distribution systems and used to lengthen the coastdown of two groups of three RIPs on loss of offsite power (LOOP), thus preventing the departure from nucleate boiling on an all-pump trip. No new safety concerns are associated with the use of RIPs, as compared with external recirculation loops. There is significant operating experience with the RIPs. In Europe, nearly 100 pumps are operating in BWRs, approaching 600 pump-years of experience. Few major forced outages have been reported as a result of pump-related problems. Thus, RIP operation did not require the development of new technology.

Tests conducted to verify that the core flow pattern for 9-pump operation is not significantly different from 10-pump operation showed that the flow distribution to the reactor core is uniform even if one RIP is idle.

GE originally submitted the design description and the ITAAC for the recirculation flow control system. At the time the DFSE was issued, the ITAAC review was in progress. Therefore, this was DFSE Open Item 5.4.1-1. GE has since provided revised design description and ITAAC. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

Since the operating limits such as maximum critical power ratio are calculated in Chapter 15 of this report for abnormal transients using RIP coastdown and are found to be acceptable, the staff concludes that the ABWR reactor recirculation system is acceptable.

5.4.2 Steam Generators

This section is not applicable because the ABWR does not use steam generators for power generation.

5.4.3 Reactor Coolant Piping

This section does not apply because the reactor internal pumps (RIPs) are located inside the reactor pressure vessel (RPV) and there is no major external reactor coolant piping connected to it.

5.4.4 Main Steam Line Flow Restrictions

The functional requirements of the main steam line restrictors are reviewed in Section 15.4.3 of this report.

Reactor Coolant System and Connected Systems

5.4.5 Main Steam Isolation Valve Leakage Control System

This section is not applicable to the ABWR because a main steam isolation valve leakage control system is not used. This is discussed in Section 10.3 of this report.

5.4.6 Reactor Core Isolation Cooling System

The staff evaluated the reactor core isolation cooling (RCIC) system for conformance to SRP Section 5.4.6. The staff's review criteria are based on meeting the following:

- (1) GDC 4 as related to dynamic effects associated with flow instabilities and loads.
- (2) GDC 5 as related to structures, systems, and components important to safety not being shared among nuclear power units unless it can be demonstrated that sharing will not impair its ability to perform its safety function.
- (3) GDC 29 as related to the system being designed to have an extremely high probability of performing its safety function in the event of anticipated operational occurrences.
- (4) GDC 33 as related to the system capability to provide reactor coolant makeup for protection against small breaks in the reactor coolant pressure boundary so the fuel design limits are not exceeded.
- (5) GDC 34 as related to the system design being capable of removing fission product decay heat and other residual heat from the reactor core to preclude fuel damage or reactor coolant pressure boundary overpressurization.
- (6) GDC 54 as related to piping systems penetrating primary containment being provided with leak detection and isolation capabilities.

Unlike most current BWR designs, the RCIC system in the ABWR is a part of the emergency core cooling system (ECCS). The initiation logic is diversified by adding a high drywell pressure input as well as by maintaining the typical system initiation on RPV Level 2. In the ABWR design, system reliability is improved by including a bypass line to the turbine steam inlet valve (F045 dc powered) to provide for a smoother turbine start and reduce the possibility of an overspeed trip. Since the RCIC system is a part of the ECCS, full-flow testing

capability is provided using the safety-related suction source (i.e., suppression pool).

The RCIC system is designed as a high-pressure reactor coolant makeup system that will start independent of the ac power supply. All motor-operated valves will be dc operated, except the inboard steam isolation valves. Steam supply inboard isolation valve F035 and inboard bypass valve F048 will be powered from ac power sources; however, valve F035 will normally open and fail as is; therefore, loss of ac power will not prevent RCIC system operation. Inboard bypass valve F048 will be closed during the system operation; and hence, loss of ac power will not prevent RCIC system operation. The system will provide sufficient water to the reactor vessel to cool the core and to maintain the reactor in a standby condition if the vessel becomes isolated from the main condenser and experiences a loss of feedwater flow. The system also is designed to maintain reactor water inventory, in the event of a loss of normal feedwater flow, while the vessel is depressurized to the point at which the RHR system can function in the shutdown cooling mode.

In reviewing Amendment 32 to the SSAR, the staff noted that the previously described RCIC system capability, without ac power, of "8 hours" had been changed to "at least 2 hours." The staff raised a concern to GE that changing the capability from 8 hours to 2 hours would result in a measurable increase in the core damage frequency estimate as related to station blackout. To clarify the changed position, GE stated and the staff agreed that an RCIC system capability of up to 8 hours could only be adequately demonstrated during startup tests when plant steam is available after fuel loading. SSAR Section 5.4.6 stated that the RCIC system is designed to perform its function without ac power for at least 2 hours with a capability up to 8 hours. It further stated that the COL applicant will provide analyses for the as-built facility to demonstrate the 8-hour capability. This is acceptable and resolved the staff's concern.

The RCIC system consists of a steam-driven turbine-pump unit and associated valves and piping capable of delivering makeup water to the reactor vessel through the feedwater system. The steam supply to the RCIC turbine is taken from main steamline B at the upstream side of the inboard MSIV. The steam supply to the RCIC turbine will be ensured even if MSIVs are closed. Fluid removed from the reactor vessel following a shutdown from power operation will be normally made up by the feedwater system and supplemented by inleakage from the control rod drive system. If the feedwater system is inoperable, the RCIC system will start automatically when the water level in the reactor vessel reaches the Level 2 (L2) trip set point or

will be started by the operator from the control room. The system is capable of delivering rated flow within 29 seconds of initiation. Primary water supply for the RCIC system comes from the condensate storage tank (CST), and a secondary supply comes from the suppression pool.

The RCIC system design operating parameters, as shown in SSAR Figure 5.4-9b, are consistent with expected operational modes.

Essential components of the RCIC system are designated seismic Category I (in accordance with RG 1.29, Revision 3) and QG B (in accordance with RG 1.26, Revision 3), as discussed in Section 3.2 of this report. The proposed initial test programs are discussed in Chapter 14 of this report.

The RCIC system is housed in the reactor building, which will provide protection against wind, tornados, floods, and other weather phenomena. Compliance with the requirements of GDC 2 is discussed in Section 3.8 of this report. In addition, the system is protected against pipe whip inside and outside the containment, as required by GDC 4 and as discussed in Section 3.6 of this report.

The HPCF and RCIC systems are located in different rooms of the reactor building for additional protection against common-mode failures. Different energy sources will be used for pump motivation (steam turbine for RCIC pump, electric power for HPCF pumps) and different power systems for control power. This diversity conforms to SRP Section 5.4.6.

To protect the RCIC pump from overheating, the RCIC system contains a miniflow line that will discharge into the suppression pool when the line to the reactor vessel is isolated. When sufficient flow to the vessel is achieved, a valve in the miniflow line will automatically close, thus directing all flow to the reactor.

The makeup water system connection to the RCIC system will maintain the pump discharge line in a filled condition up to the injection valve. The makeup water system operation eliminates the possibility of an RCIC pump discharging into a voided pipe and minimize waterhammer effects. A high point vent is provided, and the system will be vented periodically. The RCIC system includes a full-flow test line with water return to the suppression pool for periodic testing. The periodic tests will be performed according to ASME Code, Section XI, as required by the TSs. The staff requires, for any emergency core cooling system (ECCS), that the TSs include a system functional test at least every refueling outage, with simulated

automatic actuation and verification of proper automatic valve position to verify that the RCIC pump will develop a minimum flow of 182 m³/hr (800 gpm). This was DFSEER TS Item 5.4.6-1. GE has submitted the ABWR TS which include the functional testing; this is acceptable. Therefore, DFSEER TS Item 5.4.6-1 is resolved.

The suction piping of the RCIC system is designed for low pressure. A relief valve, therefore, is provided to protect against overpressurization of the line from the high-pressure piping. The suction pressure piping was upgraded from 1.37 MPa (200 psig) to 2.82 MPa (410 psig) for resolving the generic issue pertaining to interfacing systems loss-of-coolant-accident (ISLOCA) as discussed in Chapter 20 of this report.

Suitable provisions will be provided for isolation of the RCIC system from the RCS by one testable valve and a closed dc-powered valve in the RCIC system discharge line, and two normally open motor-operated valves with appropriate closure signals to terminate the leakage of the pipe as a result of a break outside containment. Inservice testing of pumps and valves is discussed in Section 3.9.6 of this report.

The RCIC system will have controls that will shut down the system if operating conditions exceed certain limits. A leak detection system is provided to detect leakage in the RCIC system.

The CST level transmitter will be supported and mounted in such a way that automatic suction transfer to the suppression pool from the non-seismic tank will take place without failure during a seismic event.

The RCIC system design meets RG 1.1, "Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps," Revision 0.

In the DSER (SECY-91-153), the staff identified Outstanding Issue 24, which required an evaluation of the TMI-2 Action Items as related to the ECCS. The staff's evaluation of these items (NUREG-0737 requirements that are incorporated into 10 CFR 50.34(f)(1)(v) and (1)(ix)), as related to the RCIC system, is provided in Chapter 20 of this report. Therefore, DSER Outstanding Issue 24 is resolved.

During an earlier review, the staff found that the testing of steam isolation valves (F035 and F036) leading to the RCIC turbines in currently operating BWRs did not include actual operating conditions such as a differential pressure of about 7,000 kPa (1,000 psig) and a temperature of 286 °C (546 °F) expected during a steam pipe break

Reactor Coolant System and Connected Systems

downstream of the valves. Thus, there is no verification that the isolation valves will close during a break as a result of dynamic steam flow forces. Generic Issue (GI)-87, "Failure of HPCI Steamline Without Isolation," addresses this concern. Tests performed as part of the NRC effort to resolve GI-87 have reinforced concerns about the operability of motor-operated valves (MOV) under these design-basis conditions. The staff concerns regarding operability of MOVs are given in Generic Letter 89-10 (June 1989) and its supplements. GE assumed closure of these valves in the steamline break analysis. Therefore, this functional requirement is incorporated into the ITAAC. In the DFSER the staff noted that the COL applicant referencing the ABWR design should verify test data showing the steam isolation valves (F035 and F036) will isolate under actual operating conditions of a differential pressure at about 7,000 kPa (1,000 psig) and a temperature of 286 °C (546 °F). This was DFSER COL Action Item 5.4.6-1. Since verification of the valves performance by the preoperational and power ascension testing is discussed in Chapter 14 of the SSAR, it need not be specified as a COL action item.

GE originally submitted the design description and the ITAAC for the RCIC system. At the time the DFSER was issued, the ITAAC review was in progress. Therefore, this was DFSER Open Item 5.4.6-1. GE has since provided a revised design description and ITAAC. The adequacy and acceptability of the ABWR design description and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

The RCIC system meets GDC 4, 5, 29, 33, 34, and 54 as identified in SRP Section 5.4.6. Compliance with GDC 4 on protecting the system against dynamic effects associated with flow instabilities and loads is discussed in Section 3.6 of this report. Since the ABWR is a single-unit plant, GDC 5 are not applicable. The RCIC system meets GDC 29 and 34 because it is designed to perform its function without the availability of any ac power and, in conjunction with the high-pressure core flooders system, is designed to ensure an extremely high probability of accomplishing its safety function. The RCIC system is used to supply reactor coolant makeup for small leaks. Accordingly the system meets GDC 33. Compliance with GDC 54 is discussed in Section 6.2 of this report.

The staff concludes that the design of the RCIC system conforms to the Commission's regulations and is, therefore, acceptable.

5.4.7 Residual Heat Removal System

The staff evaluated the residual heat removal (RHR) system according to SRP Section 5.4.7. The staff's review criteria are based on meeting the following:

- (1) GDC 2 with respect to the seismic design of systems, structures, and components whose failure could cause an unacceptable reduction in the capability of the residual heat removal system. Acceptability is based on meeting position C-2 of RG 1.29 or its equivalent.
- (2) GDC 4 as related to dynamic effects associated with flow instabilities and loads (e.g., water hammer).
- (3) GDC 5 which requires that any sharing among nuclear power units of structure, systems and components important to safety will not significantly impair their safety function.
- (4) GDC 19 with respect to control room requirements for normal operations and shutdown.
- (5) GDC 34 which specifies requirements for a residual heat removal system.

The RHR system consists of three independent loops - subsystems A, B, and C. Each loop contains a motor-driven pump, heat exchanger, piping, valves, instrumentation and controls. Each loop can take suction from either the RPV or the suppression pool and will be capable of discharging water to either the RPV or back to the suppression pool through a full-flow test line. Each shutdown cooling loop has its own heat exchanger, which will be cooled by the reactor building cooling water system. RHR subsystems B and C can be used for wetwell and drywell sprays. The RHR system will operate in the following modes:

- (1) shutdown cooling
- (2) suppression pool cooling
- (3) wetwell and drywell spray cooling
- (4) low-pressure flooders mode
- (5) fuel pool cooling
- (6) ac independent water addition

During all six modes of operation, the same major hardware components (e.g., RHR pump, heat exchanger) will be used. Shutdown cooling, fuel pool cooling, wetwell spray cooling, drywell spray cooling, and ac independent water addition will all be started manually. The low-pressure flood mode and the suppression pool cooling mode will be started automatically. Modes (2), (3), and (4) are reviewed in Sections 6.2 and 6.3 of this report; Mode (5) is reviewed in Section 9.1.3 of this report.

The normal operational mode of the RHR system is the shutdown cooling mode, which will be used to remove decay heat from the reactor core to achieve and maintain a cold shutdown condition. Shutdown cooling will be started manually when the RPV is depressurized to about 931 kPa (135 psig). The heat removed in the RHR heat exchangers will be transported to the ultimate heat sink by the reactor building cooling water system.

There are three suction lines directly from the RPV for shutdown cooling rather than from the external reactor recirculation loop. This design is an improvement over the present single suction line in current operating BWRs because the single line is more vulnerable to a loss of shutdown cooling by single failure of valves in the line. The RHR shutdown cooling return line for RHR subsystem A is routed to the RPV through the feedwater system. The RHR shutdown cooling return lines for RHR subsystems B and C are routed to the RPV directly.

The RHR system also includes an ac-independent water addition subsystem that consists of piping and manual valves connecting the fire protection system to the RHR pump discharge line on loop C downstream of the pump's discharge check valve. This flow path will allow water to be injected into the reactor vessel and the drywell spray during postulated beyond-design-basis conditions where all ac power and all ECCS pumps are unavailable. Additionally, an external hookup outside the reactor building is provided so that a fire truck pump can be connected as an alternative water source.

In Branch Technical Position (BTP) RSB 5-1 of the SRP, the staff recommends that the valves provided in the suction line (valves F010 and F011) of the RHR system to isolate it from the RCS have independent and diverse interlocks to protect the RHR system. Although the ABWR design does not explicitly meet the guidance on diversity, it does meet the intent of the guidance to provide high reliability against inadvertent opening of the isolation valves. The pressure signal that provides the interlock function will be supplied from 2-out-of-4 logic, which has four independent pressure sensor and transmitter inputs. The independence is provided by each being in a separate

instrument division. Furthermore, the inboard and outboard valves of the shutdown cooling system also will close on low reactor water level and will be powered from different electrical divisions. This meets the intent of BTP RSB 5-1 and, therefore, is acceptable. High-pressure/low-pressure system interaction protection functions are discussed in Section 7.6.1.3 of this report.

In the DSER (SECY-91-153), the staff noted that GE should describe in detail how the RHR system design meets the pressure relief requirements of BTP RSP 5-1. This was Outstanding Issue 22. This same issue was also identified as Outstanding Issue 31 in a follow up DSER (SECY-91-235). GE investigated potential overpressurization transients when the RHR system is not isolated from the RCS. GE also evaluated all the reactor nozzle penetrations to determine if they could be a source of high pressure into the RPV during shutdown. GE concluded that relatively cold water from the feedwater (condensate pump) system could be injected into the RPV through either of two valves in the feedwater line by inadvertent operator action or by a valve failing open. The condensate pumps are rated for approximately 49,210 L/min (13,000 gpm), which is one-third feedwater flow, with a shutoff head of approximately 4,137 kPa (600 psig). A pressure in the range of 2,758 to 3,447 kPa (400 to 500 psig) could occur upstream of the feedwater inlet valves in the RPV by inadvertent operation of the condensate pumps. This high pressure could lift the RHR pressure relief valve set at 2,275 kPa (330 psig). To provide protection against this situation, which would cause an overpressure condition in the RHR low-pressure piping, a design change was made by GE that will close all feed pump discharge and bypass valves by a reactor water level (L8) signal. The staff concludes that the pressure relief requirements of the RHR system satisfy BTP RSB 5-1 and are acceptable. This resolved DSER (SECY-91-235) Outstanding Issue 31.

Inservice testing of pumps and valves is discussed in Section 3.9.6 of this report. Relief valves are provided in each of the low-pressure lines that interconnect with the RCS to protect against overpressurization from RCS leakage.

The RHR pumps are motor-driven centrifugal pumps and are sized for the low-pressure flood mode of operation. The available net positive suction head for the RHR pumps is adequate to prevent cavitation and to ensure pump operability in accordance with RG 1.1, Revision 0. Environmental qualification for long-term operability of the RHR pumps is discussed in SSAR Appendix 3I and evaluated in Section 3.11 of this report.

Reactor Coolant System and Connected Systems

Three discharge line fill pumps are provided to maintain the RHR discharge header full of water and pressurized to reduce waterhammer during system initiation.

Each train of the RHR system can be tested during normal plant operation by pumping water from the suppression pool back into the pool. In the DFSER, the staff noted that the TS should include (1) verification of low-pressure flooder mode operability, (2) demonstration of the capability to start each pump from the control room, and (3) performance of a system functional test without requiring coolant injection into the reactor vessel. This was DFSER TS Item 5.4.7-1. GE has submitted the ABWR TS, which include the above three areas in Section 3.5.1. Hence, DFSER TS Item 5.4.7-1 is resolved.

The preoperational test program for the RHR system is discussed in Chapter 14 of this report.

The RHR system is designed to operate with or without offsite power. The RHR system will be controlled from the control room. RHR shutdown cooling subsystems A and B also can be operated from the Remote Shutdown Panel.

The staff concludes that the RHR system has the capability to bring the reactor to cold shutdown conditions in a reasonable time (within 36 hours as specified by SRP Section 5.4.7).

GE originally submitted the design description and the ITAAC for the RHR system. At the time the DFSER was issued, the ITAAC review was in progress. Therefore, this was DFSER Open Item 5.4.7-1. GE has since provided a revised design description and ITAAC. The adequacy and acceptability of the ABWR design description and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

The staff finds that the RHR system is designed to minimize the possibility of an ISLOCA in the following ways. The low-pressure systems at pump discharge directly interfacing with the RCS are designed with 3,447-kPa (500-psig) piping that provides for a rupture pressure of at least 6,895 kPa (1,000 psig). In addition, the high-/low-pressure motor-operated isolation valves have safety-grade, redundant pressure interlocks. Also the motor-operated valves will only be tested when the reactor is at low pressure. All inboard check valves on the RHR system are testable and have position indication.

As a part of the high-/low-pressure interface design review (Generic Issue 105, "Interfacing Systems LOCA at LWRs"), GE originally proposed to change the design pressure of the RHR suction piping from 1379 to 2827 kPa (200 psig to 410 psig). At the time the DFSER was issued, GE was still assessing this change and other interfaces in the design. Therefore, this was DFSER Open Item 5.4.7-2. The SSAR now describes design improvements to reduce the possibility of an ISLOCA. The adequacy and acceptability of the ABWR design with regard to the ISLOCA concerns are evaluated in Chapter 20 of this report. Therefore, this item is resolved.

RHR system vulnerabilities during shutdown and low-power operation are addressed in Section 19.4 of this report.

The RHR system meets the requirements of GDC 2, 4, 5, 19, and 34 as identified in SRP Section 5.4.7. It is designed to the seismic Category I recommendations of RG 1.29, as discussed in Section 3.2 of this report. It is housed in the reactor building for protection against the effects of flooding, tornados, hurricanes, and other natural phenomena. Compliance with GDC 2 is discussed in Section 3.2 of this report. Compliance with GDC 4 on protecting the system against dynamic effects associated with flow instabilities and loads are discussed in Section 3.6 of this report. Since the ABWR is a single-unit plant, GDC 5 are not applicable. The RHR system meets GDC 19 because its design includes necessary instrumentation and controls with respect to control room requirements. By meeting BTP RSB 5-1, the RHR system complies with GDC 34.

The staff concludes that the design of the RHR system conforms to the Commission's regulations and is, therefore, acceptable.

5.4.8 Reactor Water Cleanup System

The staff reviewed the reactor water cleanup system (CUW) description and piping and instrumentation diagrams in accordance with SRP Section 5.4.8. Staff acceptance of the design is based on compliance with the requirements of (1) GDC 1 as related to the design meeting standards commensurate with the system's safety function; (2) GDC 2 as related to the system being able to withstand the effects of natural phenomena; (3) GDC 14 as related to assuring the integrity of the RCPB; (4) GDC 60 as related to the capability of the system to control the release of radioactive effluents to the environment; and (5) GDC 61 as related to designing the system with appropriate confinement.

The CUW performs the following functions:

- (1) Removes solid and dissolved impurities from the reactor coolant.
- (2) Provides containment isolation which ensures that the major portion of the system is outside the RCPB.
- (3) Discharges excess reactor water during startup, shutdown, and hot standby conditions to the radwaste system or the main condenser.
- (4) Provides flow to the RPV head spray for rapid RPV cooldown.
- (5) Minimizes RPV temperature gradients by maintaining circulation in the bottom head of the RPV during periods when the reactor internal pumps are not operating.

The CUW is a closed-loop system comprised of a regenerative heat exchanger, two nonregenerative heat exchangers (cooled by RCW), two filter-demineralizers, two circulating pumps, system isolation valves, and other piping, valves, and instrumentation. The system takes its suction from the shutdown cooling suction line of RHR loop "B" and the RPV bottom head drain line. CUW discharges back to the vessel via either the feedwater system or the RPV head spray line. The system capacity is 2 percent of rated feedwater flow, which is twice the capacity of previous BWRs.

The majority of the system (including all safety-related components) is located in the nondivisional quadrant of the secondary containment portion of the reactor building. The system is classified as non-safety related with the exception of the containment isolation valves. System piping and components in the drywell, up to and including the outboard containment isolation valves, form part of the RCPB are classified as seismic Category I, QG A. The remainder of the system is classified as nonseismic, QG C. Low-pressure piping in the backwash and precoat area downstream of the block valves is classified as QG D. Nonsafety components receive power from non-class 1E supplies. Based on this QG and seismic classification design information, the staff concludes that the CUW system meets the guidelines of RGs 1.26 and 1.29 and, therefore, the requirements of GDC 1 and GDC 2 as they relate to the CUW design meeting the standards commensurate with the system's safety function and the system being able to withstand the effects of natural phenomena.

The two safety-related containment isolation valves on the suction line of the CUW system and the isolation valve on the RPV head spray line receive isolation signals from the leak detection system (LDS). These valves are powered by class-1E power sources and automatically isolate on indications of low reactor water level, high ambient main steam tunnel area temperature, high mass differential flow, high ambient CUW equipment area temperature, or initiation of the standby liquid control (SLC) system. The isolation valves are designed and tested to ensure that they close during maximum flow and differential pressure conditions. Based on this information, along with the seismic and quality group classifications for that portion of the system up to and including the containment isolation valves discussed earlier, the staff concludes that the system meets the requirements of GDC 14 as it relates to assuring the integrity of the RCPB.

GE performed pressure and temperature and cooldown analyses for CUW breaks inside the secondary containment. The resulting pressures and temperatures were within the environmental qualification envelope for components which would be affected by the breaks and were within the design pressure and temperature limits for the primary and secondary containment structures (see Chapter 6 of this report).

The CUW suction line contains a flow restrictor inside the containment to monitor flow and to restrict flow during a line break, as well as a motor-operated shutoff valve for long-term system isolation. In addition, upon indication of high radiation levels in the secondary containment, SGTS actuates and processes potentially radioactive effluents before release to the environment. FDU vents are routed to the backwash receiving tank while piping vents and drains are routed to the low conductivity portion of the liquid radwaste system. Unidentified leakage from CUW piping is collected and monitored in the reactor building floor drain sumps. These design features, along with the automatic system isolation features discussed earlier, ensure that releases of radioactive effluents to the environment are limited. This satisfies the requirements of GDC 60 as it relates to controlling the release of radioactive material.

The filter demineralizers are of the pressure precoat type using filter aid and powdered mixed ion-exchange resins as a filter and an ion-exchange medium. Spent resins cannot be regenerated and will be sluiced from the filter-demineralizer unit to a backwash-receiving tank from which they will be transferred to the radwaste system for processing and disposal. Resins will be discarded on the basis of filter-demineralizer performance, as indicated by monitored effluent conductivity, differential pressure across the unit, and sample analysis. A strainer is installed in the

Reactor Coolant System and Connected Systems

effluent line of each filter demineralizer to prevent resins from entering the system in the event of failure of a filter-demineralizer resin support. Each strainer and filter-demineralizer vessel has a control room alarm that will be energized by high differential pressure. If differential pressure increases beyond the alarm point, the filter demineralizer will isolate automatically. Based on this information (along with the information discussed earlier regarding system vents and drains), the staff concludes that the CUW design meets the requirements of GDC 61 as it relates to designing the system with adequate confinement features.

Hydrogen water chemistry that conforms to with the guidelines contained in Electric Power Research Institute (EPRI) Report NP-4947-SR, "BWR Hydrogen Water Chemistry Guidelines," 1987 Revision, will be used. This document also includes hydrogen water chemistry limits, responses to out-of-specification water chemistry, chemical analysis methods, data management and surveillance schemes, and management philosophy required to establish and implement a successful water chemistry control program. Therefore, the CUW system design satisfies the reactor chemistry limits specified in RG 1.56, "Maintenance of Water Purity in Boiling Water Reactors," Revision 1, and EPRI Report NP-4947-SR.

The Advisory Committee on Reactor Safeguards (ACRS) performed an independent review of the CUW system and provided the results to both GE and the NRC staff. GE reviewed the results and provided responses to each of the ACRS concerns identified in the ACRS report. The staff has reviewed GE's responses and has concluded that the modified CUW design adequately addresses the issues raised during the ACRS independent review and documented in the report.

Major concerns in the ACRS report included (1) CUW configuration and design features, (2) CUW processing capacity, (3) quality group and seismic classification of components, (4) quality and consistency of information in the SSAR, (5) design of system isolation valves, and (6) role of the system in safety analyses.

(1) CUW Configuration and Design Features

The ACRS concerns in this area included (a) the letdown path for excess water during plant startup, (b) reverse flow from the feedwater system into the main steam tunnel (MST) or secondary containment on a CUW pipe break, and (c) high-temperature isolation of the system's filter-demineralizer units (FDUs) and resin retention in the FDUs.

(a) LETDOWN PATH FOR EXCESS WATER

It was unclear whether the CUW system provided letdown to the condensate storage tank. GE clarified that CUW discharges excess water to the main condenser or to the liquid waste management system.

(b) REVERSE FLOW FROM THE FEEDWATER SYSTEM

GE modified the system design to ensure that reverse flow from the feedwater system to the secondary containment and steam tunnel would not occur as a result of a CUW line break in these areas. Previously, check valve G31-F014 and motor operated valve G31-F015 were located inside secondary containment with G31-F014 upstream of G31-F015. In this configuration, a break in the CUW line upstream of these valves, along with a failure of G31-F015 to close, could result in excessive blowdown of the CUW line into the secondary containment area. In response to this concern, GE modified the valve arrangement so that G31-F014 is now inside the MST next to the tunnel wall and G31-F015 is located against the outer MST wall. In this configuration, a CUW line break with a failure of G31-F015 will not result in excessive system blowdown into the secondary containment because check valves in the nuclear boiler system (NBS) (B21-F006A and B21-F006B) as well as G31-F014 are available to prevent reverse flow. In addition, two motor-operated valves in the NBS (B21-F007A and B21-F007B) are available to provide additional long-term manual isolation. These NBS valves are classified as safety-Class 2, Quality Group B, and seismic Category I.

The three containment isolation valves for CUW are designed to close on a signal from the Leakage Detection and Isolation System (LDS). The LDS provides this signal on low reactor water level, high ambient MST area temperature, high mass differential flow, high ambient CUW equipment area temperature, and SLCS actuation. Based on this information, the staff concludes that the modified CUW design provides assurance that excessive feedwater or CUW blowdown will not occur as a result of a CUW pipe failure inside secondary containment.

(c) HIGH-TEMPERATURE ISOLATION OF THE SYSTEM'S FILTER-DEMINERALIZER UNITS (FDUs) AND RESIN RETENTION IN THE FDUs

The ACRS report identified a concern regarding protection of the FDU resins from high temperature water. The high temperature setpoint for automatic bypass of the FDUs was not provided in the SSAR. GE later clarified that the FDUs are isolated and bypassed at 54 °C as shown on SSAR Figure 5.4-13, sheet 2.

(2) CUW Processing Capacity

The ACRS identified several concerns related to CUW flexibility and processing capacity. These included (a) increased processing capacity and (b) the effectiveness of the nonregenerative heat exchanger (NRHX) when used as an alternate means of decay heat removal.

(a) INCREASED PROCESSING CAPACITY

The ABWR design doubles the traditional CUW processing capacity from 1 percent in current BWRs to 2 percent in the ABWR. The ACRS report expressed concern regarding the impact of this increase. Specifically, the ACRS was concerned that design details such as resin bed volumes, resin types, and resin bed replacement criteria were not provided in the SSAR. Subsequently, GE clarified that the ABWR commits to maintaining the water chemistry requirements stated in the SSAR. The COL applicant will determine the proper components to meet the chemistry requirements.

(b) EFFECTIVENESS OF THE NON-REGENERATIVE HEAT EXCHANGER (NRHX) WHEN USED AS AN ALTERNATE MEANS OF DECAY HEAT REMOVAL

The ACRS report stated that additional heat exchanger parameters are needed to determine if the NRHX can be used as an effective means of removing decay heat during high pressure conditions with RHR unavailable. Subsequently, GE provided information which showed that the use of the NRHXs is not a primary means of accomplishing the heat removal function and that this method of heat removal has only a small impact on the PRA core damage frequency.

(3) Quality Group and Seismic Classification of Components

The ACRS report identified a concern relating to the conversion of Japanese safety classification codes to equivalent domestic codes. Specifically, ambiguities were identified in note 11 of SSAR Table 1.7-1 (this table correlates the Japanese and domestic codes). As a result, GE modified this table to remove any ambiguities.

(4) Quality and Consistency of Information in the SSAR

The ACRS expressed a concern regarding the quality and consistency of information not only in the SSAR section relating to CUW, but also in other SSAR sections. The staff has worked with GE throughout the review process to identify any ambiguities and inconsistencies found in the SSAR. These have largely been identified and corrected.

(5) Design of System Isolation Valves

The ACRS report identified a concern regarding the design adequacy of the system isolation valves. Specifically, current BWRs must meet Generic Letter 89-10 regarding reliable containment isolation valve closure against maximum differential pressure and flow. However, despite this, problems still exist. Demonstration of the reliability and capability of these valves through the use of in-service testing cannot be achieved since the worst-case conditions cannot be created in-situ. As a result, the assumptions used in the safety analysis and PRA may not be valid. In response, GE clarified that type-testing of all isolation valves are conducted in the shop at worst-case design conditions and each installed valve is tested in-situ. The requirements on these valves are provided in SSAR Section 3.9.6 and evaluated in Section 3.9.6 of this report. In addition, verifications are made as part of the ITAAC process.

(6) Role of the System in Safety Analyses

The ACRS report identified concerns related to system failure as it relates to the safety analysis. This included (a) maximum inventory loss and (b) isolability of the bottom head drain line.

(a) MAXIMUM INVENTORY LOSS.

The ACRS report stated that because of the uncertainty of whether the CUW line break was

Reactor Coolant System and Connected Systems

evaluated in Chapter 6 of the SSAR, it appeared that the total inventory loss from this break (assuming a 30-second CUW line isolation) may be more than that from a main steam line (MSL) break with subsequent MSIV isolation. Subsequently, GE clarified that the worst-case break outside containment (from a loss-of-mass perspective) occurred as a result of a feedwater line break. The CUW line break analysis assumed a 75-second isolation time (45 second time delay on system differential flow indication and 30 second closure time). In this conservative analysis the total inventory loss was less than that from a MSL break with subsequent MSIV isolation.

(b) ISOLABILITY OF THE BOTTOM HEAD DRAIN LINE

The original CUW design included a manually-operated valve on the reactor pressure vessel (RPV) bottom head drain line inside containment. The ACRS report identified a concern regarding isolability of this line. Subsequently, GE modified the design to make this a remote manually-operated shutoff valve and moved the valve to a position immediately upstream of the inboard containment isolation valve on the CUW suction line. In addition, the RPV bottom head drain line connects to the main CUW suction line at a "tee." The centerline of this connection is at least 389 mm (15.3 in.) above the top of active fuel. If an unisolated CUW line break should occur, the shutoff valve can be used to isolate the break. If this were to fail, the RPV water level would be maintained at the level of the "tee" connection. An isolated line break is discussed in SSAR Subsection 19.9.1.

Other concerns identified in the ACRS report related to the treatment of CUW in the PRA and the ITAAC. In both cases GE provided information which resolved the concerns.

Based on the information provided by GE in response to the concerns raised in the ACRS report, the staff concludes that the issues have been adequately addressed.

In the DFSER, the staff requested GE to provide adequate design description and inspections, tests, analyses, and acceptance criteria (ITAAC) relating to CUW. GE provided a revised set of design description and ITAAC. The adequacy and

acceptability of the design description and ITAAC are evaluated in Section 14.3 of this report.

The reactor water cleanup system (CUW) will be used to aid in maintaining the reactor water purity and to reduce the reactor water inventory as required by plant operations. The review has included piping and instrumentation diagrams and process diagrams along with descriptive information concerning the system design and operation.

The staff concludes that the proposed design of CUW is acceptable and meets the relevant requirements of General Design Criteria 1, 2, 14, 60, and 61. This conclusion is based on the following:

- (1) GE has met the requirements of GDC 1 by designing, in accordance with the guidelines of RG 1.26, the portion of the CUW extending from the reactor vessel to the outermost primary containment isolation valves to Quality Group A and by designing, in accordance with position C.2.C of RG 1.26, the remainder of the CUW outside primary containment (excluding the precoat unit) to Quality Group C.
- (2) GE has met the requirements of GDC 2 by designing, in accordance with positions C.1, C.2, C.3, and C.4 of RG 1.29, the portion of the CUW extending from the reactor vessel to the outermost primary containment isolation valves to seismic Category I.
- (3) GE has met the requirements of GDC 14 by meeting RG 1.56 in maintaining reactor water purity and material compatibility to reduce corrosion probabilities, and thus reducing the probability of RCPB failure.
- (4) GE has met the requirements of GDC 60 and 61 by designing a system containing radioactivity with confinement and by venting and collecting drainage from CUW components through closed systems.

Based on this information, the staff concludes that the CUW design for the ABWR is acceptable.

6 ENGINEERED SAFETY FEATURES

6.1 Engineered Safety Features Materials

6.1.1 Metallic Materials

The staff reviewed the engineered safety features (ESF) materials in accordance with SRP Section 6.1.1. The engineered safety features materials are acceptable if they meet the requirements of: GDC 1 and Section 50.55a of 10 CFR Part 50 as they relate to quality standards being used for design, fabrication, erection, and testing of ESF components and the identification of applicable codes and standards; GDC 4, as it relates to compatibility of ESF components with environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents; GDC 14, as it relates to design, fabrication, and testing of reactor coolant pressure boundary so as to have extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture; GDC 31, as it relates to extremely low probability of rapidly propagating fracture and gross rupture of the reactor pressure coolant boundary; GDC 35, as it relates to assurance that the clad metal-water reaction is limited to negligible amounts; GDC 41 as it relates to control of the concentration of hydrogen in the containment atmosphere following postulated accidents to assure that the containment integrity is maintained; and Appendix B to 10 CFR Part 50 as it relates to the requirement that measures should be established to control the cleaning of material and equipment in accordance with work and inspection instructions to prevent damage or deterioration.

To meet the requirements of GDC 1, 14, and 31 of Appendix A to 10 CFR Part 50, the requirement of 10 CFR 50.55a and to ensure that the ESF materials perform the necessary safety function, the ASME Code and industry standards should be satisfied for the ABWR design. SSAR Table 5.2-4 lists the pressure-retaining materials and materials specifications for the reactor coolant pressure boundary components. SSAR Table 6.1-1 lists the pressure-retaining materials and materials specifications for the primary containment system, emergency core cooling systems (ECCSs) and their auxiliary systems, and the standby liquid control system. In the DSER (SECY-91-153), the staff stated that the ESF materials should satisfy GDC 1, 14, and 31 and 10 CFR 50.55a to ensure low probability of leakage, rapidly propagating failure, and gross rupture. To do so, the ESF materials selected should satisfy Appendix I, "Design Stress Intensity Values, Allowable Stresses, Material Properties, and Design Fatigue Curves," to Section III of the ASME Code and Parts A, B, and C of Section II of the code. In its transmittal of October 9, 1991, GE addressed this item by revising SSAR Section 6.1.1.1.1 to state that the ESF

materials will satisfy Appendix I to Section III of the ASME Code and Parts A, B, and C of Section II of the code. This revision was acceptable and resolved part of Open Item 25 from the DSER (SECY-91-153).

To meet the requirements of GDC 4, 14, and 41, the water used in the ESF systems should be controlled to provide assurance that stress corrosion cracking of unstabilized austenitic stainless steel components will not occur. In the DSER (SECY-91-153), the staff made the following recommendations:

- GE should follow Electric Power Research Institute (EPRI) report, "Boiling Water Reactor (BWR) Water Chemistry Guidelines," NP-3589-SR-LD, April 1985, for the water used for ECCS and spray systems.
- GE should follow EPRI report, "Guidelines for Permanent BWR Hydrogen Water Chemistry Installations - 1987 Revision," NP-5283-SR-A, September 1987, for the design, construction, and operation of hydrogen water chemistry installations.
- GE should follow EPRI report, "BWR Hydrogen Water Chemistry Guidelines: 1987 Revision," NP-4947-SR, October 1988, for water chemistry limits, responses to out-of-specification water chemistry, chemical analysis methods, and data management and surveillance and management philosophy required to establish and implement a successful water chemistry control program.

In the DSER (SECY-91-153), the staff also stated that SSAR Sections 6.1.1.1.2 and 6.1.1.2 should have adhered to the above recommendations. This was DSER Open Item 25.

In its letter of March 11, 1992, GE responded to this item by revising SSAR Section 9.3.9 (SSAR Amendment 21) to state that the hydrogen water chemistry system will conform to the guidelines in EPRI reports NP-5283-SR-A and NP-4947-SR. This response was acceptable and resolved Open Item 25 from the DSER (SECY-91-153); however, because GE also might use pure water chemistry, the staff concluded in the DFSE that GE should commit to follow EPRI Report NP-3589-SR-LD. This was identified as DFSE Confirmatory Item 6.1.1-1. GE advised the staff in its letter of March 8, 1993, that compliance with EPRI report NP-3589-SR-LD was no longer appropriate because it had committed to use hydrogen water chemistry and was following the guidelines of EPRI report NP-4947-SR. The staff finds this commitment acceptable. This resolved DFSE Confirmatory Item 6.1.1-1.

Engineered Safety Features

SSAR Section 6.1.1.1.3 adheres to the materials specifications that satisfy Appendix I to Section III of the ASME Code and Parts A, B, and C of Section II of the code; therefore, Section 6.1.1.1.3 satisfies GDC 4, 14, and 35 and Appendix B to 10 CFR Part 50 as they pertain to materials specifications.

The controls to be placed on concentrations of leachable impurities in nonmetallic thermal insulation used on ESF components comply with the guidelines of RG 1.36, "Nonmetallic Thermal Insulation of Austenitic Stainless Steels," Revision 0. Therefore, by complying with RG 1.36, the ESF thermal insulation specifications meet the requirements of GDC 1, 14, and 31.

GE has met GDC 1, 14, and 31 and 10 CFR 50.55a with respect to ensuring an extremely low probability of leakage, of rapidly propagating failure, and of gross rupture, since the materials selected for the ABWR ESF satisfy Appendix I to Section III of the ASME Code and Parts A, B, and C of Section II of the code. During bending and fabrication, the bend radius, the material hardness, and the surface finish or ground surface are to be controlled. If the controls are not met, the materials are required to be solution heat-treated again. Fracture toughness specifications for the ferritic materials meet the requirements of the code.

The controls on the use and fabrication of the austenitic stainless steel of the systems satisfy the guidance of RG 1.31, "Control of Ferrite Content of Stainless Steel Weld Metal," Revision 3, and RG 1.44, "Control of the Use of Sensitized Stainless Steel," Revision 0. Fabrication and heat treatment practices performed in accordance with these requirements provide added assurance that the probability of stress corrosion cracking will be reduced during the postulated accident time interval.

Conformance with the code and RGs and with the staff positions mentioned above constitutes an acceptable basis for meeting the requirements of GDC 1, 4, 14, 35, and 41; Appendix B to 10 CFR Part 50; and 10 CFR 50.55a, according to which the systems are to be designed, fabricated, and erected so that they can perform their functions as required.

As discussed above, GE has met GDC 1, 14, 31, and 35 and Appendix B to 10 CFR Part 50 with respect to ensuring that the ABWR reactor coolant boundary and associated auxiliary systems will have an extremely low probability of leakage, of rapidly propagating failures, and of gross rupture.

Based on the above discussion, the staff concludes that the ESF metallic materials satisfy the requirements of GDC 1,

4, 14, 31, 35, and 41; and the requirements of 10 CFR Part 50.

6.1.2 Protective Coating Systems (Paints) and Organic Material

The staff performed its review in accordance with SRP Section 6.1.2. The protective coating systems are acceptable if the protective coatings that will be applied inside the containment meet the requirements of Appendix B to 10 CFR Part 50 as it relates to the quality assurance (QA) requirements for the design, fabrication, and construction of safety-related structures, systems, and components. The Appendix B to 10 CFR Part 50 requirements are satisfied if the protective coatings conform to the testing requirements of ANSI N101.2-1972, "Protective Coatings (Paints) for Light Water Reactor Containment Facilities," and the QA guidelines of RG 1.54, "Quality Assurance Requirements for Protective Coating Applied to Water-Cooled Nuclear Power Plants." This provides assurance that the protective coatings will not fail under design-basis-accident (DBA) conditions and generate significant quantities of solid debris that would impair the performance of the ESF.

GE stated that the coating system materials to be used on the exposed surfaces within the drywell will be qualified in accordance with ANSI N101.2-1972. The protective coating system for the containment will be applied in accordance with RG 1.54. However, GE has indicated that the COL applicant may use materials in relatively small areas in the drywell that will not conform to RG 1.54.

Since GE will meet the QA requirements of Appendix B to 10 CFR Part 50 and since the containment coating systems have been evaluated for suitability to withstand a postulated DBA environment, the staff concluded in the DFSE that the protective coating systems and their applications were acceptable.

However, the staff also concluded in the DFSE that the COL applicant referencing the ABWR design should indicate the total amount of protective coatings and organic materials used inside the containment that did not meet the requirements of ANSI N101.2-1972 and RG 1.54. This was identified as DFSE COL Action Item 6.1.2-1.

GE included this information in SSAR Section 6.1.3, which requires that the COL applicant indicate the total amount of protective coatings and organic materials used inside the containment that do not meet the requirements of ANSI N101.2 and RG 1.54. This is acceptable.

6.2 Containment Systems

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

6.2.1 Primary Containment Functional Design

The ABWR primary containment is designed with the following main features:

- A drywell that consists of two volumes: (1) an upper drywell (UD) volume surrounding the reactor pressure vessel (RPV) and housing the steam and feedwater lines and other connections of the reactor primary coolant system, safety/relief valves (SRVs), and the drywell heating, ventilation, and air conditioning (HVAC) coolers; and (2) a lower drywell (LD) volume housing the reactor internal pumps (RIPs), control rod drives, and under-vessel components and servicing equipment.

The UD is a cylindrical, steel-lined, reinforced-concrete structure with a removable steel head and a reinforced-concrete steel diaphragm floor. The cylindrical RPV pedestal, which is connected rigidly to the steel diaphragm floor, separates the LD from the wetwell. Ten UD-to-LD connecting vents (DCVs), approximately 1 m x 2 m (3.3 ft x 6.6 ft) in cross-section, are built into the RPV pedestal. The DCVs extend downward through steel pipes with an inside diameter of 1.2 m (4 ft), each of which has three horizontal vent outlets into the suppression pool.

The drywell, which has a net free volume of 7,350 m³ (259,563 ft³), is designed to withstand design pressure and temperature transients following a loss-of-coolant accident (LOCA) and also the rapid reversal in pressure when the steam in the drywell is condensed by ECCS flow during post-LOCA flooding of the RPV. A wetwell-to-drywell vacuum relief system will prevent backflooding of the suppression pool water into the LD and protect the integrity of the steel diaphragm floor slab between the drywell and wetwell and the drywell structure and liner. The drywell design pressure and

temperature are 310 kilopascal (kPa) (45 pounds per square inch gauge (psig)) and 171 °C (340 °F), respectively. The design drywell-to-wetwell differential pressures are +172.4 kPa (25 psig) and -13.8 kPa (-2 psig). The design drywell-to-reactor building negative differential pressure is -13.8 kPa (-2 psig).

- A system of drywell-to-wetwell vent channels that will blow down from the drywell and discharge into the suppression pool following a LOCA. There are 30 vents in the vertical section of the LD below the suppression pool water level, each with a nominal diameter of 0.7 m (2.3 ft). These vents are arranged in 10 circumferential columns, each containing three vents. The three-vent centerlines in each column are located at 3.5 m (11.48 ft), 4.87 m (15.98 ft), and 6.24 m (20.48 ft) below the suppression pool water level when the suppression pool is at the low water level.
- A wetwell that consists of an air volume and suppression pool, with a net free-air volume of 5,960 m³ (210,475 ft³) and a minimum pool volume of 3,580 m³ (126,427 ft³) at low water level.

The wetwell is designed for an internal pressure of 310 kPa (45 psig) and a temperature of 103.9 °C (219 °F). The design wetwell-to-reactor building negative differential pressure is -13.8 kPa (-2 psig). The suppression pool, which is located inside the wetwell annular region between the cylindrical RPV pedestal wall and the outer wall of the wetwell, is a large body of water that will serve as a heat sink for postulated transients and accidents and as a source of cooling water for the ECCS. In the case of transients that result in a loss of the ultimate heat sink, energy will be transferred to the pool by the discharge piping from the reactor system's SRVs. In the event of a LOCA in the drywell, the drywell atmosphere will be vented to the suppression pool through the system of drywell-to-wetwell vents.

This primary containment design basically uses combined features of the Mark II and Mark III designs, except that the drywell consists of UD and LD volumes. The vents to the suppression pool are a combination of the vertical Mark II and horizontal Mark III systems. The wetwell is similar to a Mark II wetwell.

6.2.1.1 LOCA Chronology

Following a postulated LOCA, the drywell pressure increases as a result of blowdown of the reactor coolant system. Pressurization of the drywell causes the water initially in the vent system to be accelerated into the pool

Engineered Safety Features

until the vents are cleared of water. During this clearing process, the water leaving the horizontal vents forms jets in the suppression pool and causes water jet impingement loads on the structures within the suppression pool and on the containment wall opposite the vents.

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

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

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

Data from PSTF air tests indicate that after the pool surface has risen approximately 1.6 times the initial submergence of the top vent (which translates to 3.66 m (12 ft) above the initial pool surface for the Mark III design) the thickness of the water ligament could be as small as 0.61 m (2 ft) or less, significantly reducing the impact loads. This phase is referred to as "incipient breakthrough," that is, the ligament begins to break up. To account for possible nonconservatism in the test facility arrangement and instrumentation error bands, the staff has determined that the breakthrough height should be set at 5.5 m (18 ft) above the initial pool surface for the Mark III design. The staff's evaluation of the breakthrough height for the ABWR design is given in Section 6.2.1.6 of this report.

As the drywell air flow through the horizontal vent system decreases and the air/water suppression pool mixture experiences gravity-induced phase separation, pool upward movement stops and the fallback process starts. During

this process, floors and other flat structures experience downward loading, and therefore, the containment wall theoretically can be subjected to a small pressure increase. However, this pressure increase has not been observed experimentally.

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

Toward the end of the reactor blowdown, the top row of vents is capable of condensing the reduced blowdown flow and the two lower rows are totally recovered. As the blowdown steam flow decreases to very low values, the water in the top row of vents starts to oscillate, causing what has become known as vent chugging. This action results in dynamic loads on the top vents and on the RPV pedestal wall opposite the upper row of vents. In addition, an oscillatory pressure loading condition can occur on the drywell and wetwell. Because this phenomenon is dependent on a low steam mass flux (the chugging threshold appears to be in the range of 49 kg/sec/m^2 (10 lb/sec/ft^2)), it is expected to occur for all break sizes. For smaller breaks, it may be the only mode of condensation that the vent system will experience.

The staff's evaluation of these LOCA-related pool dynamic loads is given in Section 6.2.1.6 of this report.

Shortly after onset of a DBA, the ECCS pumps automatically start and pump suppression pool water into the RPV. This water floods the reactor core, and the water starts to cascade into the drywell from the break. When this occurs will depend on the size and location of the break. Because the drywell is full of steam at the time of vessel flooding, the sudden introduction of cool water causes rapid steam condensation and drywell depressurization. When the drywell pressure falls below the wetwell airspace pressure, air from the wetwell redistributes between the drywell and wetwell through the wetwell-to-drywell vacuum relief system. Eventually enough air returns to equalize the drywell and wetwell pressures; however, during this drywell depressurization transient, there is a period of negative pressure on the drywell structure. A negative load condition of -13.8 kPa (-2 psig) is, therefore, specified for drywell design. The staff's evaluation of this drywell-to-wetwell negative differential pressure is given in Section 6.2.1.5.1 of this report.

6.2.1.2 Containment Analysis

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

Containment Analytical Model

In its calculation of the short-term and long-term containment pressure-temperature response to postulated high-energy line breaks, GE used the same analytical models and conservative assumptions that were presented and reviewed for the Mark III containment in General Electric Standard Safety Analysis Report (GESSAR) II (NUREG-0979). The staff found these to be acceptable, using independent confirmatory analyses with the CONTEMPT-LT28 computer code. These models and assumptions are discussed in the ABWR SSAR and NEDO-20533 and its Supplement 1, "The G.E. Mark III Pressure Suppression Containment Analytical Model." In response to the staff's request for additional information (RAI), GE stated that the analytical models described in NEDO-20533 are appropriate to calculate the ABWR containment responses to postulated accidents. Although originally written for predicting of Mark III transients, these models, which simulate the transient conditions in the containment, can be adapted for the ABWR containment configuration. These models simulate the drywell, vent systems, and wetwell (suppression pool and airspace). They are, therefore, adaptable to other containment configurations having the same basic components.

As indicated earlier, the ABWR containment design uses combined features of Mark II and Mark III designs, except for the unique feature of two drywell volumes (upper and lower). The vent system is a combination of vertical (Mark II design) and horizontal (Mark III design) drywell-to-wetwell vent systems, and the wetwell (suppression pool and airspace) is similar to a Mark II.

The staff was unable to conclude in the DSER (SECY-91-355) that the assumptions and models used to predict the containment pressure and temperature transients following a LOCA in the ABWR containment were acceptable. In the DSER (SECY-91-355), the staff identified the issue of containment pressure and

temperature analysis as Open Items 3 and 10. Specific concerns are discussed below.

As discussed in the DSER (SECY-91-355), GE had not provided a detailed discussion to describe how the two ABWR drywell volumes (Open Item 3) and the combination vertical and horizontal vent system are modeled in the computer code to represent the physical geometry of the containment, and how the air carryover from the two drywell volumes to the wetwell is treated in the computer code. The impact of any difference in the hydrodynamic force, caused by venting, between the Mark III design (vent annulus) and ABWR design (pipe vents) was also unclear. In addition, the staff required tests or analyses to verify that following a LOCA:

- the combination vertical and horizontal drywell-to-wetwell vent system will perform (to demonstrate venting clearing, condensation, and chugging) as predicted
- the containment will perform (air carryover and containment pressure and temperature responses) as predicted by the analytical model

In a meeting held on May 6, 1992, GE clarified its model assumed for drywell volumes and the vent system. The same as for the Mark III containment model, GE assumed that the UD and LD volumes act as a single node and the drywell-to-wetwell vent system as a horizontal vent for its ABWR analysis. The contents of the LD start transferring to the wetwell as soon as the pressure starts decreasing. GE assumed 50 percent of the LD contents is transferred into the wetwell. The staff raised questions about the validity of these assumptions because they do not reflect the actual ABWR configuration. By a letter dated May 22, 1992, GE provided additional information to justify the conservatism of those assumptions. GE calculated the drywell pressure using a more realistic two-node drywell model and compared it with the drywell pressure described in the SSAR using the single-node drywell model. The results show that the single-node model results in a higher peak drywell pressure. Therefore, the single-node model is acceptable in that it is more conservative. This resolved Open Item 3, which was identified in the DSER (SECY-91-355). Furthermore, because the peak containment pressure occurs substantially later, after the vent-clearing process is complete, the slight geometrical differences are expected to result in negligible effects on the containment peak pressure and overall vent-clearing process. GE's use of NEDO-20533 for the ABWR is acceptable.

Regarding the test programs, GE stated that the ABWR test program is described in SSAR Appendix 3B. The test

Engineered Safety Features

program includes 24 tests of LOCA condensation oscillation and chugging loads. These loads for the ABWR are identified to be potentially different from those for the MARK III design. GE expects other test data that were developed for the Mark III to be applicable for the ABWR because the total area ratio of the horizontal vents and vertical vents in the ABWR will be comparable with that in the Mark III. This was identified as Open Item 3 in the DSER (SECY-91-355), which is resolved. The hydrodynamic loads described in Appendix 3B are evaluated in Section 6.2.1.6 of this report. DSER Open Item 10 is dealt with in that section. The staff concludes that GE's assumptions and models for ABWR containment response analysis are acceptable.

6.2.1.3 Short-Term Pressure Response

The maximum drywell-to-wetwell differential pressure occurs during the blowdown phase (short term) of a LOCA. GE has performed analyses of various postulated primary system breaks, including a double-ended rupture of the main feedwater line, a double-ended rupture of the main steamline, and small-break accidents. The analyses show that the main feedwater line break (FWLB) yields the limiting drywell-to-wetwell differential pressure and peak drywell and wetwell pressure and is, therefore, the design-basis accident for the drywell and wetwell. The main steamline break (MSLB) yields the limiting drywell temperature. GE has provided comparative plots of drywell and wetwell short-term pressure and temperature responses to design-basis, 0.0465 m² (0.5 ft²), 0.093 m² (0.1 ft²), and 0.0093 m² (0.01 ft²) breaks in both the main feedwater and main steamline piping inside the drywell. These figures substantiate the large guillotine breaks resulting in the highest drywell and wetwell pressure and temperature. However, in the DSER (SECY-91-355), the staff stated that these figures, comparing different size pipe breaks, did not indicate the same value of peak drywell and wetwell pressure as reported in SSAR Table 6.2-1. The staff asked GE to clarify these discrepancies.

In a letter dated May 22, 1992, GE compared the reported values in Table 6.2-1 and the corresponding calculated values in SSAR Figures 6.2-6, 6.2-8, 6.2-13, and 6.2-17. It demonstrated the consistency of the reported values between the table and figures in the SSAR. This is acceptable and this item is resolved. It was not tracked in the DSER or DFSE as an open item.

SRP Section 6.2.1.1C states that, for Mark III plants at the construction permit stage, the containment design pressure should provide at least a 15-percent margin above the peak calculated containment pressure and the design differential pressure between the drywell and the containment should provide at least a 30-percent margin above the peak

calculated differential pressure. GE's calculated drywell peak pressure for the FWLB is 268 kPa (39 psig) and maximum calculated temperature is 170 °C (338 °F) resulting from the MSLB. The design pressure for the drywell is 310 kPa (45 psig), which provides a margin of 15 percent above the peak calculated pressure in the drywell and is equal to the margin recommended in the SRP. Therefore, this design margin for containment pressure is acceptable.

The calculated wetwell peak pressure and maximum temperature are 179 kPa (26 psig) and 97.2 °C (207 °F), which is 6.7 °C (12 °F) below the design temperature of 103.9 °C (219 °F) resulting from the FWLB. The design pressure for the wetwell is 310 kPa (45 psig), which provides a margin of 42 percent above the peak calculated pressure in the wetwell.

The calculated drywell-to-wetwell peak differential pressure is 110.3 kPa (16 psig) and the design drywell-to-wetwell differential pressure is 172.3 kPa (25 psig), which provides a design margin of 56 percent.

The staff concludes that the containment pressure and temperature transients following a LOCA in the ABWR containment are consistent with SRP Section 6.2.1 and, therefore, are acceptable.

6.2.1.4 Long-Term Response

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

In the long-term analysis, GE accounted for potential postaccident energy sources, including decay heat, pump heat rate, sensible heat, and metal-water reaction energy. GE's long-term model also assumed that the containment

atmosphere would be saturated and equal to the suppression pool temperature at any time. Therefore, the containment pressure is equal to the sum of the partial pressure of air and the saturation pressure of water corresponding to the pool temperature.

GE calculated a peak suppression pool temperature of 96.9 °C (206.46 °F). The calculated long-term secondary peak containment drywell and wetwell pressures are well below the calculated short-term peak pressures. GE's analysis for long-term response following a LOCA in the ABWR containment is acceptable in accordance with SRP Section 6.2.1.

6.2.1.5 Reverse Containment Pressurization

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

- inadvertent drywell/wetwell spray actuation during normal operation
- post-LOCA drywell depressurization as a result of condensation of the steam by the spilled ECCS subcooled water
- wetwell spray actuation following a stuck-open relief valve

6.2.1.5.1 Drywell Depressurization

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

- post-LOCA drywell depressurization as a result of condensation of the steam by the spilled ECCS subcooled water
- inadvertent drywell spray actuation during normal operation

GE states that drywell depressurization following an FWLB results in the most severe negative pressure transient in the drywell. Without vacuum relief, this negative pressure transient may create a drywell-to-wetwell negative pressure differential of -275.8 kPa (-40 psig). This pressure differential is much greater than the design

negative drywell-to-wetwell pressure differential of -13.8 kPa (-2 psig). Therefore, this transient is used to determine the size and the number of wetwell-to-drywell vacuum breakers.

On the basis of its analysis, GE further states that with a typical vacuum breaker diameter of 50.8 cm (20 in.), a loss coefficient, K, of 3, and one single failure, eight wetwell-to-drywell vacuum breakers are required to maintain the drywell-to-wetwell and drywell-to-reactor building negative pressure differentials below the design negative pressure differentials of -13.8 kPa (-2 psig).

In the DSER (SECY-91-355), the staff identified Open Item 4 regarding the adequacy of the vacuum breaker design. Specifically, GE needed to clarify the arrangement of vacuum breakers (e.g., LD or UD, two valves in series for bypass single-failure protection) and the test program that would demonstrate the performance of the vacuum breaker system.

In a telephone conversation on August 9, 1991, GE stated that analyses had been performed using first-principle analytical models. These analyses were similar to those performed for other BWRs and assumed that the spray efficiency was 100 percent. Subsequently, in a letter dated May 22, 1992, GE explained that there are eight vacuum breaker valves, including one valve to meet the single-failure criterion.

Each penetration opening into the LD has one valve. These 50.8-cm (20-in) swing check valves open passively on negative differential pressure and require no external power to actuate them. They will be installed horizontally in the wetwell airspace. Position locations of these valves are shown in Figures 1.2-3C and 1.2-13K in SSAR Amendment 6. Also, vacuum breakers are equipped with position switches facilitating the monitoring of valve position inside the control room. The design of the wetwell-drywell vacuum breakers is acceptable.

Concerning the testing aspect of the DSER (SECY-91-355) open item, the ABWR technical specifications (TS) must require the periodic inspection and testing of vacuum breakers during outages to ensure their operability. This was DFSER TS Item 6.2.1.5.1-1. Such testing is included in the BWR Owners Group Standard Technical Specifications, which GE is adopting for the ABWR. Since GE has included an acceptable testing requirement, this item is resolved.

DSER Open Item 4 is resolved because the design of the wetwell-drywell vacuum breakers has been found to be acceptable in accordance with SRP Section 6.2.1, and

Engineered Safety Features

because the system will be tested as specified in the TS for the ABWR.

In a letter dated May 26, 1994, GE provided the results of an analysis that evaluated the effect of drywell spray actuation following a LOCA. The results of this analysis are bounded by the post-LOCA drywell depressurization case resulting from steam condensation by the spilled ECCS subcooled water.

6.2.1.5.2 Wetwell Depressurization

Wetwell depressurization, which creates a wetwell-to-reactor building negative pressure differential, can be caused by the following events:

- inadvertent drywell and/or wetwell spray actuation during normal operation
- wetwell spray actuation following a stuck-open relief valve
- drywell and wetwell spray actuation following a LOCA

GE states that the limiting negative pressure transient in the wetwell corresponds to wetwell spray actuation following a stuck-open relief valve. The effect of relief valve discharge into the suppression pool also heats the wetwell airspace, thus increasing its pressure. When the pressure in the wetwell becomes greater than the drywell pressure of 1.7 kPa (+0.25 psig), the wetwell-to-drywell vacuum relief system allows the flow of air from the wetwell to the drywell, thereby pressurizing both drywell volumes. Wetwell pressure and temperature peak when the reactor decay heat decreases below the heat removal capability from continued pool cooling and wetwell spray. Wetwell temperature and pressure decrease, but the drywell pressure remains at its peak value. When the pressure difference between the two volumes becomes greater than the hydrostatic head of water above the top vent, air flows back into the wetwell airspace, slowing down wetwell depressurization. The pressure differential between the drywell and the wetwell is maintained constant at the hydrostatic head above the top row of horizontal vents. The final pressure in the wetwell is lower than the drywell pressure because more air is transferred to the drywell during wetwell pressurization than is received during wetwell depressurization.

Inadvertent drywell or wetwell spray actuation during normal operation can cause depressurization of the sprayed volume because of the resultant condensation of vapor present in the airspace. However, the magnitude of this depressurization is less than the post-LOCA or stuck-open

relief valve cases because of the relatively smaller mass of condensable vapor present during normal operation.

Calculation of the peak wetwell-to-reactor building negative differential pressure is based on an energy balance of the containment atmosphere before and after spray activation, assuming that the final air-vapor mixture is at 100-percent relative humidity (RH) and that there are no reactor building-to-wetwell vacuum breakers. Using these assumptions, the peak calculated wetwell-to-reactor building negative differential pressure was determined by GE analysis, assuming worse-accident conditions, to be -12.2 kPa (-1.77 psig). This is 10 percent less than the ABWR design value of -13.8 kPa (-2.0 psig). In a meeting with GE on May 6, 1992, the staff questioned the appropriateness of the assumption of 20-percent initial drywell RH in the analysis. In response to the staff's concern, GE performed a sensitivity study assuming a conservative drywell RH of 60 percent and a minimum suppression pool temperature of 23.9 °C (75 °F) for the postulated drywell and wetwell spray actuation during normal operation. The resulting maximum negative differential pressure shown in a letter dated June 10, 1992, is -10.2 kPa (-1.48 psi), which is less limiting than the case reported in the SSAR. The limiting case of wetwell spray actuation following a stuck-open relief valve is independent of the drywell humidity.

From its analysis given in the SSAR, GE concluded that vacuum relief from the reactor building to the wetwell was not required. The staff asked GE in what way was the design of the ABWR different from the designs of Mark II and III plants, which do require vacuum relief. GE stated that the wetwell spray system flow will be limited by design to a maximum flow of 1,893 L/min (500 gpm), which has the effect of limiting the negative pressure transient response to -12.2 kPa (-1.77 psig). Therefore, the staff concludes that the limiting case reported in the SSAR is acceptable. The staff also finds that the initial conditions, assumptions, and methodology used in the GE analysis are acceptable, as discussed above, and that vacuum relief from the reactor building to the wetwell is not needed.

6.2.1.6 Suppression Pool Dynamic Loads

GE submitted proprietary SSAR Appendix 3B to address the issue of suppression pool dynamic loads for the ABWR. The scope of Appendix 3B encompasses SRV actuation and LOCA phenomena, SRV discharge loads, LOCA loads, submerged structure loads, and load combinations. Although similar to the Mark III containment design, the ABWR has the following distinctive features that affect suppression pool dynamic loads: wetwell airspace pressurization, an LD volume, a

smaller number of horizontal vents (30 in the ABWR compared with 120 in the Mark III), horizontal vent extension into the pool, vent submergence, and suppression pool width.

SRV actuation and LOCAs are the events that can impose dynamic loads on the suppression pool. SRVs discharge steam from the RPV through discharge piping that is routed into the suppression pool and fitted at the suppression pool end with a quencher to enhance heat transfer between the hotter SRV discharge fluid (steam and air) and the cooler suppression pool water.

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

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

The ABWR SRV discharge is directed to the suppression pool through X-quenchers that GE has stated are identical to the quenchers used for the Mark II and Mark III designs. Therefore, GE concluded that the hydrodynamic load methodology, developed for the Mark II and III designs, was applicable for both the ABWR suppression pool geometry and the X-quencher configuration. However, in the DSER (SECY-91-355), the staff questioned how GE had addressed the SRV loads that would result from a second opening of the SRV while the SRV tailpipe is still hot from the initial SRV discharge (commonly referred to by the staff as "subsequent actuation" or "consecutive actuation" in NUREG-0802, "Safety/Relief Valve Quencher Loads: Evaluation, for BWR Mark II and III Containments") (Open Item 5). A subsequent SRV valve actuation becomes a concern because following the first actuation of the SRV, the second actuation could generate higher loads on the structure.

The water reentering the tailpipe (reflood) after the initial actuation of the SRV has been found in experiments to be transitory because the water column within the tailpipe does not reach equilibrium quickly. In addition, the tailpipe wall has not cooled to its initial temperature before the second actuation of the SRV. If the reactor system pressure should rise again to greater than the SRV set point, the SRV would discharge. However, the noncondensable gas from the drywell atmosphere that reentered the tailpipe through the vacuum breaker would have been heated by the tailpipe wall. This discharge (commonly called SRV air-clearing loads) could produce hydrodynamic wall pressures in the pool that might be significantly different from the initial air-clearing loads because of the higher noncondensable gas temperature in the tailpipe. As a result, the staff concluded that loads from both the initial and second actuation should be considered in the design of the system. GE stated in its submittal of June 1, 1992, that both the first and possible second actuation were considered and the structures will be analyzed to accommodate these second actuation loads. GE stated that these analyses are scheduled to be completed before final design approval. This item became DFSE Open Item 6.2.1.6-1.

GE stated in SSAR Section 3.9.3.3.1 that loading of the main steam and discharge piping resulting from SRV discharge was calculated on the basis of simultaneous actuation of all SRVs followed by a second actuation of all SRVs. The methodology used by GE to calculate hydrodynamic loading on SRV discharge piping resulting from initial and subsequent SRV actuations is consistent with the methodology used for earlier BWR (Mark II and III) designs. The effect of subsequent valve actuation is considered by assuming the SRV discharge pipe has been

Engineered Safety Features

heated by the initial SRV actuation. GE assumed an initial SRV pipe temperature of 149 °C (300 °F) for the drywell piping and 93 °C (199 °F) for the wetwell piping. These temperature values are based on measured data from in-plant SRV blowdown tests. This methodology, which the staff finds acceptable, is consistent with that used for earlier BWRs and resolved DFSEER Open Item 6.2.1.6-1.

The staff also questioned, in the DSER (SECY-91-355), the acceptability of the complete elimination of suppression pool temperature limits. GE had stated in its earlier submittal that suppression pool temperature limits for steady-state steam condensation were no longer needed. In the basis for this conclusion, GE referenced an analysis submitted to the staff by the BWR Owners Group. However, the staff had not yet completed its review of the BWR Owners Group request for the elimination of suppression pool temperature limits. This was Open Item 6 of the DSER (SECY-91-355). In addition, the date for completing the staff's review was shown to be very close to the date necessary for final closure of all issues for the ABWR. In light of these uncertainties, GE documented in its submittal of June 1, 1992, that the same suppression pool temperature limits will be used for the ABWR as those used in current Mark I, II, and III plants to ensure steady condensation of the SRV discharge. These criteria are documented in NUREG-0783, "Suppression Pool Temperature Limits for BWR Containments."

As currently implemented in the Mark I, II, and III designs, the suppression pool temperature limits involve a three-tier approach. The lowest temperature threshold requires the operator to take such actions as activating pool cooling to reduce the suppression pool temperature. The plant, however, can continue to operate at power during this time. The intent of this threshold is to ensure that the operator takes action to reduce pool temperature. This temperature is typically 35 °C (95 °F). Operation can continue until the suppression pool reaches 43 °C (110 °F). At this temperature, an automatic scram on high suppression pool temperature occurs. Finally, if the pool reaches 49 °C (120 °F), the TS require depressurization of the reactor coolant system and initiation of cold shutdown conditions. This was DFSEER TS Item 6.2.1.6-1. GE included this item in its TS, which the staff has found are acceptable in Chapter 16 of this report, thus resolving this TS item.

This process ensures that suppression pool temperature limits for reactor scram and reactor depressurization, as defined in NUREG-0783, will not be reached. The staff finds acceptable GE's commitment to maintain current suppression pool temperature limits. This part of Open Item 6 in the DSER (SECY-91-355) is resolved.

Another part of Open Item 6 from the DSER (SECY-91-355) was associated with the justification used to establish the pool dynamic loads for the ABWR. The ABWR suppression pool contains structures above and below the normal level of the pool (such as SRV tailpipes, access tunnels to the LD, and a walkway in the suppression pool) that would be subject to pool drag and pool swell impact loads during the initial vent clearing and pool swell phenomena. The staff requested that GE provide the specific tests used to support the methodology to establish the hydrodynamic loads for the ABWR. GE committed to provide these tests in the SSAR. GE had identified the general database that will be used and the methodology to develop the hydrodynamic loads, but had not submitted data on the actual primary or secondary loads (wall pressure, thermal, drag, impact, etc.). This methodology for defining hydrodynamic loads was considered acceptable in the DFSEER. This resolved both parts of Open Item 6 identified in the DSER (SECY-91-355). This resulted in DFSEER Confirmatory Item 6.2.1.6-1 - GE would provide the specific load definition design information in a future SSAR amendment.

GE provided a typical pressure time history at the bottom of the pool boundary in SSAR Figure 3B-29. The staff finds this plot to be an acceptable hydrodynamic load definition for design certification. This resolved DFSEER Confirmatory Item 6.2.1.6-1.

The ABWR configuration is similar to the Mark II containment design in that the suppression pool volume of the ABWR wetwell is approximately equal to the Mark II and III suppression pool volumes. Also, the sizes of the drywell volumes are quite similar. GE stated in its submittal of June 1, 1992, that the ABWR wetwell airspace volume is 5,947 m³ (210,000 ft³) as compared with a Mark II wetwell volume of 4,672 m³ (165,000 ft³). Therefore, it considered the methodology used to evaluate the bulk backpressure due to suppression pool gas space compression from pool swell for the Mark II design to be equally applicable for the ABWR.

The PICSM computer code, which models the transient behavior of suppression pool swell surface elevation, pressure in the wetwell airspace, and pool surface velocity, is described in GE technical report NEDE-21544 (proprietary). GE validated test data generated for the Mark II design; however, the code was not reviewed and approved by the staff.

The staff identified Open Item 10 in the DSER (SECY-91-355), related to modeling assumptions used by GE. For ABWR applications, the PICSM code was used to compare Mark III suppression pool swell test data from the pressure suppression test facility (PSTF) with analytical

predictions from PICSIM. The parameters that were compared were gas backpressure, water slug velocity, and water slug swell height. GE concluded that the PICSIM code was technically adequate to model the suppression pool gas space pressure; however, the calculated water slug surface elevation required some modeling adjustments because of the horizontal vent configuration of the Mark III. A horizontal vent configuration introduces an air bubble into the suppression pool that does not spread uniformly across the entire suppression pool; therefore, it produces a swell that is not radially uniform. The pool swell has a higher rise on the inside radius of the pool.

The Mark III vent design is quite similar to that of the ABWR. Because of this similarity, GE felt that comparison with the Mark III data would be sufficient to validate the program for use on the ABWR design. This type of comparison also was necessary to determine if the difference in vent design (such as submergence of the vent and number of vents) would significantly affect the calculated results using the PICSIM code.

Because of the horizontal vents for both the ABWR and the Mark III, the air bubble does not penetrate the entire width of the pool as was described above. As a result, the suppression pool water slug is not a constant thickness. The PICSIM code does not implicitly model this nonuniform pool swell elevation. This nonuniform pool swell is different from the Mark II design in which the air bubble is uniformly distributed over the entire suppression pool. The vertical Mark II downcomers then distribute the air bubble throughout the entire pool cross-section. As a result, the air is injected into the pool in an almost uniform manner yielding an almost constant water slug thickness.

Because of the difference in vent configuration between the Mark II (which the PICSIM code would model accurately) and the Mark III, GE found it necessary to develop a correlation for the PICSIM code to account for the uneven pool slug rise observed in the Mark III PSTF tests. GE stated that by modeling 80 percent of the suppression pool horizontal surface area, it achieved agreement between PICSIM calculated pool swell velocity and elevation and the PSTF test results. GE included this comparison in its submittal of June 1, 1992.

GE showed that the PICSIM code without the correction factor can correctly predict the suppression pool airspace pressure time histories resulting from a reactor system blowdown in the drywell. By using a reduction factor of 80 percent for pool surface area, GE correctly predicted the swell elevation. On the basis of these comparisons, GE believes that the PICSIM code has been validated for use on the ABWR.

GE also noted that the pool swell phenomenon is dependent on a reactor system blowdown into the drywell and that the largest postulated pipe break in the ABWR is a feedwater line break (FWLB). The Mark II and III designs must accommodate a recirculation line break that GE has eliminated in the design of the ABWR. A postulated FWLB into the ABWR drywell is calculated to be a lower pressurization event, causing less blowdown flow, because of the smaller pipe break size and energy input than was considered for Mark II and III designs. Because of the lower mass and energy input into the drywell from an FWLB, the suppression pool response is expected to be less severe.

GE also stated that the pool-to-vent area ratio is larger for the ABWR than for the Mark II and III designs. For the ABWR, the pool-to-vent area ratio is 38.5, for Mark II the ratio is typically 20.0, and for Mark III it is typically 12.0. GE believes that the larger pool relative to the vent area will cause the pool hydrodynamic loads to be reduced. This position is supported by NUREG-0808, "Mark II LOCA-Related Hydrodynamic Load Definition."

The staff documented its evaluation of definition of the Mark II design containment hydrodynamic load in NUREG-0808. In the evaluation of the pool swell phenomena (discussed in Section 2.1 of the NUREG report), the staff relied on comparisons with a substantial amount of data from tests conducted by both GE and Japan Atomic Energy Research Institute. These tests were directly applicable to the Mark II design. The computer code used for these comparisons was a GE-developed program called PSAM (GE topical report NEDO-21061, Revision 0, November 1975). It was used as part of the Mark II hydrodynamic load evaluation program. The staff has reviewed the Mark II program and approved the methodology and PSAM in NUREG-0808. However, it did not find GE's methodology within PSAM acceptable. Rather, the staff based its acceptance on the favorable comparisons with the database. As a result, the use of the program for configurations other than those encompassed by the test data would not be accepted without further comparisons with applicable test data.

The PICSIM code, which GE referenced in its submittal of June 1, 1992, had not been reviewed by the staff. In addition, the use of the 80-percent area reduction factor may significantly alter the results of the program. GE's justification for use of an 80-percent reduction on pool surface area to achieve correct swell heights is based on the comparison with PSTF test heights for a Mark III-type suppression pool. However, GE did not discuss other such critical parameters of pool swell as water slug velocity and the way velocity is affected by the 80-percent reduction factor. In addition, the 80-percent reduction factor also

Engineered Safety Features

may affect such factors as the effect of gas backpressure on the water slug because of the reduced water slug surface area exposed to the gas backpressure and inertia of the water slug may be affected by the reduced surface area. GE did not discuss these effects or the PICS code's treatment of them in its submittal. PICS is a newer version of the PSAM computer code; however, GE has not indicated the specific differences between the codes.

The staff agrees that the ABWR suppression pool design is similar to the Mark III design in terms of the water slug shape and that PSTF test data for Mark III may have limited applicability to the ABWR. This closes out Open Item 18 identified in the DSER (SECY-91-355). However, since the database discussed in the submittal of June 1, 1992, did not contain specific test results for an ABWR geometry (actual vent size and configuration were not used for ABWR-specific tests), the staff evaluated the uncertainties of the test results and whether these uncertainties can be tolerated in the ABWR design.

As a result, the staff evaluated the ABWR design and the PICS load predictions to determine the degree of margin associated with each safety-related component. At a height of 7 m (23 ft) above the normal pool surface level, there are eight drywell-to-wetwell vacuum breakers that are mounted on the suppression pool wall and potentially subject to pool swell loads. The design indicates 7 m (23 ft) from the normal liquid level in the suppression pool to the bottom of the valve. The vents are equally spaced around the circumference of the suppression pool. Underneath the vacuum breakers, a continuous catwalk surrounds the pedestal wall within the suppression pool at the 6-m (20-ft) elevation, which is approximately 1 m (3 ft) below the vacuum breakers. The flooring is grating except for the area immediately below each vacuum breaker. GE states that a solid steel plate in this area will protect the valves from any direct effect of the pool swell water slug.

PICS predicts liquid impact up to 6 m (21 ft) and froth impact beyond 6 m (21 ft). As a result, the design is based on the solid decking of the catwalk absorbing 100 percent of any possible liquid pool swell load. Additionally, the valves are designed to withstand froth loads assuming no load reduction as a result of the steel plate. This approach seems to be bounding in nature, and the loadings assessed for the valves are acceptable. However, GE has not specified the method for calculating the load to which the steel plate is designed or the specific load to which the plate will be subjected. This was DSER (SECY-91-355) Open Item 6.2.1.6-2, which was not tracked in the DFSE. The staff also stated in the DFSE that Tier 1 design information and inspections, tests,

analyses, and acceptance criteria (ITAAC) were required for pool swell impact loads. (The ABWR ITAAC are discussed in Chapter 14 of this report.) The staff has determined that the design of the plate and supporting analysis are not needed for design certification but can be provided in the COL application, because of GE's commitment to have the analysis performed in accordance with the Mark II and III design methodology.

As specified in SSAR Section 6.2.7.4, the COL applicant will review the issue of providing appropriate structural features for protecting these vacuum breakers from pool swell loads and propose to the NRC staff an appropriate design for ensuring that these valves are adequately protected. The structural shielding features for pool swell loads will be determined on the basis of the methodology approved for the Mark II and III designs. For design of structural shielding features, pool swell loads will be defined to the maximum practical extent possible. This commitment to a COL action item is acceptable, and resolved DSER Open Item 6.2.1.6-2.

In the DSER (SECY-91-355), the staff raised an issue concerning the modeling assumptions used for calculating suppression pool boundary loads during the pool swell period occurring as a result of a postulated hydrodynamic event caused by a LOCA. The question was raised concerning the modeling assumptions used for conducting the tests to determine the resultant pool loads.

GE stated in its submittal of June 1, 1992, that it conducted a test program for suppression pool hydrodynamic loads specific to the ABWR because of anticipated differences in condensation oscillation (CO) and chugging (CH) loads in the ABWR design when compared to the Mark III design. GE anticipated differences in loads because the airspace ($5,947 \text{ m}^3$ ($210,000 \text{ ft}^3$)) in the ABWR suppression pool is much smaller than that in the Mark III design ($4,672 \text{ m}^3$ ($1,140,000 \text{ ft}^3$)). This smaller wetwell airspace volume increases the pool backpressure and may dampen pool response. Also, the LD in the ABWR may have an effect in that the total drywell volume of the ABWR is $7,334 \text{ m}^3$ ($259,000 \text{ ft}^3$) compared to $7,787 \text{ m}^3$ ($275,000 \text{ ft}^3$) for Mark III. However, the LD is a physically separate volume from the UD and is connected to the UD through the vent duct system compared with Mark III, where the drywell is one large volume. The ABWR also has a reduced number of horizontal vents (30) at a deeper submergence of 3.5 m (11.48 ft) from the top vent relative to the Mark III, which has 120 vents at a submergence of 2 m (7 ft). The suppression pool for the ABWR is also wider (7.5 m (24.6 ft)) than that in the Mark III (6.2 m (20.5 ft)).

The staff identified Open Item 9 in the DSER (SECY-91-355) concerning the scaling laws used by GE for developing the ABWR load definition. GE conducted 24 simulated blowdown tests on a single-cell model (a 36-degree sector of the ABWR suppression pool) that represented a single vent pipe system. The test facility used was scaled for two sizes of blowdown experiments. The subscale (SS) tests represented the ABWR scaled by a reduction factor of 2.5 and a scaled single vertical and single horizontal vent pipe system. The partial full-scale (FS) tests were conducted with the same scale factor for pool dimensions but a full-scale vertical and horizontal vent system and two horizontal vent pipes into the pool. The ABWR has three horizontal vent pipes into the suppression pool.

Using the SS and FS configuration in its test facility, GE ran 24 blowdown tests to obtain data on CO and CH loads to produce wall pressure data to be used to generate the load definition. Thirteen tests were conducted for CO loads on the SS test configuration and eleven were conducted for CH loads on the FS test configuration. For the SS tests, GE stated that linear dimensions were scaled, but thermodynamic properties associated with steam condensation such as pressure, temperature, and enthalpy were maintained at full-scale values.

The staff evaluated the above approach and concludes that these additional tests have added to the overall database relative to pool condensation loads and are acceptable. This resolved Open Items 7, 8, and 9, as identified in the DSER (SECY-91-355). However, GE had not addressed the use in the ABWR design of Mark III data from the PSTF blowdown tests that were conducted to verify the GE methodology used for the Mark III load definition studies. These tests were conducted and demonstrated the use of scaled tests, as reported in NUREG-0978, which showed close agreement between 1/3 and 1/9 scaled tests. The PSTF tests for Mark III on all scaled tests reported in NUREG-0978 were conducted with full-scale vent lengths and all three horizontal vents.

GE also had not demonstrated that the past condensation tests can or should be neglected in the development of the load definition for the ABWR. In addition, the staff was concerned about the adequacy of using only the SS and FS tests for which one and two horizontal vent pipes into the pool were used instead of the three vent pipes used in the ABWR and the PSTF tests for Mark III.

For previous tests conducted for Mark III at the PSTF, three horizontal vent pipes that were full scale were used. For the ABWR SS tests, the vent system was scaled, which the staff believes will affect the measured frequency spectrum. The use of scaled frequency responses for CO

has not been demonstrated, nor has there been any attempt to justify the approach on the ABWR docket. The Mark III tests for unstable CO were conducted on a full-size vent system, which would not interfere with the frequency content of the measured wall pressures in the test facility.

Therefore, to resolve DSER Open Item 10, the staff concluded that GE should address the differences in load definition using the Mark III and II test databases and determine the effect on structural response of possible differences from frequency signatures for both CO and CH loads. The use of the ABWR SS and FS tests without correlation to the Mark III test database had not been demonstrated to be sufficient for modeling without the full-scale vent pipe configuration and was unacceptable. This was DSER Open Item 6.2.1.6-3.

SSAR Figure 3B-22 shows a pressure time history representative of CO loads associated with the ABWR using the source load approach. SSAR Figure 3B-23 shows a pressure time history representative of Mark III CO loads. This pressure time history is based on the Mark III CO correlation as described in the GESSAR II (NUREG-0979). This comparison showed higher pressure amplitudes in the ABWR that were attributed to (1) a greater submergence depth (3.5 m (11.5 ft) compared with 2.3 m (7.5 ft)), (2) all three horizontal vents remaining open in the ABWR tests whereas the bottom two vents were closed in the Mark III tests at the onset of CO conditions, and (3) the contribution from increasing wetwell overpressure during the CO period in the ABWR tests. ABWR CO loads were not compared to Mark II test data because Mark II has a vertical vent while the ABWR has a horizontal vent. GE stated that a typical large chug from the full-scale database as shown in Figure 3B-24 exhibited characteristic features similar to chugging data from Mark II and Mark III testing. For the ABWR, GE defined the chugging load by using the key-chug approach, which was the same approach as that used for Mark II. GE also stated that the data supported the understanding (observed from prior tests) that chugging has some dependence on system parameters, such as mass flux and pool temperature, along with a substantial degree of randomness.

The staff believes that GE has adequately considered the differences between the Mark II, Mark III, and ABWR databases to determine that the suppression pool wall pressures for ABWR do not exhibit any unusual characteristics when compared to the Mark III wall pressures. The SS and FS tests appear to be adequate representations of the ABWR downcomer vents for predicting the suppression pool hydrodynamic response for unstable CO and CH loads. The staff finds that the

Engineered Safety Features

proposed load definition methodology for unstable CO and CH loads acceptable. This resolved DFSER Open Item 6.2.1.6-3.

6.2.1.7 Subcompartment Pressure Analysis

Internal structures in the drywell, wetwell, and secondary containment form subcompartments or restricted volumes that are subjected to differential pressure after postulated pipe ruptures. In the drywell there are two such volumes: (1) the RPV annulus, which is the annular region formed by the RPV and the reactor shield wall, and (2) the drywell head, which is a cavity surrounding the RPV head.

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

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

The staff evaluated the aforementioned information in accordance with the requirements and guidance in RG 1.70 (Rev. 3) and SRP Section 6.2.1.2. The staff requested in Open Item 11 in the DSER (SECY-91-355) the following additional information:

- mass and energy release rates assumed for the subcompartment analyses
- methodology (i.e., computer codes), if any, used in calculating subcompartment pressurization
- nodalization sensitivity studies for the individual subcompartments to justify the final model

- basis for selecting subcompartment initial thermodynamic conditions
- individual subcompartment design pressure differential

Within the limitations of the available information, the staff made the following additional observations in Open Item 12 in the DSER (SECY-91-355).

- The selected subcompartment initial humidity specified in SRP Section 6.2.1.2, Item II.B.1, is 0 percent. Because of the capability of water vapor to absorb more energy than dry air, a humidity level of 0 percent results in a maximum peak differential pressure during a high-energy line break in a subcompartment. In its analysis, GE used a higher value for initial humidity.
- On the basis of subcompartment volume and relief vent properties, the trend of calculated peak differential pressure for rooms with the same pipe break was analyzed. A number of calculated subcompartment peak pressures did not follow the basic trend that was expected (that is, for the same pipe break, peak pressure should increase with smaller room volume and/or smaller vent area). The subcompartments with questionable peak pressures were SA7, SA4, SR5, SR4, and SR9.
- Using the COMPARE MODE 1A computer code, subcompartment and vent properties from SSAR Tables 6.2-3 and 6.2-4 and main steamline break mass and energy release data from SSAR Figures 6.2-24 and 6.2-25, the staff performed a review calculation for the pressurization of rooms SS1 and ST1 (steam tunnel and turbine building). This analysis showed pressures for the steam tunnel and for the turbine building that were significantly different from those reported in the SSAR.

These observations showed inconsistencies in subcompartment peak pressure trends, subcompartment pressures, and analytical assumptions. These differences, when considered collectively, may result in a less conservative structural design of containment subcompartments. The staff identified the above concerns as Open Items 11 and 12 in the DSER (SECY-91-355).

In a letter dated May 22, 1992, GE stated that its proprietary computer code, SubCompartment Analysis Method (SCAM), was used for the subcompartment analyses. GE stated that the SCAM code was used for the Mark III standard plant and was benchmarked against 13 NRC-specified subcompartment standard problems. GE has not submitted SCAM to the NRC for computer code review and approval. SCAM calculates the transient thermal-hydraulic response of connected subcompartments

to high-energy line breaks. Each volume is modeled as a homogeneous mixture with connecting flow paths using a self-choking compressible flow model and the volume pressure and temperature calculated for each time step. The vent flow model was extensively verified by experimental comparison in GE report NEDO-20533. SCAM treats air as an ideal gas and initializes the contents of each volume as a homogeneous mixture of air and water vapor. SCAM does not include gravitational potential energy in its flow energy equation and does not include the effects of heat transfer between materials in the volumes and the flowing fluid.

By a letter dated May 11, 1992, GE provided additional information on mass and energy release rates assumed for its subcompartment analyses. GE calculated the mass and energy release rate for main steam and RWCU line breaks by including friction losses in the pipe segment between the RPV and the postulated break location as well as flow choking (i.e., critical flow) and inventory depletion effects. Isolation valve closure determines the time for cessation of break flow. Friction losses included pipe length, fittings, elbows, valves, and other equipment with a hydraulic resistance in the pipe segment. This methodology resulted in different mass and energy release rates being calculated for the same pipe type and size break in different subcompartments. GE indicated in a telephone conversation on May 19, 1992, that the inconsistency identified in Open Item 12 of the DSER (SECY-91-153), may be explained by the use of these mass and energy release data. The pipe break mass and energy release data used by GE are based on estimated pipe lengths, fittings, elbows, valves, and other hydraulic resistance components in the pipeline. This assumed hydraulic resistance (i.e., f/D) constitutes part of the design basis for the ABWR and must be confirmed and adhered to in the actual as-built plant, since deviations in such parameters as pipe routing and numbers and types of valves could affect the results of the subcompartment analysis. The staff indicated in the DFSE that this information should be included in the ITAAC for the ABWR containment. This was identified as a part of DFSE Open Item 6.2.1.7-1. On further evaluation, the staff revised its previous position and determined that the information on f/D should be included in the SSAR rather than the ITAAC. In Amendment 30, GE provided this information (f/D) in Table 6.2-4A and mass and energy release data in Table 6.2.4B of the SSAR. This is acceptable. This part of DFSE Open Item 6.2.1.7-1 is resolved. In addition, the staff and GE agreed that subcompartment analyses using as-built f/D data and effects on the mass and energy releases should be included in the ITAAC. This is discussed below.

The GE SCAM subcompartment model for the ABWR consisted of a total of 23 volumes and 35 flow junctions.

Most of the flow junctions are blowout panels that have a blowout pressure of 1.45 kPa (0.5 psig). Three different types of double-ended pipe breaks were analyzed in the subcompartment analysis: 71.1 cm (28 in.) main steam, 15.2 cm (6 in.) main steam, and 20.3 cm (8 in.) RWCU. All blowdowns and corresponding peak subcompartment pressures occurred over a period of less than 1 minute. All subcompartments were initialized at maximum expected air temperature and pressure and a humidity of 10 percent. Junctions were modeled with their flow area, forward and reverse loss coefficient and associated inertia based on flow length and area. The SCAM analyses did not include any nodalization sensitivity studies and relied on internal time step selection routines.

As part of the review of the ABWR subcompartment analyses, the staff performed independent audit calculations with the COMPARE MOD1A computer code, which is a subcompartment analysis code developed by Los Alamos National Laboratory for the NRC and approved for this application in SRP Section 6.2.1.2 (Item III). The staff developed a model of the ABWR subcompartment volumes, junction vents, and mass and energy release data based on the information provided by GE. Five of the pipe breaks were analyzed using COMPARE. These five pipe breaks resulted in peak pressures for 8 of the 23 subcompartments in the ABWR. The results of the COMPARE calculations of peak differential pressure were similar in magnitude and trend to those reported by GE. In all cases, the GE-calculated peak differential pressure was larger than that calculated by COMPARE. This is attributed to differences in modeling blowout panels and vent flow correlations between SCAM and COMPARE. However, the SCAM results were all bounding. Based on the COMPARE audit calculations, the staff confirmed in the DFSE that the results of the GE subcompartment pressure analyses were acceptable.

Review of the ABWR subcompartment analysis shows that the analysis was performed in accordance with SRP Section 6.2.1.2. Independent check calculations with the COMPARE MOD1A computer code confirmed both the trends and magnitude of the peak subcompartment differential pressures reported by GE in the SSAR.

Although SRP Section 6.2.1.2 states that zero initial humidity should be assumed for volumes, GE assumed a 10-percent humidity. GE stated that the subcompartment initial thermodynamic conditions were those corresponding to plant normal operating conditions, as defined in SSAR Appendix 3I. The staff's COMPARE analyses confirm that the effect of the 10-percent initial humidity is insignificant.

Engineered Safety Features

Item II.B.2 of SRP Section 6.2.1.2 requires that an appropriate nodalization scheme be chosen and confirmed by sensitivity studies so that there is no substantial pressure gradient within a node. In a meeting of May 6, 1992, GE stated that on the basis of its judgment of adequacy, no specific nodalization sensitivity studies were performed.

Furthermore, in a conference call on May 19, 1992, and in a letter dated May 22, 1992, GE committed to revise the SSAR to include the values of design differential pressure for each subcompartment in order to confirm that the margin between calculated and design peak differential pressure of 40 percent required by SRP Section 6.2.1.2 (Item II.B.5) is achieved for the ABWR. The staff found this 40-percent design margin acceptable subject to the SSAR revision to reflect this commitment. This was identified as Confirmatory Item 6.2.1.7-1 in the DFSER. In Amendment 30, GE provided this information in SSAR Table 6.2-3. There are a few exceptions in which the 40-percent margin cannot be satisfied. However, there still exist substantial margins such as 28 percent, 34 percent, or 38 percent shown in Table 6.2-3. The staff has determined that these few exceptions are acceptable provided the subcompartment analyses using as-built data are included in the ITAAC. Therefore, the staff concludes that DFSER Confirmatory Item 6.2.1.7-1 is resolved. In addition, DSER Open Items 11 and 12 (SECY-91-153) were resolved in the DFSER.

Following the issuance of the DSER (SECY-91-355), GE modified the reactor shield wall, extending it to a height of 0.1 m (0.33 ft) below the containment top slab. This results in a smaller vent area for a pipe break in the shield annulus. The staff requested GE to confirm the adequacy of the revised shield wall design. In Amendment 30, GE states that the results of its subcompartment pressurization re-analysis, accounting for the reduced vent area and redefined DBA break in the annulus, demonstrate the adequacy of the shield wall structure and the RPV and its internal structures. Therefore, this new issue of annulus pressurization resulting from the modification of the shield wall is resolved.

Therefore, on the basis of the above discussion, the staff concludes that GE's subcompartment analysis is acceptable.

GE submitted the design description and ITAAC for the ABWR containment. The ITAAC were being reviewed by the staff when the DFSER was issued. This was identified as DFSER Open Item 6.2.1.7-1. GE subsequently submitted a revised design description and ITAAC. In ITAAC Section 2.14.1, "Primary Containment System," GE has included containment pressure and temperature analyses of the DBA using as-built data. In addition, the

verification of the subcompartment analysis of the reactor building is included in ITAAC Section 2.15.10, "Reactor Building," as part of the structural analysis reconciling as-built data, which is described in Section 3H.5.3 of the SSAR. The staff finds the above commitments regarding subcompartment analyses acceptable. Therefore, DFSER Open Item 6.2.1.7-1 is resolved. The adequacy and acceptability of the design description and the ITAAC are evaluated in Section 14.3 of this report.

6.2.1.8 Steam Bypass of the Suppression Pool

The staff evaluated GE's analysis of steam bypass of the suppression pool provided in SSAR Section 6.2.1.1.5 in accordance with SRP Section 6.2.1.1.C. Containment pressurization from steam bypass of the suppression pool is strongly affected by the containment size and wetwell-to-drywell volume ratio. Of the Mark I, II, and III containments, the ABWR containment design is most similar to the Mark II in size and wetwell-to-drywell volume ratio. Therefore, the staff concludes that the acceptance criteria in SRP Section 6.2.1.1.C for Mark II containments are the most applicable. These acceptance criteria specify (1) an effective steam bypass capability for small breaks ($A/K^{1/2}$) of 46.5 cm^2 (0.05 ft^2) (2) automatic initiation of wetwell sprays, (3) preoperational high-pressure leak test, (4) periodic low-pressure leak tests, (5) periodic visual inspection of the vacuum relief system, (6) redundant position indicators and alarms on all vacuum breakers, and (7) monthly operability test of vacuum breakers.

In a pressure suppression-type containment, such as the ABWR, steam released from the primary system following a postulated LOCA is collected in the drywell volume and directed through the drywell connecting vents to the suppression pool where it is condensed. Therefore, no steam is supposed to enter the wetwell air volume. However, the potential exists for steam to bypass the suppression pool by leakage through the vacuum breakers or directly from leak paths in the drywell-to-wetwell boundary. Such bypass could lead to undesirable pressurization of the containment, as the pressure suppression function has been lost. BWRs have the capability to accommodate limited amounts of suppression pool bypass. An allowable amount of suppression pool bypass can be defined as the amount that will not result in the containment pressure exceeding the containment design pressure.

6.2.1.8.1 Analyses

GE performed the analyses given in SSAR Sections 6.2.1.1.5.3 and 6.2.1.1.5.4 to investigate the sensitivity of the containment design to suppression pool

bypass. In general, slow reactor depressurization events lead to more limiting suppression pool bypass cases. This is the case because the horizontal vents in the suppression pool remain covered and all of the steam released from the reactor passes through the bypass pathway into the wetwell airspace.

For the first analysis, GE assumed a controlled reactor cooldown rate of 55.6 °C (100 °F) per hour for approximately 6 hours. The assumptions were conservative, and credit was not taken for structural heat sinks or actuation of the containment sprays. The analysis indicates a maximum allowable effective leakage path ($A/K^{1/2}$) of 5 cm² (0.0054 ft²).

For the second analysis, GE also assumed a controlled reactor cooldown rate of 55.6 °C (100 °F) per hour for approximately 6 hours and credited operation of the wetwell portion of containment sprays with a flow rate of 114 m³/hr (4,026 ft³/hr) and heat transfer to the containment boundary surfaces. A spectrum of postulated reactor coolant system pipe breaks were evaluated to determine the most limiting case. The containment sprays were assumed to initiate 30 minutes after the containment reached a pressure of 103.0 kPa G (14.9 psig). This pressure is above the emergency operating procedure spray actuation pressure as specified in SSAR Section 18A. This analysis is similar to that performed for the Mark II containments. The analysis indicates a maximum allowable effective leakage path ($A/K^{1/2}$) of 50 cm² (0.054 ft²).

The staff concludes that sufficient capability exists in the containment to handle limited amounts of steam bypass of the suppression pool.

6.2.1.8.2 Wetwell Sprays for Mitigation of Steam Bypass

As discussed in Section 6.2.1.8.1 above, GE assumed a wetwell spray flow of 114 m³/hr (4,026 ft³/hr) to mitigate the consequences of steam bypass. Wetwell spray cannot be operated in isolation. Combined wetwell and drywell sprays are the normal mode of operation. With operator intervention, the wetwell sprays could be combined with either the suppression pool cooling mode or low pressure vessel injection mode. An orifice is included in the wetwell spray line to limit the flow to the stated amount. Wetwell spray flow is balanced by the orifice to provide (1) enough flow to mitigate pressurization of the wetwell due to steam bypass and (2) not too much flow to exceed the negative design pressure of the wetwell as discussed in SSAR Section 6.2.1.1.4.

GE deviates from the SRP in Section 6.2.1.1.5.6.1 of the SSAR by not requiring automatic actuation of the wetwell sprays 10 minutes after a LOCA signal. GE's analyses in Section 6.2.1.1.5.3 of the SSAR show that approximately 30 minutes is available for initiation of the containment sprays after the wetwell pressure has passed the lower limit of the containment spray actuation levels.

The staff concludes that GE has demonstrated that sufficient time is available to initiate containment sprays for the analyzed bypass scenario. Therefore, the staff agrees that automatic initiation of containment spray is not necessary.

6.2.1.8.3 Periodic Testing and Inspection

The acceptance criteria in SRP Section 6.2.1.1.C indicate the need for a preoperational high-pressure leak test, periodic low-pressure leak tests, and periodic visual inspection of the vacuum relief system. In Section 6.2.1.1.5.7 of the SSAR, GE committed to perform these tests and inspection. The acceptance criterion for both the high- and low-pressure leak tests is a measured bypass leakage area that is less than 10 percent of the effective suppression pool bypass capability ($A/K^{1/2}$) of 50 cm² (0.054 ft²).

The preoperational high-pressure leak test is discussed in SSAR Sections 6.2.1.1.5.7.1 and 14.2.12.1.41. The low-pressure leak test is discussed in SSAR Section 6.2.1.1.5.7.2 and is included in Surveillance Requirement 3.6.1.1.3 of TS 3.6.1.1, "Primary Containment." The periodic visual inspection of the vacuum relief system is discussed in SSAR Section 6.2.1.1.5.7.4 and will be performed each refueling outage in accordance with TS 3.6.1.6, "Wetwell-to-Drywell Vacuum Breakers."

The staff concludes that these testing and inspection requirements will ensure that the suppression pool steam bypass capability of the containment is not exceeded.

6.2.1.8.4 Vacuum Breaker Position Indicators and Alarms

In Section 6.2.1.1.5.8.1 of the SSAR, GE committed to provide redundant position indicators on all vacuum breakers with redundant indication and alarm in the control room. The sensitivity of the indicator system is adequate to detect a total valve opening, for all valves, that is less than the design bypass capability for a small break. The detectable valve opening is based on the assumption that it is evenly divided among all vacuum breakers.

In its discussions with GE, the staff stated that the redundant indicator and alarm system should be tested and

Engineered Safety Features

calibrated periodically and included in the TS. TS 3.6.1.6 requires a channel calibration of the vacuum breaker position indicator system every 18 months, thus ensuring the system's sensitivity. The staff finds this acceptable.

The staff concludes that the vacuum breaker position indicator and alarm system will ensure that the contribution of steam bypass of the suppression pool through the vacuum breaker system is within the containment design basis.

6.2.1.8.5 Monthly Vacuum Breaker Operability Test

GE deviates from the SRP in Section 6.2.1.1.5.6.2 of the SSAR by not requiring the monthly operability test of the vacuum breakers. This test was designed to ensure that the vacuum breakers adequately perform the design function of opening and then returning to a fully closed position.

There are eight vacuum breaker lines in the ABWR containment with one swing check valve per line, as opposed to most operating BWRs, which have two valves per penetration. In the DFSE, the staff concluded that the wetwell-drywell vacuum breaker system was acceptable. Failure of one of the vacuum breakers to open is within the design basis of the plant, as only seven are required to relieve pressure differentials.

Given the acceptability of a single barrier in the vacuum breaker lines, monthly operability testing could result in one of the valves failing to reseat because of the binding of the mechanical operator. Failure of one of the vacuum breakers to close could result in the containment pressure exceeding its design basis.

With only one barrier per vacuum breaker line, the staff concludes that monthly operability testing, which could result in failure of the valves to reseat or in binding of the mechanical operators, may not be necessary because the valves are designed for single failure to open. However, the staff believes that the single barrier per line places additional importance on the vacuum breaker position indicator and alarm system and wetwell spray system as discussed above. These systems will ensure that the design-basis steam bypass of the suppression pool is not exceeded through the vacuum breakers. In addition, TS 3.6.1.6 requires a plant shutdown within 12 hours when one vacuum breaker is not closed or cannot be demonstrated to be closed.

6.2.1.8.6 Conclusion

The staff concludes that GE's analysis, provisions for periodic testing and inspection, and vacuum breaker

position indicator and alarms are acceptable and meet SRP Section 6.2.1.1.C (Rev. 6), "Pressure-Suppression Type BWR Containments," relative to steam bypass of the suppression pool. In addition, GE has provided adequate justification for not providing automatic actuation of wetwell sprays and not requiring monthly operability (stroke) tests of the vacuum breakers. Since failure of one of the vacuum breakers to close could result in the containment pressure exceeding its design basis, this acceptability relies greatly on the instrumentation provided to detect opening of the vacuum breaker lines and places added importance on the wetwell spray system.

6.2.1.9 Containment Debris Protection for ECCS Strainers

The emergency core cooling system (ECCS) suction strainers are located in the suppression pool, and their function is to ensure that debris in the suppression pool does not lead to clogging of ECCS pumps, heat exchangers, valves, and spray nozzles. To accomplish this function, debris in the suppression pool will be filtered out on the surface of the suction strainers. An excessive accumulation of debris on the strainer surface could lead to inadequate net positive suction head (NPSH) and failure of the ECCS pumps.

In 1985, the NRC issued RG 1.82, Revision 1, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident," which contains guidance on the sizing criteria for ECCS strainers. Recent events at operating reactors involving the clogging of ECCS strainers have led the staff to conclude that the guidance in RG 1.82, Revision 1, may not be conservative enough to eliminate this concern. To address this issue for operating reactors, the NRC issued Information Notice (IN) 92-71, "Partial Plugging of Suppression Pool Strainers at a Foreign BWR"; IN 93-34, "Potential for Loss of Emergency Cooling Function Due to a Combination of Operational and Post-LOCA Debris in Containment"; Supplement 1 to IN 93-34; and Bulletin 93-02, "Debris Plugging of Emergency Core Cooling Suction Strainers." The staff is still working on resolving this issue for operating reactors.

To address this issue for the ABWR, GE committed to the following: (1) the guidance for strainer sizing in RG 1.82, Revision 1, will be met; (2) for surfaces in contact with the suppression pool, stainless steel will be used; (3) the upper drywell connecting vent openings will be protected with horizontal plates and vertical trash racks; (4) the strainers will be in a "T" configuration; (5) a suppression pool cleanup system will be used; and (6) the COL applicant will develop a program for maintaining suppression pool cleanliness.

In the advance SER, the staff stated that it believed that the actions specified by GE were appropriate; however, GE did not address the possible lack of conservatism in RG 1.82, Revision 1, due to the deleterious effect of finely fragmented insulation. Reducing the total amount of insulation in the containment would not resolve this problem because the sizing criterion is based on correlations in the regulatory guide. Therefore, less insulation would lead to smaller strainers. The staff believed an acceptable resolution to this issue was to size the strainers in accordance with RG 1.82, Revision 1, but provide a factor of 3 sizing margin to account for uncertainty in the synergetic effects of strainer clogging from insulation, corrosion products, and other debris. In the advance SER, the staff identified this as Open Item F6.2.1.9-1.

In Amendment 35, GE committed to size the RHR suction strainers 3 times the area that results from RG 1.82 for all breaks. The HPCF and RCIC system suction strainers are sized according to RG 1.82, but with conservatism in the mass of debris assumed to be deposited on the strainers from a design basis aspect. GE also committed to provide a 10 percent margin in the net positive suction head available from the static head of the suppression pool for conservatism. Based on the commitments in Amendment 35, the staff finds this acceptable. This resolved Open Item F6.2.1.9-1.

6.2.2 Containment Heat Removal System

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

The containment heat removal system encompasses the following RHR operating modes:

- Low-Pressure Flooder (LPFL) Mode

After a LOCA, containment cooling starts as soon as the LPFL injection flow begins. During this mode, water from the suppression pool is pumped through the RHR heat exchangers and injected into the reactor vessel. The LPFL mode is automatically initiated by a low water level in the reactor vessel or high pressure in the drywell. In addition, each loop in the RHR

system can be placed in operation by means of a manual initiation push-button switch.

- Suppression Pool Cooling Mode

After a LOCA, the suppression pool cooling subsystem provides a means to remove heat released into the suppression pool. During this mode of operation, water is pumped from the suppression pool through the RHR heat exchangers and back to the suppression pool. This mode is automatically initiated, as needed, by closing the LPFL injection valves and opening the suppression pool return valves. In response to a request for additional information (RAI), GE stated that the heat removal function will be initiated within 10 minutes after a LOCA. The staff found this to be sufficiently conservative and adequate to achieve the necessary containment cooling function.

- Containment (Wetwell and Drywell) Spray Cooling Mode

Two of the RHR loops include containment spray cooling subsystems. Each subsystem provides both wetwell and drywell spray cooling. This subsystem provides steam condensation and primary containment atmospheric cooling after a LOCA by pumping water from the suppression pool, through the RHR heat exchangers, and into the wetwell and drywell spray spargers in the primary containment. The normal mode of containment spray operation is combined wetwell and drywell sprays. However, the wetwell sprays can be operated in conjunction with either the suppression pool cooling mode or low pressure flooder mode. The drywell sprays can be operated in isolation through a series of operator actions; however, this is not intended to be used in isolation following an accident.

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

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

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

Engineered Safety Features

is acceptable. The suppression pool makeup system will provide additional water from the condensate storage tank (CST) through the suppression pool cleanup (SPCU) system to the suppression pool by gravity flow during normal conditions. Following a LOCA, the ECCS will take suction from the suppression pool. The quantity of water is sufficient to account for all conceivable postaccident entrapment volumes (i.e., places where water can be stored while maintaining long-term of the drywell vents with water).

The staff concludes that the containment heat removal systems satisfy SRP Section 6.2.2 and RG 1.1 and are acceptable.

6.2.3 Secondary Containment Functional Design

The ABWR secondary containment boundary completely surrounds the primary containment and is designed to remove fission products released from the primary containment during a DBA to limit the whole-body and thyroid doses to 10 CFR Part 100 requirements at the site boundary and the GDC 19 limit for the control room operator, as discussed in Chapters 12, 15, and Section 6.4 of this report. The staff reviewed the secondary containment functional design in accordance with SRP Section 6.2.3. The design is considered acceptable if the relevant requirements of GDC 4, 16, and 45 and Appendix J to 10 CFR Part 50 are complied with. The relevant requirements are as follows. GDC 4 as it relates to structures, systems, and components important to safety being designed to accommodate the effects of normal operation, maintenance, testing and postulated accidents, and being protected against dynamic effects (e.g., the effects of missiles, pipe whipping, and discharging fluids) that may result from equipment failures. GDC 16 as it relates to reactor containment and associated systems being provided to establish an essentially leak-tight barriers against the uncontrolled release of radioactivity to the environment. GDC 43 as it relates to atmosphere cleanup systems having the design capability to permit periodic functional testing to assure system integrity, the operability of active components, and the operability of the system as a whole and the performance of the operational sequence that brings the system into operation. 10 CFR Part 50, Appendix J as it relates to the secondary containment being designed to permit preoperational and periodic leakage rate testing so that bypass leakage paths are identified.

The components of the secondary containment are designed to withstand missiles, pipe whip, postaccident environments, seismic events, a single active failure, and a loss of offsite power coincident with an accident, in accordance with GDC 4. The two systems that fulfill this function are the secondary containment heating, ventilation

and air conditioning (HVAC) system and the standby gas treatment system (SGTS). They are both located inside secondary containment.

The HVAC system will maintain a slightly negative pressure in the secondary containment during normal operation to prevent any radioactivity from escaping to the environment. The SGTS will provide postaccident filtration and removal of airborne halogens and particulates from the secondary containment. The SGTS is designed to maintain a negative pressure of -0.25 in. water gauge (secondary containment to environment) after any postulated accident. GE indicates that testing and inspection of the integrity of the secondary containment is part of the testing of the SGTS. The staff's evaluation of the SGTS is given in Section 6.5.3 of this report.

SRP Section 6.2.3 states that all openings, such as personnel doors and equipment hatches, should be under administrative control. These openings should be provided with position indicators and alarms having readout and alarm capability in the main control room (MCR). The effect of open doors or hatches on the functional capability of the depressurization and filtration systems should be evaluated. The staff requested (in RAI Q430.34) that GE provide a list of the secondary containment openings and the instrumentation used to ensure that each one is closed during a postulated DBA. In response to RAI Q430.34, GE submitted SSAR Table 6.2-9, which lists all secondary containment penetrations along with their elevation and diameter; however, GE did not sufficiently address the staff concerns. The staff identified this issue as Open Item 14 in the DSER (SECY-91-355).

To address the staff's concern, GE revised SSAR Table 6.2-9 and provided notes to the table, marking the applicable penetrations with an asterisk. The notes state that (1) the HVAC openings have safety-related isolation valves with both local monitoring and remote (in control room) monitoring and (2) the doors are monitored in the control room in accordance with SSAR Section 13.6.3.4. The staff finds that GE's response is acceptable because the other penetrations, not covered by the notes, are fluid piping systems that are designed under containment isolation provisions. Therefore, DSER Open Item 14 is resolved.

Furthermore, in RAI Q430.32 and Q430.34, the staff asked for information on the capability of the SGTS to draw a negative pressure following an accident assuming that all lines that do not receive an isolation signal are open and assuming the worst-case single failure of a secondary containment isolation valve (CIV) to close. In addition, the staff asked GE to identify the instrumentation used to ensure that the CIVs shut. GE's initial response

did not specify the alarm capability for these indicators and did not address alarms available in the control room.

In response to RAI Q430.32, GE revised SSAR Section 6.5.1.3.1 to state that a secondary containment draw-down analysis will be performed to demonstrate the capability of the SGTS to maintain the designed negative pressure following a LOCA, including inleakage from the open, nonisolated penetration lines identified during construction engineering and in the event of the worst single failure of a secondary CIV to close. GE also added SSAR Section 6.5.5.1 to state that the COL applicant will perform an SGTS dose, functional damage, and draw-down analysis in accordance with SSAR Sections 6.5.1.2.3.7 and 6.5.1.3.1(5). The staff found GE's response acceptable. This was identified as COL Action Item 6.2.3-1 in the DFSEER.

To further address this issue, GE submitted a list of secondary containment penetrations in SSAR Table 6.2-9. All piping and cable-tray penetrations will be sealed with a sealing compound for leakage and fire protection and are all provided with containment isolation features. To ensure that the SGTS is capable of maintaining the design negative pressure, GE stated that the draw-down analysis would assume 50-percent containment inleakage regardless of valve positions. During construction engineering, the COL applicant will identify the piping that could cause inleakage. SSAR Section 6.5.1.3.2 states that each SGTS fan is sized to individually establish a continuously negative differential pressure (considering the effect of wind) within 10 minutes after SGTS initiation. The dose analysis assumes direct leakage from the containment to the environs for twice the required draw-down period. The staff finds that the SGTS design includes some conservatism when considering any malfunction of secondary containment isolation features. The staff concludes that it does not need to evaluate the results of the draw-down analysis for its FSER safety finding. Therefore, COL Action Item 6.2.3-1 is acceptable as included in the SSAR.

The staff has reviewed the secondary containment functional design and the capability of the SGTS to maintain a negative pressure in the secondary containment following a LOCA. The SGTS has two parallel and redundant filter trains, both of which will be automatically actuated (one train will be placed in the standby mode) by a high drywell pressure signal or a low reactor water level signal, or when high radioactivity is detected in the secondary containment or refueling floor ventilation exhaust. The system will be on standby during normal plant operation and may be manually initiated from the control room for primary containment de-inerting prior to plant shutdown. Normal operation of the SGTS at power

is limited to 90 hours per year for both trains combined. There are alarms and indicators for all LOCA and high-radiation mitigation systems.

The staff concludes that the secondary design and the SGTS meet the requirements of GDC 16 as it relates to establishing an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment, and are adequately designed to ensure that failure of any active component, assuming loss of offsite power, cannot impair the capability of the system to perform its safety function and, therefore, is acceptable.

Secondary Containment Bypass Leakage

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

BTP CSB 6-3 states that the evaluation of bypass leakage involves both the identification of bypass leakage paths and the determination of leakage rates. Potential bypass leakage paths are formed by penetrations which pass through both the primary and secondary containment boundaries. Penetrations that pass through both the primary and secondary containment may include a number of barriers to leakage (e.g., isolation valves, seals, gaskets, and welded joints). While each of these barriers aid in the reduction of leakage, they do not necessarily eliminate leakage. Therefore, in identifying potential leakage paths, each of these penetrations should be considered, together with the capability to test them for leakage in a manner similar to the containment leakage tests required by Appendix J to 10 CFR Part 50.

In RAI Q430.33, the staff requested GE to specify and justify the maximum allowable fraction of primary containment leakage that may bypass the secondary containment boundary. In its response, GE stated that only valve leakage through process piping can bypass the secondary containment and that leakage will be monitored via the containment local leakage test (Type C test) on the outboard CIV valves. In response to the related RAI

Engineered Safety Features

Q430.52 (b and c), GE stated that information on potential bypass leakage paths is given in SSAR Section 6.2.3 and SSAR Table 6.2-10.

GE has evaluated each penetration in the secondary containment in accordance with the guidance of BTP CSB 6-3. SSAR Table 6.2-10 identifies where each path terminates, the leakage barriers for the path, and whether the path is considered a potential bypass leakage path. The potential bypass leakage paths identified in SSAR Table 6.2-10 are the drywell and wetwell purge systems and the main steam system. Each of these paths has redundant containment isolation valves which are Type C tested in accordance with Appendix J. Additionally, the drywell and wetwell purge systems terminate inside the secondary containment where the SGTS collects and treats any leakage. GE has accounted for 4.2 m³/hr (140 standard cubic feet per hour (SCFH)) leakage through the MSIVs as the secondary containment bypass leakage rate to the environment, and this leakage has been treated separately in the offsite dose analysis. The testing and inspection of the integrity of the secondary containment will be part of the testing of the SGTS as described in SSAR Section 6.5.1. Therefore, the staff finds that the ABWR meets the requirements of GDC 43 and 10 CFR Part 50, Appendix J regarding the inspection and testing of the secondary containment system, and is acceptable.

Tier 1 design information and ITAAC are required for the functional design of the secondary containment. GE submitted the design description and ITAAC for the secondary containment. However, the results of the staff's review was not complete when the DFSEER was issued. This was identified as Open Item 6.2.3.1-1 in the DFSEER. GE subsequently submitted a revised design description and ITAAC. The adequacy and acceptability of the design description and ITAAC are evaluated in Section 14.3 of this report. Therefore, DFSEER Open Item 6.2.3.1-1 is resolved.

The staff concludes that the secondary containment functional design and the SGTS are in compliance with GDC 4, 16, and 43, and Appendix J to 10 CFR Part 50, and with the guidance in SRP Section 6.2.3 and BTP CSB 6-3 and, therefore, are acceptable.

6.2.4 Containment Isolation System

The containment isolation system includes containment isolation valves (CIVs) and associated piping and penetrations necessary to allow normal or emergency passage of fluids through the primary containment while preserving the capability to prevent or limit the escape of fission products from the containment boundary in the event of a LOCA. The staff's review of this system

includes the number and location of isolation valves, valve actuation signals and valve control features, the positions of valves under various plant conditions, the protection afforded isolation valves from missiles and pipe whip, and the environmental design conditions specified in the design of components. The containment isolation system design bases and containment isolation provisions should conform to GDC 1, 2, 4, 16, 54, 55, 56, and 57, as appropriate. Justification should be provided if deviations from these requirements exist.

GDC 1, 2, and 4 as they relate to systems important to safety being designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed; systems being designed to withstand the effects of natural phenomena (e.g., earthquakes) without loss of capability to perform their safety functions; and systems being designed to accommodate postulated environmental conditions and protected against dynamic effects (e.g., missiles, pipe whip, and jet impingement), respectively. GDC 16 as it relates to a system, in concert with the reactor containment, being provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment. GDC 54 as it relates to piping systems penetrating the containment being provided with leak detection, isolation, and containment capabilities having redundant and reliable performance capabilities, and as it relates to design provision incorporated to permit periodic operability testing of the containment isolation system, and leak rate testing of isolation valves. GDC 55 and 56 as it relates to lines that penetrate the primary containment boundary and either are part of the reactor coolant pressure boundary or connect directly to the containment atmosphere being provided with isolation valves as follows:

- (a) One locked closed isolation valve inside and one locked closed isolation valve outside containment; or
- (b) One automatic isolation valve inside and one locked closed isolation valve outside containment; or
- (c) One locked closed isolation valve inside and one automatic isolation valve outside containment; or
- (d) One automatic isolation valve inside and one automatic isolation valve outside containment.

GDC 57 as it relates to lines that penetrate the primary containment boundary and are neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere being provided with at least one

locked closed, remote-manual, or automatic isolation valve outside containment.

The containment isolation provisions of the ABWR for system lines that penetrate the containment can be classified into three areas:

- (1) system lines that meet the explicit requirements of GDC 54, 55, 56, and 57 regarding leak detection, isolation, and valve location and testing
- (2) system lines that differ from the explicit requirements of GDC 54, 55, 56, and 57, but that difference has been justified under the specific guidelines given in SRP Section 6.2.4, which constitute acceptable alternate containment isolation provisions
- (3) other lines that must be reviewed on a case-by-case basis to determine if an acceptable alternative basis exists for allowing a deviation from the explicit GDC on grounds not previously articulated in SRP Section 6.2.4.

During its review, the staff requested GE to provide additional information regarding the containment isolation provisions. RAI Q430.31, Q430.32, Q430.34, Q430.35, Q430.36, Q430.37, Q430.39, Q430.40, Q430.41, Q430.43, and Q430.44, all involve issues affecting the containment isolation system design. GE responded to all of these questions, and its responses are acceptable with the exception of those to Q430.34 and Q430.36.

Portions of GE's responses to Q430.32 and Q430.34, regarding secondary containment isolation related to the SGTS draw-down analysis, are evaluated in Section 6.2.3 of this report. In the DSER (SECY-91-355), the staff stated that the SGTS draw-down analyses should be included in the SGTS ITAAC. This was incorrectly identified in the DFSE as Open Item 6.2.4.1-1. It should have been DFSE Open Item 6.2.4-1. GE submitted a revised set of design descriptions and ITAAC. The adequacy and acceptability of the design description and ITAAC are evaluated in Section 14.3 of this report. On the basis of this evaluation, DFSE Open Item 6.2.4-1 is resolved.

In its submittal of March 11, 1992, GE further responded to RAI Q430.34 by stating that instrumentation requirements for the secondary containment openings were contained in SSAR Section 6.2.3.5. Subsequently, GE added a statement in SSAR Section 6.2.3.2 that all piping and cable-tray penetrations will be sealed with a sealing compound for leakage and fire protection. All doors are vestibule type with card reader access security systems that

are monitored. The HVAC penetrations are designed to close in the event of a design-basis accident. The required testing procedures and frequency are included in the TS.

In Q430.36, the staff asked GE to identify the systems and the relevant interface requirements for CIVs not within the ABWR scope and discuss essential and nonessential systems in accordance with RG 1.141, "Containment Isolation Provisions for Fluid Systems," Revision 0. GE stated that all isolation valves are within the scope of the ABWR standard plant. However, it did not provide containment isolation provisions for the essential and nonessential systems. This was identified as Open Item 15 in the DSER (SECY-91-355).

In response to DSER Open Item 15, GE stated that containment isolation provisions for the isolation valves were specified in SSAR Table 6.2-7, which gives information on CIVs on a system-by-system basis. However, the table did not identify the system lines as essential or nonessential to address the Three Mile Island (TMI) action plan requirements (NUREG-0737, Item II.E.4.2). RG 1.141 contains guidance on the classification of essential versus nonessential systems. Each nonessential penetration (except instrument lines) is required to meet GDC 54, 55, 56, and 57 and will be isolated automatically by the containment isolation signal. Essential systems, such as systems related to engineered safety features (ESF) or systems needed for safe shutdown of the plant, may include remote-manual CIVs.

The staff finds that SSAR Table 6.2-7 contains the valve information on a system-by-system basis with containment isolation provisions for both ESF and non-ESF systems. It concludes that the valve actuation for the ESF system reflects the function of the essential system, and is acceptable. Therefore, DSER Open Item 15 is resolved.

In the DSER (SECY-91-355), the staff stated that although GE had specifically committed to meet GDC 54, 55, 56, and 57, there was no commitment to meet GDC 1, 2, 4, and 16 in accordance with SRP Section 6.2.4. GDC 1 requires that structures, systems, and components (SSCs) important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. GDC 2 requires that SSCs important to safety be protected from the effects of natural phenomena. GDC 4 requires that these SSCs be protected against dynamic effects. GDC 16 requires that the reactor containment and associated systems be provided to establish an essentially leaktight barrier against the uncontrolled release of radioactivity to the environment. This lack of commitment to meet GDC was identified as DSER Open Item 16.

Engineered Safety Features

In its response to this open item, GE stated that the ABWR commitments to GDC 1, 2, 4, and 16 are contained in SSAR Sections 3.1.2.1.1, 3.1.2.1.2, 3.1.2.1.4, and 3.1.2.2.7, respectively. The staff reviewed SSAR Section 3.1 and found that GE had met the above-cited GDC in the ABWR design. However, SSAR Section 3.1.1 states that the GDC are intended to guide the design of all water-cooled nuclear power plants and the GDC are subject to a variety of interpretations. In reviewing these criteria in SSAR Section 3.1.2, the staff noted that some of the wording pertaining to the GDC did not match the wording in 10 CFR Part 50. The staff pointed out these discrepancies and asked GE to change the wording. GE made the changes in SSAR Section 3.1.2. The staff finds this acceptable. Therefore, DSER Open Item 16 is resolved.

In addition, in response to Q430.41, GE stated that instead of meeting the requirements of GDC 56 for the high-pressure core flooders (HPCF) and residual heat removal (RHR) test and pump miniflow bypass lines, reactor core isolation cooling (RCIC) turbine exhaust and pump miniflow bypass lines, and suppression pool cleanup (SPCU) suction and discharge lines, it will use GE Safety Standard 20, Nos. 8 and 9. The staff stated that these standards did not meet the explicit requirements of GDC 56 and GE must provide justification for this deviation. This was identified as Open Item 17 in the DSER (SECY-91-355) and as Open Item 6.2.4-3 in the DFSER.

GE stated that GE Standard 20, Nos. 8 and 9, are GE's criteria to meet GDC 56 on another defined basis. Specifically, No. 8 is the criterion applicable to a line that (1) penetrates the containment, (2) communicates with the containment interior, (3) is not an instrument line, and (4) is not a suppression pool effluent line. No. 9 is similar to No. 8, but is applicable to the effluent line that communicates with the suppression pool. The term "communicates" is intended to mean that these lines are not closed loops, they may leak, or open to the containment interior or the suppression pool.

GE Safety Standard 20, No. 8, requires that each of these lines be provided with two isolation valves. At least one valve should be located outside the containment; the other valve may be located either inside or outside the containment. Alternatively, one isolation valve outside the containment, which will be normally closed (or a blind flange) and will not receive a signal to open after an accident, may be used. On influent lines having two valves, one may be a check valve, and the valve outside the containment must be capable of automatic or remote-manual closure, or should be normally locked closed. On effluent lines or where a second valve is not provided on an influent line, these valves should be capable of

automatic and remote-manual closure or should be normally locked closed. The valves should be located as close as practicable to the containment. GE Safety Standard 20, No. 9, requires that the suppression pool effluent lines be provided with one remote-manual valve outside the containment that should be located as close to the containment as practicable. GE also provided additional information in SSAR Table 6.2-7 (Amendment 34) by adding notes h, i, j, k, l, q, and r to address the containment isolation provisions for these system lines.

The staff has reviewed the valve arrangement in SSAR Table 20.3.2-3 and SSAR Section 6.2.4.3.2.2 (Amendment 34) and finds that these system lines should not be automatically isolated following a containment isolation signal because of the specific safety function they will perform. Specifically, the RCIC system, in conjunction with the HPCF and RHR system, will provide core cooling to prevent excessive fuel temperature during a LOCA. To protect the RCIC pump from overheating, the RCIC system contains a pump miniflow bypass line that will discharge into the suppression pool when the line to the reactor vessel is isolated. When flow through the pump to the vessel is sufficient, the isolation valve in the miniflow line will close automatically, thus directing all flow to the reactor. The HPCF and RHR test and pump miniflow bypass lines have isolation capabilities commensurate with the importance to safety of isolating these lines. The RCIC pump miniflow bypass line will be isolated by a normally closed, remote manually actuated valve outside the containment. The RCIC turbine exhaust line, which penetrates the containment and will discharge to the suppression pool, is equipped with a normally open, motor-operated, remote-manual gate valve. The RHR and HPCF test and miniflow bypass lines each has a motor-operated valve outside the containment. The SPCU suction and discharge lines each has two isolation valves outside the containment because the penetration is under water. All these lines terminate below the suppression pool water level and will be sealed from the containment atmosphere following a LOCA.

Because of the specific safety function that these system lines will perform, their automatic isolation following a containment isolation signal is not appropriate. The staff finds that the containment isolation provisions do not degrade containment isolation capability and conform with the system's function. SRP Section 6.2.4 contains guidance for satisfying the GDC on another defined basis in that remote-manual actuation is provided for accident protection. Therefore, the staff concludes that the containment isolation design for the system lines meets GDC 56 on another defined basis (GE Safety Standard 20,

Nos. 8 and 9) and is acceptable. This resolved DFSER Open Item 6.2.4-3.

During its review of SSAR Table 6.2-7, which delineates CIV information in response to Q430.35, the staff identified some isolation valve design features that did not conform to SRP Section 6.2.4. In the table, the normal valve positions are either "open" or "closed." There is no way to determine if closed is the same as locked closed as it is stipulated in SRP Section 6.2.4. In addition, GE did not justify the isolation valve closure times of less than 30 seconds for some relatively large-diameter valves for the drywell atmosphere systems, such as atmospheric control system (ACS) 55 cm (22 in.)-diameter valve T31-F004 and flammability control system (FCS) 15 cm (6 in.)-diameter valve T49-F006A. The selection of the drywell atmosphere systems closure times, as specified in SRP Section 6.2.4, should be based on the analysis of radiological consequences for the DBA. In the DSER (SECY-91-355), the staff asked GE to identify which valves were "open," "closed," or "locked closed" and provide the technical basis for drywell atmosphere closure times. This was identified as part of Open Item 17 in the DSER (SECY-91-355).

GE stated that valve condition is given on the system piping and instrumentation diagram (P&ID) as locked open/closed, normally open/closed, and normally energized/deenergized. For clarity, the staff prefers that these valve positions are also specified in SSAR Table 6.2-7. In discussions with the staff, GE committed to make the change and this became Confirmatory Item 6.2.4-1 in the DFSER. In a subsequent amendment, GE added a note to SSAR Table 6.2.7 which states that the P&ID's identify which CIV's are locked closed. This is acceptable. This resolved DFSER Confirmatory Item 6.2.4-1.

In its submittal of March 11, 1992, GE stated that ACS isolation valves (T31-F001, -F002, -F004, -F006, and -F009) are 55 cm (22 in.)-diameter butterfly valves with an air operator that can travel 90° to close. These valves will be normally closed because their main function is to support containment purging and nitrogen inerting when the reactor is at less than 15-percent power. The smaller valves (T31-F005, -F039, -F040, and -F041) are 5 cm (2 in.)-diameter, air-operated globe valves that will close within 15 seconds. These air-operated valves may be opened for short periods during reactor operation to lower primary containment pressure or to keep the primary containment pressurized (by adding nitrogen) for preventing any air in-leakage. T31-F008 is a 25 cm (10 in.)-diameter, normally closed outboard air-operated butterfly isolation valve that will be connected to the SGTS and opened in series with T31-F005. T31-F025 is a 41 cm

(16 in.)-diameter, normally closed outboard butterfly isolation valve that will be opened only during initial inerting of the primary containment when the reactor is below 15-percent power. Open/close times of 20 seconds or less are planned for all except the 5 cm (2 in.)-diameter valves.

FCS isolation valves (T49-F001, -F002, -F006, and -F007) are normally closed 15 cm (6 in.)-diameter gate valves. These valves may be actuated individually for testing during reactor operation. Two of these valves will be air or nitrogen operated and two valves will be motor operated. This system will not be activated for days following a LOCA and a closure time of less than 30 seconds is planned.

In the DSER (SECY-91-355), the staff stated that the closure times for these valves, including instrumentation delay, should be determined by demonstrating compliance with 10 CFR Part 100 regarding offsite radiological consequences. The closure time should be justified on the basis of an analysis of the radiological consequences of a LOCA. However, GE had not explained how the valve open/close times were determined, or that the radiological consequences following a LOCA meet the requirements of 10 CFR Part 100. This was identified as Open Item 6.2.4-4 in the DFSER, which was part of Open Item 17 in the DSER (SECY-91-355).

GE stated that the FCS is a closed-loop system that will be capable of controlling combustible gas concentrations in the containment atmosphere for the design-basis LOCA without relying on purging and without releasing radioactive material to the environment. The time limit for valve closure is not important, since offsite radiological consequences need not be considered for a closed-loop system. Therefore, the staff finds that the closure time of less than 30 seconds for the FCS is acceptable.

SRP Section 6.2.4 states that the closure times of the containment purge and vent isolation valves should be established on the basis of minimizing the release of containment atmosphere to the environs and that isolation valve closure times of about 5 seconds or less may be necessary. SSAR Section 6.2.4.3.2.2.2.3 (Amendment 34) states that the ACS CIV closure time is equal to or less than 20 seconds. The ABWR TS require the 55 cm (22 in.)-diameter ACS CIVs to be normally sealed closed and to be opened only when reactor power is less than 15 percent during the 24-hour period following entry into Mode 3 for inerting the containment (startup) and the 24-hour period before entering Mode 4 for de-inerting the containment (shutdown). If open, they will automatically close on drywell high pressure or high radiation detected in the exhaust flow. The exhaust radiation detectors are

Engineered Safety Features

designed to be very sensitive and will be set at a lower setpoint compared to the ones inside the containment to ensure early detection. GE stated that the difference between 5 and 20 seconds is considered to be insignificant and that on the basis of its analysis, the closure times of these valves are significantly shorter than necessary to prevent any radiological impact. Additionally, GE stated that valves with moderate speed will be more reliable and, therefore, ensure containment integrity.

The staff finds that the closure time of 20 seconds or less for the ACS isolation valves is acceptable based on GE's accident analysis and justification that slower acting valves are more reliable. The accident analysis in Chapter 15 of the SSAR assumes, among other things, that all containment isolation valves isolate within 60 seconds following a postulated LOCA for calculating the offsite doses. The 20-second or less closure time of the ACS CIVs is small compared to this assumption. Additionally, the analysis assumes primary containment releases through the penetrations to be 0.5-percent by weight per day of the containment free volume. The contribution to the offsite doses due to the leakage through the ACS CIVs for the period from 5- to 20-seconds following a LOCA is insignificant when compared to the offsite doses due to the assumed leakage through the penetrations, and is acceptable. This resolved Open Item 6.2.4-4.

Containment Purge System

The staff requested GE (in RAI Q430.42) to address the containment isolation provisions for the purge lines that show conformance with BTP CSB 6-4, "Containment Purging During Normal Plant Operations," of the SRP. In its initial response, GE incorporated a new proprietary SSAR Section 9.4.5.6, "Primary Containment Supply/Exhaust System," which gives design information on the primary containment purge supply/exhaust system. The system consists of the supply fan, a high-efficiency particulate air (HEPA) filter, a purge fan, duct work, and controls. GE also provided additional information in SSAR Table 6.2-7 and Figure 6.2-39a (the atmospheric control system P&ID).

In response to RAI Q430.42, GE stated that the containment purge supply and exhaust lines, connected to both the drywell and wetwell, consist of one supply and one exhaust penetration each for the drywell and wetwell. Both the purge supply and exhaust lines, each of which is connected to both the drywell and wetwell, have two parallel isolation valves that will be located as close as possible to the outside of the primary containment. One of these valves (55 cm (22 in.)-diameter) is intended for use for (high-volume) inerting and purging. The other valve (5 cm (2 in.)-diameter) will be used for any necessary

nitrogen makeup during power operation. The two 5 cm-diameter exhaust valves will be used for any necessary venting (e.g., for pressure control) during operation. All these isolation valves are air operated and will automatically be closed by high drywell pressure or Level III low reactor vessel water level signals. They also will fail in the closed position if actuating power is lost. The large-diameter valves are butterfly-type valves with a closure time of less than 20 seconds. The small-diameter valves are globe-type valves and have closure times of less than 15 seconds. The staff finds that the containment isolation provisions for the purge valves are in conformance with BTP CSB 6-4. However, this valve configuration does not comply with GDC 56, which requires one isolation valve inside and one isolation valve outside the containment for each penetration. This was identified as Open Item 6.2.4.1-1 in the DFSER and as Open Item 18 in the DSER (SECY-91-355).

GE stated that purge and vent lines do not extend into the containment and have both inboard and outboard containment isolation valves (CIVs) located outside the primary containment so that they are not exposed to the harsh environment of the wetwell and drywell and are accessible for inspection and testing during reactor operation. SSAR Section 6.2.4.3 (Amendment 34) states, in part, that the CIVs for the atmospheric control system located outside the containment will be located as close to the containment as practical. The piping from the containment up to and including both valves is an extension of the primary containment boundary and is designed in accordance with ASME Code, Section III, Class 2 requirements. The CIVs are protected from the effects of flood and dynamic effects of pipe breaks in accordance with SSAR Sections 3.4 and 3.6. The arrangement of the isolation valves and connecting piping is such that a single failure of an inboard valve, or a single active or passive failure in the connecting piping or an outboard valve, cannot prevent isolation of the ACS. The valves are air operated with a pilot dc solenoid valve that will fail closed on loss of air or loss of electrical power. The power for the dc solenoids will be supplied from independent electrical divisions.

The staff's position on this issue is that locating both isolation valves outside the containment is acceptable if piping and valve design criteria are sufficiently conservative to preclude a breach of integrity. In general, the isolation barriers should be designed to ESF criteria and protected against floods, missiles, pipe whip, and jet impingement. GDC 56 permits containment isolation provisions for lines penetrating the primary containment boundary that differ from GDC 56, provided the basis for acceptability is defined. The staff concludes that this valve arrangement precludes a breach of piping integrity, meets

the single-failure criterion, and is protected against the effects of flooding, missiles, pipe whip, and jet impingement and meets GDC 56 on another defined basis, and is acceptable. Therefore, DFSER Open Item 6.2.4.1-1 is resolved.

In the DSER (SECY-91-355), the staff identified a number of criteria that are delineated in BTP CSB 6-4, but that GE did not address, including

- radiological consequence analysis for a LOCA with the purge system initially open (BTP CSB 6-4, B.5.a)
- system structural integrity design under LOCA thermal-hydraulic conditions (BTP CSB 6-4, B.5.b)
- design provisions to ensure that isolation valve closure is not prevented by debris entrained in escaping air and steam (BTP CSB 6-4, B.1.g)
- during emergency core cooling system (ECCS) backpressure containment pressure reduction analysis for a LOCA which the purge system is initially open (BTP CSB 6-4, B.5.c)
- evaluation of case-by-case purge isolation valve maximum allowable leak rate (BTP CSB 6-4, B.5.d)
- technical justification for a purge system isolation valve closure time of more than 5 seconds (BTP CSB 6-4, B.1.f)
- design provisions to ensure that simultaneous venting of the wetwell and drywell will not occur

In its submittal of March 11, 1992, GE addressed these criteria as follows:

- Radiological consequence analysis for a LOCA with the purge system initially open relates to SSAR Section 16.3.6.3.2, regarding the technical specification requirements for the oxygen concentration in the primary containment. The atmospheric control system (ACS) 55 cm (22 in.)-diameter purge isolation valves normally will be sealed closed during plant operation and will be opened only during the inerting (startup) and de-inerting (shutdown) process when reactor power is less than 15 percent and within 24 hours of reactor shutdown or startup. Within such a limited time, if a LOCA does occur, these valves will have closed before the onset of fuel failure. In the event of a radioactivity leak during inerting/de-inerting, the radiation detectors at the purge and vent exhaust line will detect the condition and isolate the ACS CIVs. GE stated that the exhaust radiation detectors are designed to be very

sensitive and will be set at a lower setpoint compared to the ones inside the containment to ensure early detection. Therefore, the potential for a LOCA to occur while the ACS 55 cm (22 in.)-diameter purge isolation valves are opened is small because the valves are allowed to be opened for limited times and the radiological consequence analysis shows that should a LOCA occur while they are open, site radiological limits will not exceed the limits of 10 CFR Part 100.

- Penetrations, piping, isolation valves, and rupture discs will maintain their structural integrity for all accident conditions. Periodic Type C tests of the isolation valves will be conducted at the containment peak pressure. The design meets seismic Category I and safety Class 2 requirements. Isolation valves will close automatically following LOCA signals and will fail closed on loss of instrument air. Purge system isolation valves (AO-F001, AO-F002, AO-F003, AO-F004, AO-F006, and AO-F025) will be locked closed whenever the reactor is above 15-percent power. Purge exhaust isolation valve AO-F005 normally will be closed and will open only for 90 hours per year along with operation of the standby gas treatment systems (SGTS) for containment pressure control. Nitrogen will be added to the primary containment during reactor operation by opening isolation valves AO-F0039 and AO-F0040 for the drywell and AO-F0039 and AO-F0041 for the wetwell.
- Drywell and wetwell purge penetrations will have seismic Category I debris screens.
- ECCS systems with suction from the suppression pool are designed with the primary containment at atmospheric pressure and without crediting net positive suction head for containment backpressure.
- Case-by-case maximum allowable leakage rates for isolation valve are based on valve size, type, and containment peak pressure for Type C tests as required by the TS. Test connections are provided. This was DFSER TS Item 6.2.4.1-1. Since GE included the testing requirements in the ABWR TS, the TS item is resolved.
- Purge system isolation valves AO-F005, AO-F040, and AO-F041 will be opened for short periods during reactor operation for containment pressure control. These valves are all 5 cm (2 in.)-diameter and are capable of full closure within 5 seconds.

GE provided adequate information to address these criteria with the exception of simultaneous venting of the drywell and wetwell. BTP CSB 6-4 prohibits simultaneous venting

Engineered Safety Features

of the drywell and wetwell. In the DFSER, the staff stated that GE should show how this will be ensured. This was identified as Open Item 6.2.4.1-2.

In its submittal of January 22, 1993, GE proposed to add Section 6.2.4.3.5 to the SSAR. This section (Amendment 34) states, in part, that the large (55 cm (22 in.)) purge and vent lines for the ACS will not be used for purging or venting during normal reactor operation. The isolation valves in these lines are normally closed and they are not needed for pressure control of the containment during normal operations. Pressure control of the containment during operation will be maintained by a single, small (5 cm (2 in.)) nitrogen line and a single, small (5 cm (2 in.)) vent line. The small vent line will be attached to the 55 cm (22 in.) drywell purge exhaust line and bypass the closed 55 cm (F004) valve. There is no equivalent vent line from the wetwell. Therefore, the drywell and wetwell will not be vented simultaneously during operation, and the system has only one supply and one exhaust line as required by BTP CSB 6-4. The staff finds that GE's justification is acceptable. DFSER Open Item 6.2.4.1-2 is resolved.

The staff also requested (in RAI Q430.254) that GE explain how the drywell and wetwell purge supply and exhaust subsystems can meet BTP CSB 6-4. GE committed to provide such information. This was identified as Confirmatory Item 6.2.5-1 in the DFSER and as part of Open Item 21 in the DSER (SECY-91-355).

In response, GE stated that drywell and wetwell purging is not required during normal operation. The staff finds that BTP CSB 6-4 does not apply to the plant that does not have an online purge system. Therefore, DFSER Confirmatory Item 6.2.5-1 is resolved.

GDC 54 requires "containment capability having redundancy" for piping that penetrates the containment. However, SSAR Figure 6.2-39 shows common CIVs for the containment purge supply (T31-F001 from the reactor building HVAC system, T31-F025 from the 40 cm (16 in.) nitrogen purge line, and T31-F039 from the 5 cm (2 in.) nitrogen supply line) and exhaust (T31-F009 to the reactor building HVAC system and T31-F008 to the SGTS). The staff was concerned that this valve arrangement would leave the system vulnerable to common-mode failures. Each purge and exhaust line should have redundant and independent CIVs to comply with GDC 54. This was identified as Open Item 6.2.4.1-3 in the DFSER and as part of Open Item 18 in the DSER (SECY-91-355).

In its submittal of January 22, 1993, GE stated, in part, that the containment purge system has redundant CIVs,

each powered from an independent electrical division, and the CIVs are arranged so that any single active failure will not compromise the integrity of the containment. GE also stated that this arrangement has adequate redundancy and independence and is not unduly vulnerable to common mode failures.

The staff reviewed the valve arrangement in SSAR Figure 6.2-39 and finds that the common CIVs mentioned above are the outboard CIVs, which are redundant. There are independent inboard CIVs for containment purge supply (T31-F002, -F003), nitrogen supply (T31-F040, -F041) and exhaust (T31-F004, -F005, -F006, -F007). All of these inboard CIVs will be closed automatically on receipt of a containment isolation signal and are designed to fail in the closed position on loss of air or loss of power to the pilot solenoid valves. Therefore, the staff concludes that the valve arrangement does not degrade containment integrity, meets the single-failure criterion, and is acceptable. DFSER Open Item 6.2.4.1-3 is resolved.

10 CFR 52.47(a)(1)(ii) requires that an application for a standard design certification to include a demonstration of compliance with any technically relevant portions of the TMI-related requirements identified in 10 CFR 50.34(f). In accordance with 10 CFR 50.34(f)(2)(xiv), GE has incorporated the following containment isolation provisions in the ABWR design:

- The design ensures all nonessential systems will be automatically isolated by the containment isolation system on receipt of a containment isolation signal.
- Each nonessential system line (except instrument lines) is designed with two isolation barriers in series.
- The design allows resetting the isolation signal without automatically reopening the CIVs. The ABWR system design ensures that resetting the isolation signal will not result in the automatic reopening of the CIVs. The reopening of any CIV is on a valve-by-valve basis once the isolation signal has cleared and following a subsequent logic reset.
- The design utilizes a containment pressure setpoint for initiating containment isolation that is as low as compatible with normal operation. Specifically, GE has committed to a high drywell setpoint pressure of 0.14 kg/cm²g (2 psig) to isolate nonessential penetrations.
- The design includes automatic closing on a high radiation signal for all systems that provide a path to the environs. Specifically, the containment purge and vent isolation valves will close on high radiation levels

in the secondary containment HVAC air exhaust or in the fuel handling area HVAC air exhaust.

The staff finds that the containment isolation provisions for the ABWR design meet the requirements of 10 CFR 50.34(f)(2)(xiv).

GE submitted the design description and the ITAAC relating to the containment isolation system. The staff's review of this material was not complete when the DFSE was issued. This was identified in the DFSE as Open Item 6.2.4.1-4. GE subsequently provided a revised design description and ITAAC. The adequacy and acceptability of the design description and the ITAAC are evaluated in Section 14.3 of this report. On the basis of this evaluation, DFSE Open Item 6.2.4.1-4 is resolved.

The staff has reviewed the information in the SSAR and information submitted in response to staff questions concerning the containment isolation system to ensure conformance to SRP Section 6.2.4 and BTP CSB 6-4. It concludes, as described in the preceding section, that the containment isolation system meets GDC 1, 2, 4, 16, 54, 55, 56, and 57.

6.2.5 Combustible Gas Control in Containment

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

(1) Atmospheric Control System

The atmospheric control system (ACS) is designed to maintain the primary containment oxygen concentration below the maximum permissible limit (3.5 percent) in accordance with RG 1.7, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident," Revision 2, during normal, abnormal, and accident conditions to ensure an inert atmosphere. The containment atmosphere will be inerted using adequately sized nitrogen storage tanks that are provided with makeup capability. The ACS is designed to withstand missiles, pipe whip, flooding, tornados, a safe shutdown earthquake, LOCA environment,

and a single active failure. However, GE states that the ACS is non-safety grade, whereas the SRP Section 6.2.5 acceptance criteria to satisfy GDC 41 state that the combustible gas control system should be safety grade because this system is relied on to ensure that containment integrity is maintained following an accident. The staff identified this as Open Item 19 in the DSER (SECY-91-355).

GE stated that the ACS consists of a nitrogen supply, injection, and exhaust lines and the containment overpressure protection system. The safety-related functions normally associated with the ACS will be performed by the safety-related high-pressure nitrogen system (HPIN). The only portions of the ACS that are safety related are the containment penetrations and isolation valves that isolate the ACS from the HPIN. This is acceptable. In addition, the staff's review of the HPIN design is contained in Section 9.3.1 of this report, and the staff finds this HPIN design is acceptable. This resolved DSER Open Item 19.

(2) Containment Atmosphere Monitoring System

The containment atmosphere monitoring system (CAMS), as addressed in SSAR Section 7.6.1.6, is a safety-grade, seismic Category I system designed to meet GDC 41 by monitoring the drywell and suppression chamber for high levels of hydrogen, oxygen, and radiation during accident conditions. The system will allow operators to confirm that the containment is inerted so that containment nitrogen purging can be terminated and also to verify that de-inerting is complete to ensure safe personnel entry into the primary containment. The system has a measurement range of 0 to 25 percent (by volume) at 100-percent relative humidity.

The CAMS consists of two independent but redundant Class 1E divisions, which are electrically and physically independent to remain operable as a result of a single active failure coincident with a loss of offsite power. The system has two subsystems: radiation monitoring subsystem and hydrogen/oxygen monitoring subsystem. The radiation monitoring subsystem contains two channels per division; one for monitoring the drywell and the other for monitoring the suppression chamber. Each monitoring channel consists of an ionization chamber, a log rate meter, and a recorder. Each channel has a range of 1 to $10E+7/R$ hr and will initiate an alarm in the control room on high radiation level or on system failure. This subsystem can be initiated auto-

Engineered Safety Features

matically on a LOCA signal or manually actuated from the control room. The radiation monitors do not provide any bypass or interlock capability.

The hydrogen/oxygen monitoring subsystem consists of two divisions that will take samples of the drywell and wetwell and feed them to monitors for measurement, recording, and control room alarm. The piping is stainless steel and continuously heat traced to remove moisture from the system during measurements. Gas calibration racks are provided to perform equipment calibrations during operating conditions. A thermal delay bypass feature will allow system testing during power operation. There are no automatic isolation functions associated with the system.

Each CAM subsystem will be powered from a divisional 120-V ac Class 1E instrument bus. This power source also will supply the heat-tracing blanket used for the sampling lines.

The staff concludes that the CAMS has adequate capability for monitoring the containment atmosphere for hydrogen/oxygen control, thus satisfying GDC 41 regarding containment atmosphere cleanup.

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

Post-LOCA primary containment backup purging capability is required in accordance with RG 1.7 as an aid for containment atmosphere cleanup following a LOCA. During normal plant operation, the bleed line will function, in conjunction with the nitrogen purge line, to maintain primary containment pressure at about 5.2 kPag (0.75 psig) and oxygen concentration below 3.5 percent by volume. This will be done by making up the required quantity of nitrogen in the primary containment through the makeup line or relieving pressure through the bleed line. Flow through the bleed line will be directed through either the SGTS or the reactor building secondary containment HVAC system and will be monitored for radiation. However, GE provided neither the purge rate that would be required to maintain the oxygen concentration below 3.5 percent by volume nor the radioactive consequence analysis for the staff to review. This was identified as Open Item 6.2.5-1 in the DFSE and as Open Item 20 in the DSER (SECY-91-355).

GE stated that postaccident containment backup purging capability is not needed to maintain the oxygen concentration below 3.5 percent because the containment is inerted. Pressure in the containment during normal operation will be maintained by a 5 cm (2 in.) nitrogen supply line and a 5 cm (2 in.) vent line. No continuous purging is required during normal operation. The staff finds this acceptable. This resolved DFSE Open Item 6.2.5-1.

With respect to postaccident hydrogen generation analysis, GE stated that the analytical model described in GE report NEDO-22155, "Generation and Mitigation of Combustible Gas Mixtures in Inerted BWR Mark I Containment" (nonproprietary) was used to compute hydrogen and oxygen generation from radiolysis. NEDO-22155 was reviewed by the staff for the EPRI requirements document certification and found unacceptable. As a result, GE performed a new hydrogen generation analysis using the RG 1.7 methodology. The results of the analysis are provided in Appendix E of SSAR Chapter 19. On the basis of its review of GE's analysis, the staff concludes that the ABWR will be able to withstand a 100-percent fuel-clad metal-water reaction.

(4) Flammability Control System (Hydrogen Recombiner)

SSAR Section 6.2.5.2.7 states that there will be two permanently installed recombiners in the secondary containment. However, GE did not provide information on dedicated redundant containment penetrations to demonstrate that the recombiners could perform their safety function assuming a single active failure. Also, GE did not indicate whether the recombiners are safety grade. This was identified as Open Item 6.2.5-2 in the DFSE and as Open Item 21 in the DSER (SECY-91-355).

In Amendment 34, GE revised SSAR Section 6.2.5.2.7 which states, in part, that the flammability control system (FCS) consists of two permanently installed safety-related thermal hydrogen recombiners located in separate rooms in the secondary containment and controlled from the main control room. Independent drywell and suppression chamber penetrations are provided for the two recombiners. Each penetration has two normally closed isolation valves. The staff has reviewed GE's FCS design and finds that the recombiner configuration meets the single-failure criterion and is acceptable. Therefore, DFSE Open Item 6.2.5-2 is resolved.

10 CFR 52.47(a)(1)(ii) requires that an application for a standard design certification include a demonstration of compliance with any technically relevant portions of the TMI requirements set forth in 10 CFR 50.34(f). The staff's evaluation of the TMI-related requirements on hydrogen control is provided as follows:

(a) 10 CFR 50.34(f)(1)(xii), "Evaluation of Alternative Hydrogen Control Systems," requires the applicant to perform an evaluation of alternative hydrogen control systems that would satisfy the requirements of 10 CFR 50.34(f)(2)(ix). At a minimum, consideration should be given to a hydrogen ignition and postaccident inerting system and the evaluation should include

- a comparison of costs and benefits of the alternative systems considered
- analyses and test data to verify compliance with the requirements of 10 CFR 50.34(f)(2)(ix) for the selected system
- preliminary design descriptions of equipment, function, and layout for the selected systems

GE has provided information for the last two items as identified above but has not provided cost and benefit information for alternative systems. If the alternative systems are used in the design, this information must be provided by the COL applicant. This was identified as COL Action Item 6.2.5-1 in the DFSER.

In its submittal of March 5, 1993, GE added Section 6.2.7 to the SSAR. SSAR Section 6.2.7.1 (Amendment 32), regarding alternative hydrogen control, states that the COL applicant will provide a comparison of costs and benefits for alternative hydrogen control in accordance with SSAR Section 6.2.5. Therefore, COL Action Item 6.2.5-1 has been clarified and found to be acceptable. The staff concludes that the ABWR design meets the requirements of 10 CFR 50.34(f)(1)(xii).

(b) 10 CFR 50.34(f)(2)(ix), "Hydrogen Control System Preliminary Design," requires the applicant to provide a system for hydrogen control that can safely accommodate hydrogen generated by the equivalent of a 100-percent fuel-clad metal-water reaction. This system and any associated systems should provide, with reasonable assurance, that

- uniformly distributed hydrogen concentrations in the containment do not exceed 10 percent during and following an accident that releases an amount of hydrogen that would be equivalent to that generated from a 100-percent fuel-clad metal-water reaction, or that the post-accident atmosphere will not support hydrogen combustion
- combustible concentrations of hydrogen will not collect in areas where unintended combustion or detonation could cause loss of containment integrity or loss of appropriate mitigating features
- equipment necessary for achieving and maintaining safe shutdown of the plant and maintaining containment integrity will perform its safety function during and after its exposure to the environmental conditions attendant with the release of hydrogen generated by the equivalent of a 100-percent fuel-clad metal-water reaction including the environmental conditions created by activation of the hydrogen control system

In SSAR Section 19B.2.18, GE states that the ABWR containment will have an inert atmosphere and will be able to withstand the pressure and energy addition from a 100-percent fuel-clad metal-water reaction. The staff concludes that the inerted containment and the provision for permanently installed hydrogen recombiners are acceptable as hydrogen control measures and adequately address 10 CFR 50.34(f)(2)(ix).

(c) 10 CFR 50.34(f)(3)(iv) requires the applicant to provide one or more dedicated containment penetrations, equivalent in size to a single 0.9 m (3-ft)-diameter opening, in order not to preclude future installation of systems to prevent containment failure, such as a filtered vented containment system. This requirement is a followup of one of the requirements identified under TMI action plan, Item II.B.8 of NUREG-0660.

The staff's evaluation of the size of the penetration that can be used for venting the containment is provided in Section 20.5.44 of this report.

(d) 10 CFR 50.34(f)(3)(vi), "Dedicated Hydrogen Penetrations (Hydrogen Recombiners)," requires the applicant to provide redundant dedicated containment penetrations for plant designs with external hydrogen recombiners so that, assuming a

Engineered Safety Features

single active failure, the recombiner systems can be connected to the containment atmosphere.

SSAR Figure 6.2-40 shows redundant dedicated hydrogen recombiner penetrations. This satisfies 10 CFR 50.34(f)(3)(vi). However, in the DFSEER, the staff stated that GE had not provided design information for the recombiners in the SSAR, but had committed to include this information in a future SSAR amendment. This was identified as Confirmatory Item 6.2.5-2 in the DFSEER.

In Amendment 32, GE provided this information in SSAR Section 6.2.5.2.7. The staff reviewed this section and P&ID Figure 6.2-40 (Amendment 34) and finds that GE has addressed the FCS design requirements in the SSAR. This is acceptable. Therefore, DFSEER Confirmatory Item 6.2.5-2 is resolved.

- (e) 10 CFR 50.34(f)(2)(xv), "Containment Purging/Venting," requires the applicant to provide a capability for containment purging/venting designed to minimize purging time consistent with ALARA (as low as is reasonably achievable) principles for occupational exposure. It also requires assurance that the purge system will reliably isolate under accident conditions.

In SSAR Section 19A.2.27, GE states that during normal operation, all large valves in containment ventilation lines will be closed and only small, 5 cm (2 in.), nitrogen makeup valves will be opened. These are air-operated valves with rapid closure times that prevent substantial releases from the containment in the event of a transient requiring containment isolation. GE also states that the 5 cm (2 in.) nitrogen bleed lines will be sufficient to maintain normal containment pressure during normal operation when used in conjunction with containment spray and the drywell cooling system. However, SSAR Figure 6.2-39 shows that T31-F007 and T31-F010 (36 cm (14 in.) valves) also will be open during normal operation. In the DFSEER, the staff requested GE to correct this discrepancy. This was identified as Confirmatory Item 6.2.5-3.

In its submittal of March 5, 1993, GE revised SSAR Section 19A.2.27 to state, in part, that all large valves in containment ventilation lines will be closed during normal operation with the exception of two large valves in the containment overpressure protection system (COPS) where flow will be prevented by rupture discs in the piping.

Furthermore, in a submittal dated May 7, 1993, GE revised SSAR Table 3.9-8 to specify that valves T31-F007 and T31-F010 will be leakage rate tested every 2 years during refueling outages in accordance with the requirements of the of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code). The staff finds that this discrepancy is clarified, and is acceptable. Therefore, DFSEER Confirmatory Item 6.2.5-3 is resolved.

In SSAR Section 19A.3.3, GE states that a testing program will be provided to ensure that the large ventilation valves will close within the limits ensured in the radiological design bases. In the DFSEER, the staff stated that the test program should include valves T31-F007 and T31-F010 and that the COL applicant should submit details of these tests. This was identified as COL Action Item 6.2.5-1 in the DFSEER.

In response, GE stated that valves T31-F007 and -F010 will normally be open and will be closed every 2 years for leakage rate testing in accordance with the requirements as addressed in SSAR Table 3.9-8, "In-service Testing of Safety Related Pumps and Valves." The staff finds that these valves are passive pressure control valves, not the primary containment isolation valves. Containment integrity is preserved by the rupture discs downstream of the valves. These valves are subject to leak rate tests in accordance with ASME/American National Standards Institute (ANSI) OM-1987, Part 10, to verify their normally open position and their capability to close. Therefore, these valves should not be included in the ventilation valve testing program. On the basis of this review, COL Item 6.2.5-1 is deleted and need not be included in the SSAR.

In response to a request for additional information (RAI), GE agreed to amend the ABWR TS to allow a 24-hour (rather than a 72-hour) window at the beginning and end of a fuel cycle, during which the large-diameter (55-cm (22-in.)) purge lines can be open. GE has made this modification. This was DFSEER TS Item 6.2.5-1 and is resolved.

The staff finds that the methods used to maintain containment pressure without backup purging capability meet the requirements of 10 CFR 50.34(f)(2)(xv).

Three RAI questions (Q430.45, Q430.46, and Q430.47) on the combustible gas control system were transmitted to GE. These questions dealt with the subjects of scope and interface, compliance with RG 1.7, and BTP ASB 9-2 of the SRP regarding hydrogen and oxygen production and accumulation. GE has responded to all of these questions and amended SSAR Section 6.2.5. GE states in that section that the combustible gas control systems, consisting of the FCS and atmospheric control system, are completely within the scope covered by the ABWR SSAR and that there are no interfaces with equipment or systems outside the scope of this submittal. GE also states that the analysis of hydrogen and oxygen production is based on the parameters listed in RG 1.7. The fission product decay energy model used is that presented in SRP Section 9.2.5, BTP ASB 9-2, and inputs to the analysis are provided in SSAR Section 6.2.5.3. The staff finds that GE's response and the SSAR as amended adequately address the issues identified in the staff's RAIs.

Generic Letter (GL) 89-16, "Installation of Hardened Wetwell Vent," addressed the need for modifications of BWR containment designs to reduce their vulnerability to severe accident challenges. The staff finds that the ABWR design has included the containment overpressure protection system which addresses this GL. The ABWR design for severe accident conditions is evaluated in Section 19.2 of this report.

The staff concludes that the design of the combustible gas control systems including the containment ACS and FCS are acceptable and meet the requirements of 10 CFR Part 50, Sections 50.44 and 50.34 and GDC 41, 42, and 43 based on the following:

- (1) GE has resolved all the open issues on combustible gas control. The design meets the requirements of 10 CFR 50.44 with respect to means for controlling hydrogen and capability for measuring hydrogen concentration and inerting in the containment.
- (2) GE has demonstrated compliance with all technically relevant portions of the TMI requirements set forth in 10 CFR 50.34(f).
- (3) GE has met the requirements of GDC 41 with respect to systems being provided to control hydrogen and oxygen concentration in the containment following postulated accidents, GDC 42 with respect to periodic inspection of the systems, and GDC 43 with respect to periodic testing of the systems.

- (4) The design of the combustible gas control system meets the acceptance criteria set forth in SRP Section 6.2.5 and the limitation of hydrogen and oxygen concentration specified in RG 1.7.

6.2.6 Containment Leakage Testing

The staff reviewed GE's containment leakage testing program described in the SSAR for compliance with the containment leakage testing requirement in Appendix J to 10 CFR Part 50 (Appendix J). Such compliance provides adequate assurance that the containment leaktight integrity can be verified throughout the service lifetime and that the leakage rates will be checked periodically during service on a timely basis to maintain such leakage within the specified limits. Maintaining containment leakage within limits provides reasonable assurance that, if any radioactivity is released within the containment, the loss of the containment atmosphere through potential leak paths is not in excess of the limits specified for the site.

The staff reviewed the containment leakage testing program to ensure that the containment penetrations and system isolation valve arrangements are designed to satisfy the containment integrated and local leakage rate testing requirements of Appendix J.

Type A Test

The preoperational containment integrated leakage rate (Type A) test is intended to measure the primary containment overall integrated leakage rate after the containment has been completed and is ready for operation and the local leak rate tests (LLRTs) of all mechanical, fluid, electrical, and instrumentation systems penetrating the containment pressure boundary have been performed.

The objectives of the initial integrated leakage rate test (ILRT) are to

- verify that the containment integrated leakage rate does not exceed the containment design-basis accident DBA leakage rate, L_a , which is 0.5 percent by weight of the containment atmosphere in 24 hours, at peak containment accident pressure, P_a related to DBA
- establish a minimum allowable leakage rate, L_i , at reduced pressure, P_i , which is used during subsequent ILRTs
- obtain data that may be used to develop the leakage rate characteristics and history of the containment system

Engineered Safety Features

- demonstrate by a verification test the accuracy of the integrated leakage rate instrumentation to satisfactorily determine the containment integrated leakage rate

The preoperational test will be performed at both the reduced pressure, P_1 , and the peak containment accident pressure, P_a . P_1 will be chosen so that it is greater than $0.5 P_a$. After the initial ILRT, a set of three Type A tests will be performed at approximately equal intervals during each 10-year service period with the third test of each set coinciding with the end of each 10-year major inservice inspection shutdown. The total measured containment leakage rate, L_{tm} , at reduced pressure, P_1 , should not exceed $0.75 L_1$ as established by the initial ILRT. The accuracy of the leakage rate tests will be verified by using a supplemental method of leakage measurement.

In conducting a Type A test, certain systems that are normally filled with water and operating under post-LOCA conditions need not be vented to the containment atmosphere. In addition, systems required to function during the Type A test should be operable in their normal mode and need not be vented, but the test results for such systems should be added to the Type A test total. GE has confirmed that the leakage test values of such system lines will be added to the Type A test results. All other system lines will be vented or drained before the Type A test. However, during its initial review, the staff found that although GE's Type A test program was acceptable, GE had not identified the systems or the reasons why these systems would not be vented or drained during the ILRT. This issue was identified as a part of Open Item 27 in the DSER (SECY-91-153).

In response to this concern, GE stated that SSAR Section 6.2.6.1.3 provides additional criteria for the integrated leak rate test, thus addressing this issue. The staff reviewed SSAR Section 6.2.6.1.3 and finds that the criteria for system lines not to be vented or drained during a Type A test are acceptable. This part of DSER Open Item 27 is resolved.

In reviewing the certified design material for the primary containment system, the staff found that the design description and ITAAC stated that the main steam isolation valve (MSIV) leakage was not included in the primary containment allowable leakage of 0.5 percent. GE stated that the analytical predictions of radiological consequences in the SSAR had been based on the assumption of a containment leakage of 0.5 percent per day plus a separate MSIV leakage. The next revision of the SSAR was to clarify that MSIV leakage is to be excluded from the 0.5 percent per day. The staff reviewed SSAR Section 6.2.1.1.2 and finds that this issue has been adequately clarified.

Type B Tests

Containment penetrations whose designs incorporate resilient seals, gaskets, or sealant compounds; piping penetrations fitted with expansion bellows serving as the containment boundary; airlock door seals; equipment and access doors with resilient seals; and other testable penetrations are to be Type B tested during preoperational testing and thereafter at periodic intervals during the lifetime of the unit in accordance with Appendix J. The Type B tests are necessary to ensure the continuing integrity of the penetrations.

To facilitate LLRT, GE has proposed a permanently installed system consisting of a pressurized gas source (nitrogen or air) and the manifolding and valving necessary to subdivide the testable penetrations into groups of two to five. Each group will then be pressurized, and if any leakage is detected (by pressure decay or flow meter), individual penetrations can be isolated and tested until the source and nature of the leak is determined.

GE states, in SSAR Section 6.2.6, that the combined local leakage rate of all components subject to Type B and Type C tests (described in subsequent paragraphs) will not exceed 60 percent of L_a . Type B tests will be performed at peak containment accident pressure during each reactor shutdown for major fuel reloading, but in no case at intervals greater than 2 years. Airlocks will be tested at initial fuel loading and at least once every 6 months thereafter. Additionally, whenever they are opened during periods when containment integrity is required, they will be tested within 3 days of opening in accordance with Appendix J. These tests are required for the airlocks since they contain inflatable seals.

In the DSER, the staff stated that because the intent of the Appendix J testing program has never been to require a forced reactor shutdown just to conduct these tests within preset test intervals, GE would either have to clarify whether provisions for conducting all Type B tests at power exist in the ABWR design or request an exemption from the requirement for conducting Type B tests at 2-year intervals and justify the request. Also, GE had not provided (1) the acceptance criteria for testing the airlocks, (2) a list of all containment penetrations that are subject to Type B tests, and (3) a list of all penetrations that are excluded from Type B tests (if any) and the rationale for such exclusions. In the DSER, the staff stated that it could not conclude that the proposed Type B testing program for the ABWR was acceptable. These issues were identified as part of Open Item 27 in the DSER (SECY-91-153).

GE subsequently provided information on Type B test criteria and listed all containment penetrations in SSAR

Table 6.2-8. The staff finds that the Type B test criteria specified in SSAR Section 6.2.6 comply with Appendix J and SRP Section 6.2.6. SSAR Section 6.2.6.2 specifies that containment penetrations whose designs incorporate resilient seals, bellows, gaskets, or sealant compounds, airlocks and lock door seals, equipment and access hatch seals, and electric canisters, and other such penetrations are to be leak tested during preoperational testing and at periodic intervals thereafter in conformance with Type B leakage rate tests defined in Appendix J. SSAR Section 6.2.6.4 states that Type B and C tests may be conducted at any time during normal plant operations or during shutdown periods, as long as the time interval between tests for any individual Type B or C test does not exceed 2 years. The staff reviewed SSAR Section 6.2.6 and SSAR Table 6.2-8 and finds that the Type B test requirements are acceptable. This part of DSER Open Item 27 is resolved.

Type C Tests

All primary containment isolation valves whose seats are exposed to the containment atmosphere after a LOCA will be Type C tested pneumatically with air or nitrogen at P_a . Valves that are sealed by water will be leak tested with water as the test medium. The test pressure will be applied in the same direction as when the valve is required to perform its safety function, unless it can be shown that results from tests with pressure applied in a different direction are equivalent or more conservative. Type C testing will be performed by local pressurization, using either the pressure decay method (for pneumatic testing) or the flowmeter method (for both pneumatic and hydrostatic testing).

SSAR Section 6.2.6.3.1 states that Type C tests will be performed on all containment isolation valves required by Appendix J. All testing will be performed pneumatically, except that hydraulic testing might be performed on isolation valves using water as a sealant provided the valves were demonstrated to exhibit leakage rates that did not exceed those in the ABWR technical specifications. However, the SRP states that Type C testing with water is permissible only if the system line for the valve is not a potential containment atmosphere leak path. This was identified as Open Item 6.2.6-1 in the DFSER.

In response to the staff's concern, GE revised SSAR Section 6.2.6.3.1 (Amendment 21) to state that all testing will be performed pneumatically, except that hydraulic testing using water as a sealant may be performed during isolation valve Type C tests provided the system line for the valve is not a potential containment atmosphere leak path. The staff finds the statement acceptable and DFSER Open Item 6.2.6-1 is resolved.

During its initial review, the staff found that GE had not adequately responded to the staff's RAI of July 7, 1988, on a number of issues. These issues included (1) Type C test interval, (2) test pressure for main steam isolation valves, (3) test methodology for ECCS isolation valves, (4) test procedures for valves not covered by Appendix J procedures and a list of such valves, (5) a list of all primary containment isolation valves that will be Type C leak tested, (6) a list of all valves that will be hydrostatically tested and the test pressure, and (7) a list of all valves that will be tested in the reverse direction and the justification for such testing. Additionally, it was not clear whether lines that contain valves that do not have 30-day water-leg seals will be drained and the valves then pneumatically tested as required. These were identified as part of Open Item 27 in the DSER (SECY-91-153).

GE's response and the staff's evaluation of these issues follows:

- (1) SSAR Section 6.2.6.4 states that Type B and C tests may be conducted at any time during normal plant operations or during shutdown periods, as long as the time interval between tests for any individual Type B or C test does not exceed 2 years. GE has specified a Type C test interval. The staff finds this acceptable, and this part of DSER Open Item 27 is resolved.
- (2) SSAR Section 6.2.6.3.1 specifies that MSIVs and isolation valves isolated from a sealing system are to use a pressure of at least P_a . The staff finds that the test pressure meets Appendix J and is acceptable. This part of DSER Open Item 27 is resolved.
- (3) GE did not address test methodology for ECCS isolation valves. SSAR Section 6.2.6.3 does not mention this issue. This was identified as Open Item 6.2.6-2 in the DFSER.

GE stated that the test information is provided in revised SSAR Table 6.2-7. The staff reviewed SSAR Table 6.2-7 and finds that the test requirements for the ECCS isolation valves are addressed in its associated notes g, h, i, j, k, and n. This is acceptable. DFSER Open Item 6.2.6-2 is resolved.

- (4) GE did not provide test procedures for valves not covered by Appendix J and a list of such valves. This was identified as Open Item 6.2.6-3 in the DFSER.

Engineered Safety Features

GE stated that the test requirements for isolation valves not covered by Appendix J are addressed in SSAR Section 3.9.6.2 (inservice testing of safety-related valves), SSAR Section 3.9.7.3 (pump and valve inservice testing), and SSAR Table 3.9-8. These valves are subject to ASME Code, Section XI leak rate tests. The staff's evaluation of these valves is provided in Chapter 3 of this report. In response to RAI Q430.50h, GE stated that those valves not specifically Type C tested will be tested as part of the Type A test. GE has addressed the test requirements for the valves not covered by Appendix J. The staff finds this acceptable. Therefore, DFSER Open Item 6.2.6-3 is resolved.

- (5) GE listed all containment isolation valves that require Type C testing in SSAR Table 6.2-7. The staff finds that the test provisions for the values listed in the table meet Appendix J and are acceptable. This part of DSER Open Item 27 is resolved.

- (6) GE did not list all the valves subject to hydrostatic test as addressed in SSAR Section 6.2.6.3.1. This was identified as Open Item 6.2.6-4 in the DFSER, which was part of Open Item 27 in the DSER (SECY-91-153).

GE stated that the test requirements are provided in revised SSAR Table 6.2-7. The staff reviewed SSAR Table 6.2-7 (Amendment 34) and finds that GE has identified, in notes d, g, h, j, k, n, and q, those isolation valves that will be filled with water following a LOCA. SSAR Section 6.2.6.3.1 (Amendment 34) states that these valves are in lines designed to be, or remain, filled with a liquid for at least 30 days following a LOCA and will be leakage rate tested with that liquid. The liquid leakage measured will not be converted to equivalent air leakage or added to the Type B and C test total. The staff finds that the hydrostatic test criterion meets Appendix J and is acceptable. DFSER Open Item 6.2.6-4 is resolved.

- (7) GE did not justify and list the valves to be tested in the reverse direction. This was identified as a part of Open Item 27 in the DSER (SECY-91-153).

GE stated that SSAR Table 6.2-7, note e, provides the criterion for valves to be tested in the reverse direction and the valves listed in the table with that note will be tested in the reverse direction. The staff finds that the test requirement complies with Criterion III.C.1 of Appendix J and is acceptable. This part of DSER Open Item 27 is resolved.

- (8) GE did not specify whether lines that contain valves that do not have 30-day water-leg seals will be drained before the valves are pneumatically tested. This was identified as Open Item 6.2.6-5 in the DFSER and as part of Open Item 27 in the DSER (SECY-91-153).

GE stated that the valves that do not have 30-day water-leg seal are indicated in SSAR Table 6.2-7 by note (u) in the Type C test requirements entry. The staff verified that the isolation valves to be pneumatically tested are indicated by note (u) in the test requirements entry of the table, and finds that they are acceptable. DFSER Open Item 6.2.6-5 is resolved.

During its initial review of SSAR Section 6.2.6 regarding containment leakage testing, the staff also identified the following open items in the DSER (SECY-91-153).

- (1) GE did not indicate whether the test, vent, and drain connections used to facilitate ILRTs and LLRTs will be kept closed and under administrative control during normal plant operations and whether they would be subject to periodic surveillance testing to ensure their integrity and to verify the effectiveness of administrative controls. In the DFSER the staff stated that SSAR Section 6.2.6.3.1 did not address its concern. This issue was identified as Open Item 6.2.6-6 in the DFSER.

In its submittal of March 5, 1993, GE added Section 6.2.7.2 (Amendment 32) to the SSAR to state that the COL applicant will maintain the primary containment boundary by administrative controls in accordance with SSAR Section 6.2.6.3.1. SSAR Section 6.2.6.3.1 (Amendment 34) states, in part, that all test connections, vent lines, or drain lines consisting of double barriers (e.g., two valves in series, one valve and a cap, or one valve and a flange), which are connected between isolation valves and form a part of the primary containment boundary, need not be Type C tested because of their infrequent use and multiple barriers as long as the barrier configurations are maintained using an administrative control program. These lines will be surveillance inspected at cold shutdown and at 31-days intervals (internal and external to the primary containment, respectively) as required by the TS. The staff finds that these test connections are designed with multiple barriers, are under an administrative control program, and will be surveillance inspected so that any leakage can be

detected. In addition, Appendix J does not specify test requirements for these lines. This is acceptable. DFSE Open Item 6.2.6-6 is resolved.

- (2) GE relies on closed loops outside the containment as containment isolation barriers for some engineered safety feature (ESF) system containment penetrations. The staff's position is that a closed loop outside the containment that meets the criteria of Section 3.6 of ANSI/American Nuclear Society 56.2-1976 can be considered a second containment isolation barrier, thereby eliminating the need for a second containment isolation valve at the penetration. However, each barrier (i.e., the single isolation valve at each penetration and closed piping loop outside the containment) is subject to leak rate testing. Inclusion of this position in the SSAR was identified as Open Item 6.2.6-7 in the DFSE.

In its submittal of April 26, 1993, GE stated that there are two closed-loop systems in the ABWR design that provide an extension of the containment. These are the containment atmosphere monitoring system and the fuel pool cooling system. These systems will be leak tested by opening their lines to the containment atmosphere during the containment ILRT. The staff finds that their test requirements have been specified in SSAR Table 6.2-7 (Amendment 32), and are acceptable. DFSE Open Item 6.2.6-7 is resolved.

- (3) In response to the staff's RAI Q430.52b of July 7, 1988, GE submitted information regarding secondary containment inleakage and potential bypass leakage paths. However, it did not indicate whether the bypass leakage paths will be leak tested as specified in BTP CSB 6-3, "Determination of Bypass Leakage Paths in Dual Containment Plants," of the SRP. The staff found that SSAR Table 6.2-10, Table 6.5-2, and Section 6.5.1.3.2 did not address this issue adequately. Clarification of this issue was identified as Open Item 6.2.6-8 in the DFSE.

In response to this issue, GE identified all the potential leakage paths that could bypass the secondary containment in revised SSAR Table 6.2-10. The staff reviewed SSAR Table 6.2-10 (Amendment 34) and finds that the potential bypass paths are the main steamlines, feedwater lines, drywell purge suction and exhaust lines, and wetwell purge suction and exhaust lines. The staff noted that the leakage rate through the main steam isolation valves (MSIVs) has been considered as the secondary containment bypass

leakage rate and has been treated separately in the offsite dose analysis. Leakage through feedwater isolation valves (FWIVs) will be monitored by Type C test. Leakage from the purge lines also will be monitored by Type C test and can be detected by the radiation detectors at the purge and vent exhaust lines. In addition, SSAR Table 6.5-2, regarding source terms used for the standby gas treatment system (SGTS) charcoal adsorber design, specifies that leakage rates assumed for calculation are 0.5 percent per day for the primary containment and 50 percent per day for the secondary containment.

Additionally, primary containment leakage could circumvent the secondary containment and bypass the leakage collection and filtration systems. This leakage is generally not quantifiable through LLRTs, but must be considered in the radiological consequence analysis of a DBA. SSAR Section 6.5.1.3.2 (Amendment 34) states that, for the ABWR dose analysis, direct transport of containment leakage to the environment was assumed for the first 20 minutes after initiation of a LOCA (in addition to the leakage through the MSIVs). Each SGTS fan is sized to establish a continuously negative pressure within 10 minutes after SGTS initiation.

The staff finds that GE has provided all the necessary measures to control bypass leakage in the ABWR design in conformance with BTP CSB 6-3 and provide acceptable margins in the offsite dose analysis. This is acceptable. DFSE Open Item 6.2.6-8 is resolved.

- (4) GE did not provide procedures for factoring potential contributions from the hydrogen recombiners into the ILRT results. However, the staff considered this to be a COL action item. In the DFSE, the staff stated that it would ensure that COL applicants factored the potential contributions from the hydrogen recombiners into the ILRT results in accordance with SRP Section 6.2.6. This was identified as COL Action Item 6.2.6-1.

In response to this concern, GE added note (v) in SSAR Table 6.2-7 (Amendment 32) to state that the flammability control system (FCS) is a closed-loop, safety-grade system required to be functional following an accident. Whatever leakage (if any) will be returned to the primary containment. In addition, during an ILRT, these valves will be opened and the lines will be subjected to Type A tests. The staff finds that any leakage from the

Engineered Safety Features

penetrations or the lines connected to the hydrogen recombiners will be included in the ILRT results. Therefore, COL Action Item 6.2.6-1 is deleted and will not be included in the SSAR.

In Bulletin 82-04, "Deficiencies in Primary Containment Electrical Penetration Assemblies," was identified as a candidate operating experience issue. This bulletin addresses potential generic safety concerns related to electrical penetration assemblies supplied by the Bunker Ramo Company. However, the assemblies are no longer manufactured by Bunker Ramo and are not used in the ABWR design. Therefore, this operating experience issue is not applicable to the ABWR design.

GE submitted the design description and the ITAAC relating to containment leakage testing. The staff's review of this material was not complete when the DFSER was issued. This was identified in the DFSER as Open Item 6.2.6-9. GE subsequently provided a revised design description and ITAAC. The adequacy and acceptability of the design description and the ITAAC are evaluated in Section 14.3 of this report. On the basis of this evaluation, DFSER Open Item 6.2.6-9 is resolved.

On the basis of this review, the staff concludes that the ABWR containment leakage testing program complies with Appendix J to 10 CFR Part 50. Such compliance provides adequate assurance that leaktight integrity of the containment can be verified periodically throughout its service life to ensure that leakage rates are maintained within the limits of the TS. Maintaining containment leakage rates within such limits provides reasonable assurance that, in the event of any radioactivity releases within the containment, the loss of the containment atmosphere through the leak paths will not be in excess of the acceptable limits specified for the site. Compliance with Appendix J as described in this section constitutes an acceptable basis for satisfying the requirements of GDC 52 with respect to the capability for containment leakage rate testing, the requirements of GDC 53 with respect to provisions for containment testing and inspection, and the requirements of GDC 54 with respect to the capability for detecting piping leakages.

6.2.7 Fracture Prevention of Containment Pressure Boundary

The staff reviewed the ABWR measures involving fracture prevention of ferritic materials used in the containment pressure boundary in accordance with SRP Section 6.2.7. Containment pressure boundary ferritic materials are acceptable if they meet the requirements of GDC 51 as it relates to the reactor containment pressure boundary being designed with sufficient margin to assure that under

operating, maintenance, testing, and postulated accident condition the ferritic materials will behave in a nonbrittle manner and the probability of rapidly propagating fracture is minimized.

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

In RAI Q251.12, the staff requested that GE clarify the applicability of GDC 51 because it appeared that GE intended that GDC 51 be applied to the concrete portion of the containment. GE responded that GDC 51 is applicable to the removable drywell head, personnel locks, equipment hatches, and penetrations, which will be made of ferritic materials. GE responded satisfactorily to the staff's request and revised SSAR Section 3.1.2.5.2.2 accordingly. Therefore, GE's commitment to GDC 51 for the items listed above in the containment design meets SRP Section 6.2.7 and is acceptable.

6.2.8 Severe Accident Considerations

GE addresses containment performance during severe accidents in SSAR Chapter 19, and the staff's review is documented in Chapter 19 of this report.

6.3 Emergency Core Cooling System

The staff reviewed the emergency core cooling system (ECCS) in accordance with SRP Section 6.3. The staff acceptance criteria are based on meeting the relevant requirements of the following regulations:

- (1) GDC 2 as it relates to the seismic design of structures, systems, and components (SSC) whose failure could cause an unacceptable reduction in the capability of the ECCS to perform its safety function. Acceptability is based on meeting Position C2 of RG 1.29.
- (2) GDC 4 as related to dynamic effects associated with flow instabilities and loads (e.g., water hammer).
- (3) GDC 5 as it relates to SSC important to safety shall not be shared among nuclear power units unless it can be demonstrated that sharing will not impair their ability to perform their safety function.

- (4) GDC 17 as it relates to the design of the ECCS having sufficient capacity and capability to assure that specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded and that the core is cooled during anticipated operational occurrences and accident conditions.
- (5) GDC 27 requires that the reactivity control systems have a combined capability, in conjunction with poison addition by the emergency core cooling system, or reliably controlling reactivity changes under postulated accident conditions, with appropriate margin for stuck rods.
- (6) GDC 35, 36, and 37 as they relate to the ECCS being designed to provide an abundance of core cooling to transfer heat from the core at a rate so that fuel and clad damage will not interfere with continued effective core cooling, to permit appropriate periodic inspection of important components, and to permit appropriate periodic pressure and functional testing.
- (7) 10 CFR Part 5, Subsection 50.46, and Appendix K to 10 CFR Part 50 as it relates to the ECCS being designed so that its cooling performance is in accordance with an acceptable evaluation model.

The ECCS is designed to provide coolant inventory to the reactor coolant system in the event of a loss-of-coolant accident (LOCA) in the pressure boundary. The ECCS capability extends to failures as large as a double-ended rupture of the largest piping carrying water or steam, and spurious safety/relief valve operation.

6.3.1 System Description

The ECCS consists of the following:

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

Unlike that in current BWR designs, the RCIC system in an ABWR design is a part of the ECCS. The initiation logic is diversified by adding a high drywell pressure input as well as maintaining the typical system initiation on reactor pressure vessel (RPV) level 2.

The RCIC system is a high-pressure reactor coolant makeup system that will start independently of the ac power supply. The system will provide sufficient water to the reactor vessel to cool the core and to maintain the

reactor in a standby condition if the vessel becomes isolated from the main condenser and experiences a loss of feedwater flow. The system also is designed to maintain reactor water inventory, in the event of a loss of normal feedwater flow, while the vessel is depressurized to the point where the residual heat removal (RHR) system can function in the shutdown cooling mode at a reactor pressure of 1,034 kPa (150 psig).

The RCIC system consists of a steam-driven turbine-pump unit and associated valves and piping capable of delivering 3,028 L/min (800 gpm) of makeup water to the reactor vessel through the feedwater system. Fluid removed from the reactor vessel following a shutdown from power operation normally will be made up by the feedwater system and supplemented by inleakage from the control rod drive system. If the feedwater system becomes inoperable, the RCIC system will start automatically when the water level in the reactor vessel reaches the level 2 (L2) trip set point or drywell pressure reaches the high drywell pressure trip set point. The RCIC system also can be started by the operator from the control room. The system is capable of delivering rated flow within 29 seconds of initiation. Primary water supply for the RCIC system will be from the condensate storage tank (CST), and a secondary supply will be from the suppression pool.

A detailed evaluation of the RCIC system is given in Section 5.4.6 of this report.

The HPCF system will maintain the reactor vessel water level above the top of the active core in the event of a break of a 2.54 cm (1 in.)-diameter pipe or smaller and will provide cooling in the event of large-pipe breaks. Actuation of the HPCF system will not require the depressurization of the reactor vessel. The HPCF system consists of two loops. Each loop includes a single motor-driven centrifugal pump that will take suction from the CST or the primary containment suppression pool. An automatic switching feature is based on indication of low CST level. The HPCF flow rate is dependent on the reactor pressure. SSAR Table 6.3-1 states that the rated HPCF flow of 12,113 L/min (3,200 gpm) will be attained at a reactor pressure of approximately 689 kPa (100 psig), which is consistent with accident analysis assumptions. The HPCF system is designed to operate from normal offsite auxiliary ac power or from the emergency diesel generators. Each HPCF pump will be powered from a different diesel generator. The system will initiate automatically by either low water level 1.5 or high drywell pressure signals. The system also can be placed in operation manually from the main control room and the remote shutdown panel (loop B only).

Engineered Safety Features

If the RCIC and HPCF systems do not maintain the reactor water level, the ADS will reduce the reactor pressure so that flow from the RHR system, which will be operating in the LPFL mode, enters the reactor vessel in time to cool the core and limit fuel cladding temperature.

Of the 18 nuclear system pressure relief valves, eight ADS relief valves will relieve high-pressure steam to the suppression pool. The evaluation of the pressure relief valves is given in Section 5.2.2 of this report. The ADS will be actuated when the following conditions are satisfied: (1) drywell high pressure, (2) reactor low water level (level 1), and (3) a permissive signal of high RHR or HPCF pump discharge pressure. A time delay of 29 seconds will be used to confirm that the low water level 1 signal is present and is consistent with the ECCS pump startup time. The instrumentation and controls for the ADS are discussed in Chapter 7 of this report.

The LPFL system will replace reactor vessel water inventory following large-pipe breaks. The system is part of the RHR system, which consists of three independent loops (A, B, and C). Each loop has a motor-driven pump with a capacity of 15,900 L/min (4,200 gpm) that will take suction from the suppression pool and supply water to the reactor vessel. All three RHR loops include heat exchangers that will be cooled by the RHR reactor building cooling water system and transfer the decay heat from the reactor core to the ultimate heat sink. The three LPFL (RHR) pumps will be powered from ac power buses that have standby backup sources of power. RHR pumps A, B, and C will receive emergency power from the three separate diesel generators. The RHR system valve logic will require LPFL system alignment in the event of a LOCA. The LOCA event takes precedence over other RHR system functional modes. The system will initiate automatically by either low water level 1 or high drywell pressure signals. The reactor must be depressurized below the reactor low-pressure permissive signal before LPFL injection to the reactor occurs. Each of the two high-pressure core flooding loops and two of the three low-pressure flooding loops will discharge water into the core through a separate overhead flooders sparger. Low-pressure flooding loop A will discharge into the RPV through the feedwater system. Internal vessel piping connects each sparger to the vessel nozzle. The ABWR flooders design and relative location will result in reduced personnel radiation exposure as compared to the current BWR core spray design because the peripheral location of the flooders minimizes the need for work over the fuel during inservice inspection. Spray distribution in the core is not critical because there will be no core uncover during a LOCA.

6.3.2 Evaluation of Single Failures

The staff reviewed the SSAR system description and piping and instrument drawings to ensure that abundant core cooling will be provided during the injection phase with and without offsite power and assuming a limiting single failure as required by GDC 35. A low reactor vessel water level and/or containment high pressure signal is required to start pumps and open discharge valves.

GE provided in SSAR Section 6.3.3 an analysis to demonstrate that the most limiting break size, break location, and single failure had been considered for the ABWR. The most limiting combinations are given in Table 6.1 of this report. The staff finds that the SSAR information supports the finding that the ECCS systems meet the single-failure criterion.

6.3.3 Qualification of Emergency Core Cooling System

The ECCS is designed to meet seismic Category I requirements in compliance with RG 1.29, "Seismic Design Classification," (Rev. 3), as discussed in Section 3.2 of this report. The ECCS is housed in structures designed to withstand seismic events, tornados, floods, and other phenomena, in accordance with the requirements of GDC 2, as discussed in Section 3.2.1 of this report. ECCS equipment is designed in compliance with RG 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," (Rev. 4), as discussed in Section 3.2 of this report.

Protecting the ECCS against pipe whip and against discharging fluids, in compliance with the requirements of GDC 4, is discussed in Section 3.6 of this report. Evaluation of the instrumentation and controls for the ECCS, including compliance with RG 1.47, "Bypass and Inoperable Status Indication for Nuclear Power Plant Safety Systems," in accordance with the applicable requirements of GDC 27, is discussed in Section 7.3 of this report. Compliance with the inservice inspection requirements of GDC 36 is discussed in Section 6.6 of this report.

Environmental qualification of the ECCS equipment for operation under normal and accident conditions, as required by GDC 4, is discussed in Section 3.11 of this report.

Table 6.1 Single failure evaluation

Assumed Failure	Systems Remaining
Emergency diesel generator A	All ADS, RCIC, 2 HPCF, 2 RHR/LPFL
Emergency diesel generator B or C	All ADS, RCIC, 1 HPCF, 2 RHR/LPFL
RCIC injection valve	All ADS, 2 HPCF, 3 RHR/LPFL
One ADS valve	All ADS minus one, RCIC, 2 HPCF, 3 RHR/LPFL

Note: ADS = automatic depressurization system
HPCF = high-pressure core flooder
LPFL = low-pressure flooder
RCIC = reactor core isolation cooling
RHR = residual heat removal

In accordance with the pertinent requirements of GDC 35, the available net positive suction head (NPSH) for the pumps in the ECCS, as reflected in calculations submitted by GE, is adequate to prevent cavitation and ensure pump operability in accordance with RG 1.1, "Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps (Safety Guide 1)," (Rev. 0). The pump head and NPSH requirements for ECCS pumps are included in the ITAAC for verification by the COL applicant.

Each of the low-pressure lines that will discharge into the reactor coolant system has a testable check valve inside the primary containment backed up by a normally closed motor-operated gate valve outside the containment. Relief valves in the low-pressure lines will protect against leakage from the reactor coolant system. An interlock on the motor-operated valves will prevent them from opening until the reactor coolant pressure is below the low-pressure ECCS design pressure.

Containment isolation in accordance with the requirements of GDC 55 is discussed in Section 6.2 of this report. The periodic testing and leak-rate criteria for those valves that isolate the reactor system from the ECCS are discussed in Section 3.9.6 of this report. Detection of leaks from those portions of the ECCS within the primary coolant pressure boundary is discussed in Section 5.2.5 of this report.

To protect the pumps from overheating, all the ECCS pumps have minimum flow bypass lines to permit a limited amount of flow if an isolation valve between the reactor coolant system and the ECCS is closed for any reason.

When flow in the injection lines is sufficient for pump cooling, valves in the minimum flow bypass lines will close automatically, diverting all flow to the pressure vessel. Each LPFL pump suction line from the suppression pool has an open motor-operated valve outside the containment. The suction line of the HPCF from the suppression pool contains a closed motor-operated valve so the HPCF initially will draw water from the CST. When the CST water is exhausted, the suppression pool suction valve will open automatically.

Isolation of the suppression pool from the reactor building in accordance with GDC 56 is discussed in Section 6.2 of this report.

As a backup to the HPCF system, the ADS will be used to depressurize the reactor and allow the LPFL to function in the event of a small break. Nitrogen will be supplied to the valves of the ADS from seismically qualified accumulators.

One of the design requirements of the ECCS is that cooling water flow be provided rapidly following the initiation signal. By always keeping the ECCS pump discharge lines full, the lag time between the signal for pump start and the initiation of flow into the RPV can be minimized. In addition, full discharge lines reduce potentially damaging waterhammer occurrences on system startup. The RHR system has three jockey pumps, one in each loop (discharge line fill pump). Maintaining the filled status of the system is ensured by continuous indication of pump operation and pump discharge pressure. The make-up water system connections to the RCIC and HPCF

Engineered Safety Features

system will maintain the pump discharge line in a filled condition up to the injection valve.

The capability of the ECCS pumps to operate for an extended period of time during the long-term recirculation phase following a LOCA is discussed in Section 3.9.6.1 of this report.

Safety/relief valve (SRV) operability will be demonstrated during the power ascension phase of the plant startup test program by manually actuating each SRV (including the ADS valves) one at a time to show that no blockage exists in the valve discharge line. After commercial turnover, all of the SRVs will be tested in accordance with Section XI, Article IWV, of the ASME Code. Linear variable differential transformers and thermocouples are to be installed in the discharge line of each SRV to monitor valve position and SRV leakage in accordance with TMI-2 Action Plan Item II.D.3 (NUREG-0737) and are discussed in Section 20.5.23 of this report.

Environmental qualification of the SRVs is discussed in Section 3.11 of this report.

The ADS time delay is set at 29 seconds rather than 120 seconds (as in present BWRs) to satisfy the design goal of no core uncover during a LOCA. The ADS most likely will not be initiated because of the availability of three high-pressure core injection systems. The ABWR emergency procedures guidelines (EPGs) (which are provided in SSAR Chapter 18) states that the operator is allowed to prevent ADS actuation in two instances. The first is during an anticipated transient without scram (ATWS) event. Because the ABWR design incorporates an automatic ADS inhibit signal following an ATWS, operator action is to verify the automatic ADS inhibit signal as addressed in SSAR Chapter 18 (EPGs). The second case is when the operator believes that the reactor water level may go below the reactor low water level 1 setpoint but will remain above the top of the active fuel (TAF) without ADS actuation. Because no ADS timer is provided for LOCAs in the containment, where high drywell pressure occurs, and because the ADS time delay is short, operator intervention is unlikely. However, during LOCAs outside the containment or LOCAs that do not result in a high drywell pressure signal, an 8-minute timer will be initiated and the EPGs allow the operator to assess whether the level is maintained above TAF. If the high-pressure ECCS cannot control the reactor water level, it is prudent to allow the ADS to actuate and quickly depressurize the vessel to gain the additional reflooding capacity of the low-pressure ECCS. The EPGs provide sufficient guidance to ensure that operating procedures will

reflect the importance of allowing ADS actuation when high-pressure makeup systems are not available.

For transient and accident events that do not directly produce a high drywell pressure signal (e.g., stuck-open relief valve or steamline break outside the containment) and that are further complicated by the loss of all high-pressure ECCS systems, manual activation of the ADS was originally required to provide adequate core cooling. However, TMI Action Plan Item II.K.3.18 (NUREG-0737), as incorporated in 10 CFR 50.34f(1)(vii), requires ADS logic modification to eliminate operator action. GE's proposed design modification is consistent with Option 4 of the BWR Owner's Group response to TMI Action Plan Item II.K.3.18. This option requires the addition of a timer that bypasses the high drywell pressure permissive if the reactor water level is low for a sustained period and the addition of a manual inhibit switch. An 8-minute high drywell pressure bypass timer has been added to the ABWR ADS initiation logic. (An analysis was performed to evaluate the adequacy of the 8-minute bypass timer.) This timer will initiate on a low water level 1 signal. When the timer runs out, it will bypass the need for a high drywell pressure signal to initiate the standard ADS initiation logic. The bypass timer will be tested periodically. This test is required by the TS. This was DFSE TS Item 6.3.3-1. (This item has been added to the ABWR TS and has been found to be acceptable.) This timer can be inhibited as indicated in the ABWR EPGs if the operator determines that water level can be maintained above TAF. The proposed GE modification to ADS logic is consistent with the staff-approved resolution of Item II.K.3.18 and is acceptable because it ensures the initiation of the ADS and low-pressure injection systems when high drywell pressure does not occur. In the DSER (SECY-91-153) the staff identified a concern about the ADS in Open Item 28. The above discussion on bypass timers shows that DSER Open Item 28 has been resolved. Assurance of the proposed modification to the ADS logic was DFSE Confirmatory Item 6.3.3-1. GE submitted the required changes in the SSAR in Section 6.3.3.4, which is acceptable to the staff. GE also included this action item in the SSAR. This Confirmatory Item 6.3.3-1 is resolved.

6.3.4 Testing

RG 1.68, "Initial Test Programs for Water-Cooled Nuclear Power Plants," (Rev. 2) and GDC 37 require that the ECCS system be designed to permit appropriate periodic pressure and functional testing to assure the integrity of components, and the operability and performance of the components and ECCS system. GE states that ECCS integrity, operability, and performance are demonstrated by preoperational and periodic testing, which is acceptable to the staff.

6.3.4.1 Preoperational Tests

Preoperational tests will ensure proper functioning of controls, instrumentation, pumps, piping, and valves. Pressure differentials and flow rates are measured for later use in determining acceptable performance in periodic tests. GE commits to conform to the guidelines of RG 1.68 for preoperational and initial startup testing of the ECCS, as noted in Section 14.2 of this report.

6.3.4.2 Periodic Component Tests

The staff stated in the DFSE that the TS should state that the ECCS subsystems (except for the ADS) will be tested periodically to show that specified flow rates are attained. This was DFSE TS Item 6.3.4.2-1. GE has included this statement in the TS. The staff also stated in the DFSE that the COL applicant should perform a test every refueling outage in which all subsystems are actuated through the emergency operating sequence. This was DFSE COL Action Item 6.3.4.2-1. GE provided this procedural COL action item in Amendment 31 of the SSAR.

6.3.5 Performance Evaluation

GE's ABWR-specific LOCA analysis demonstrates that the reference fuel design meets the requirements of 10 CFR 50.46 and is based on the initial core design.

The staff expected the exposure-dependent maximum average planar linear heat generation rate (MAPLHGR) to be provided in the TS. This was DFSE TS Item 6.3.5-1. GE submitted the MAPLHGR in TS 3.2.1; hence this item is resolved. In Section 4.2 of this report, the specific fuel and core design used for the analyses of SSAR Sections 6 and 15 is described. Section 4.2 of this report also describes fuel design criteria that specify requirements for any alternative fuel that may be used by the COL applicant or licensee in reload cycles.

The staff used the information in the SSAR, the results of the LOCA analyses, along with the results of its review of each ECCS system to verify that the proposed ECCS meets the performance criteria in 10 CFR 50.46. Compliance with the first three criteria below is demonstrated analytically. Coolable geometry is maintained if the first two criteria are met. Long-term cooling capability is verified by the composite review of the ECCS and the various support systems. The five acceptance criteria for the ECCS, as specified in 10 CFR 50.46, are the following:

- (1) The calculated maximum peak cladding temperature (PCT) shall not exceed 1204 °C (2200 °F).
- (2) The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation.
- (3) The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding plenum volume, were to react.
- (4) Calculated changes in core geometry shall be such that the core remains amenable to cooling.
- (5) The calculated core temperature shall be maintained at an acceptably low value, and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core after any calculated successful initial operation of the ECCS.

GE has demonstrated compliance with the first three of these criteria as shown in Table 6.2 of this report.

Table 6.2 Demonstration of compliance with ECCS criteria

Criterion	Maximum	
	From Break Analyses	Allowable
Peak cladding temperature, °C (°F)	621 (1149)	1204 (2200)
Maximum cladding oxidation, %	0.03	17
Maximum total hydrogen generation, %	0.03	1

Engineered Safety Features

There is no core uncover and hence the core geometry is maintained. Moreover, a coolable geometry was demonstrated by compliance with the criteria for PCT and maximum cladding oxidation as discussed in NEDE-20566, "Analytical Model for Loss-of-Coolant Analysis in Accordance with 10 CFR Part 50, Appendix K," proprietary, November 1975.

Because there are no jet pumps in the ABWR, the core flow pattern is similar to that in BWR/2 plants that do not use jet pumps (e.g., Oyster Creek, Nine Mile Point Unit 1). The GE thermal-hydraulic code SAFER, which was approved for BWR/2 plants without jet pumps (NEDE-30996P-A, "SAFER Model for Evaluation of Loss-of-Coolant Accidents for Jet Pump and Non-Jet Pump Plants," (Volumes I and II) October 1987), was used to perform LOCA analysis for the ABWR. The SAFER code was used to calculate the long-term thermal-hydraulic behavior of the coolant in the vessel during a LOCA. Some important system parameters calculated by SAFER are vessel pressure, vessel water level, and ECCS flow rates. The SAFER code also calculates PCT and local maximum oxidation. The staff's SER (dated February 19, 1987, "Review of NEDE-30996(P), 'SAFER Models for Evaluation of Loss-of-Coolant Accident for Jet Pump and Non-Jet Pump Plants,' Volumes I and II") on NEDE 30996A-P documents agreement between SAFER predictions and test results from the ABWR full integral simulation test (FIST) facility. GE made some modifications to the FIST facility for the ABWR and conducted additional tests. Jet pump elimination and reactor internal pump (RIP) flow were simulated. GE also modified ECCS capacities, break sizes, and locations. The SAFER code was compared to TRACG-P (proprietary) for the ABWR plant. (See staff SER of February 19, 1987.)

ABWR-specific data based on tests for the RIP coastdown time following a pump trip and flow area and pressure losses through the RIPs were used as input to the SAFER model.

GE computer codes LAMB, SCAT, and GESTR were used for the ABWR LOCA analysis.

The LAMB code is used to analyze the short-term thermodynamic and thermal-hydraulic behavior of the coolant in the vessel during a postulated LOCA. In particular, LAMB predicts the core flow, core inlet enthalpy, and core pressure during the initial phase of the LOCA event (i.e., the first 5 seconds). GE used the LAMB computer model (documented in NEDE 20566P-A, "Analytical Model for Loss-of-Coolant Accident Analysis in Accordance with 10 CFR Part 50, Appendix K," January 1976, and approved by the staff) to perform

analyses presented in the ABWR SSAR. The ABWR input to the LAMB computer model is the same input as that used for operating plants except for recirculation system modeling. Because the ABWR has RIPs instead of jet pumps, the recirculation system is modeled assuming no induced flow from a jet pump. Thus, the drive flow from the recirculation pumps is set equal to the total recirculation flow.

The SCAT code is used to evaluate the short-term thermal-hydraulic response of the coolant in the core during a postulated LOCA. Using the LAMB results as input, SCAT analyzes the convective heat transfer process in the thermally limiting fuel bundle during the initial phase of the LOCA event. In particular, SCAT predicts the departure from nucleate boiling at any one of the 24 axial nodes in the fuel bundle. GE used the SCAT computer model (documented in NEDE 20566P and approved by the staff) to perform analyses presented in the ABWR SSAR. Specifically, GE used SCAT to determine the transient thermal-hydraulic conditions within a bundle and predict the time when the loss of nucleate boiling occurs. The SCAT program only models the hot bundle and sets the boundary conditions based on LAMB results. The ABWR core design is based on a GE standard fuel that is similar to BP 8 x 8R with two water rods. This type of fuel has already been analyzed using SCAT for operating BWRs. Therefore, the differences in the ABWR from previous BWRs do not represent any special application of the SCAT model.

The GESTR code is used to provide best-estimate predictions of the thermal performance of GE nuclear fuel rods experiencing variable power histories. For LOCA analysis, the GESTR code is used to initialize the fuel stored energy and fuel rod fission gas inventory at the onset of a postulated LOCA. GE used the GESTR computer model (documented in NEDE 30996P-A and approved by the staff) to perform LOCA analyses presented in the ABWR SSAR. GE used the results from GESTR to establish the initial conditions (i.e., stored energy and rod internal pressure) at the start of the LOCA within each fuel rod. As stated above, GE based its ABWR core design on a GE standard fuel, which also has been analyzed using GESTR for operating BWRs. The differences in the ABWR from previous BWRs do not affect the fuel thermal-mechanical performance predicted by the GESTR model. Therefore, the use of GESTR for the ABWR does not represent any special application of the GESTR model.

No model changes were made to any of the GE codes used in the LOCA analysis; only input data were changed for ABWR-specific design features.

The GE analyses included break sizes ranging from a bottom head drain line break 0.0020 m^2 (0.0218 ft^2) to the main steamline break outside the containment 0.39 m^2 (4.24 ft^2). (Since the bottom head drain line will tie into the RHR/cleanup system lines, the total break flow for the maximum bottom head drain line break includes flow from the vessel through the bottom head drain line penetration as well as through the RHR/cleanup system lines.) GE analyzed different break sizes in conjunction with ECCS failure combinations. The cases were evaluated to establish the trend of PCT curves (Appendix K to 10 CFR Part 50 and bounding values) versus break size. Eight break sizes are summarized in SSAR Table 6.3-4. The most limiting break is the main steamline break outside the containment, which results in a PCT of 621°C (1149°F). This is well below the 1204°C (2200°F) acceptance criterion. Because there will be no large pipe below the top of the core and no core uncover for any size LOCA, the calculated PCT of 621°C (1149°F) is low compared to the PCT for current BWRs.

The staff confirmed that the LOCA analysis methodology used by GE for the ABWR was consistent with and bounded by the staff's generically approved LOCA analysis methodology. The staff concluded that the PCT, peak local oxidation, and core-wide metal-water reaction values were well below staff acceptance criteria. Thus, the analyses and results are in accordance with NRC requirements and GE has demonstrated conformance with the ECCS acceptance criteria of 10 CFR 50.46 and Appendix K to 10 CFR Part 50.

Long-term cooling is ensured by the use of redundant RHR systems that have adequate water sources available to remove the decay heat generated in the reactor core and transfer the heat to the ultimate heat sink. GE identified no single failure that would prevent the ECCS from meeting this criterion. The systems are designed to prevent any core uncover.

The LPFL flow will be diverted manually to wetwell spray cooling for containment pressure control. The ABWR EPGs require cautions in the emergency operating procedures to deter the operator from premature flow diversion. These guidelines, which caution the operator against diversion unless adequate core cooling is ensured, have been accepted by the staff (NUREG-0737, TMI-2 Action Plan Item I.C.1, discussed in Chapter 20 of this report). LPFL diversion is identified in the procedure as secondary to core cooling requirements, except in those instances outside the design envelope involving multiple failures for which maintenance of containment integrity is required to minimize risk to the environment.

The ECCS meets GDC 17 in that its capacity and capability are sufficient to ensure that acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of anticipated operational occurrences; and the core will be cooled and vital functions will be maintained in the event of postulated accidents as given in SRP Section 6.3.

6.3.6 Certified Design Material

GE submitted the design description and the inspections, tests, analyses, and acceptance criteria (ITAAC) for the HPCF system. This was DFSE Open Item 6.3.6-1. GE subsequently provided a revised design description and ITAAC. The adequacy and the acceptability of the ABWR design descriptions and ITAAC are evaluated in Section 14.3 of this report. On the basis of this evaluation, this item is resolved.

6.3.7 Conclusions

As discussed above, the staff finds the design of the ECCS acceptable. The ECCS conforms with the review guidelines and acceptance criteria of SRP Section 6.3 and its pertinent RGs. Therefore, the ECCS meets the performance criteria of 10 CFR 50.46 and the pertinent requirements of GDC 2, 4, 17, 35, 36 and 37 as set forth in the SRP, as summarized below:

- (1) 10 CFR 50.46 and Appendix K performance requirements, as described in Section 6.3.5 of this report.
- (2) GDC 2, as it relates to the seismic design of SSC whose failure could cause an unacceptable reduction in the capability of the ECCS to perform its safety functions, as discussed in Section 3.2.1 of this report.
- (3) GDC 4, as related to dynamic effects associated with flow instabilities and loads, as discussed in Section 3.6 of this report.
- (4) GDC 5 is not applicable since the ABWR is a single unit plant.
- (5) GDC 17, as it relates to the ECCS design for sufficient capacity and capability to assure that specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded and that the core is cooled during anticipated operation occurrences and accident conditions, as discussed in Section 6.3.5 of this report.

Engineered Safety Features

- (6) GDC 27, as it relates to the reactivity control systems to have a combined capability, in conjunction with poison addition by the emergency core cooling system, or reliably controlling reactivity changes under postulated accident conditions, with appropriate margin for stuck rods is discussed in Sections 4.2, 4.6 and 9.3.5 of this report.
- (7) GDC 35, 36, and 37, as they relate to the ECCS design to provide an abundance of core cooling to transfer heat from the core at a rate so that fuel and clad damage will not interfere with continued effective core cooling, to permit appropriate periodic inspection of important components, and to permit appropriate periodic pressure and functional testing, as discussed in Sections 6.3.2, 6.3.4, and 6.6 of this report.

6.4 Control Room Habitability Systems

The staff reviewed the control room habitability systems in accordance with SRP Section 6.4. Conformance with the acceptance criteria formed the basis for the staff's evaluation of the control room habitability systems with respect to the applicable regulations of 10 CFR Part 50. Specifically, the SRP acceptance criteria require the design to meet GDC 4 as it relates to accommodating the effects of and being compatible with the environmental conditions associated with postulated accidents, including the effects of the release of toxic gases; GDC 19 as it relates to maintaining the control room in a safe, habitable condition under accident conditions by providing adequate protection against radiation and toxic gases; additional TMI requirement 10 CFR 50.34(f)(2)(xxviii) as it relates to the evaluation of potential pathways for radioactivity and radiation that may lead to control room habitability problems; and TMI Action Plan Item III.D.3.4 (NUREG-0737) requirements as they relate to providing protection against the effects of the release of toxic substances, either on or off the site. Since the ABWR design is applicable only for a single unit, GDC 5 is not applicable.

The control room habitability systems will provide (1) missile protection, (2) radiation shielding, (3) radiation monitoring, (4) air filtration and ventilation, (5) lighting, (6) personnel and administrative support, and (7) fire protection. The control building heating, ventilating, and air conditioning (HVAC) system and components are located in a seismic Category I structure that is protected from tornado, missile, pressure and flood damage. The HVAC ducting is ESF grade. HVAC hangers are designed to seismic Category I standards. The HVAC system will maintain the control room atmosphere tempera-

ture at a habitable level to permit prolonged personnel occupancy throughout a postulated design-basis accident (DBA). The system design provides for control room pressurization with respect to the surrounding spaces and filtered intake during accident situations and for purging of smoke and toxic gases. The system is capable of automatic transfer from its normal operating mode to its emergency or isolation modes on detection of adverse conditions (e.g., high radiation, smoke). The system has sufficient redundancy to ensure operation under emergency conditions, assuming the single failure of any one active component. Backup power sources are provided for the essential components of the HVAC system. Section 9.4.1 of this report provides more information on the control room HVAC system. The habitability systems are designed to detect and limit the introduction of radioactive material and smoke into the control room. The ABWR design relies on noncombustible construction and heat- and flame-resistant materials throughout the plant to minimize the likelihood of fire and consequent fouling of the atmosphere with smoke or noxious vapor. Further, the number of individual respirators (subject to periodic operational testing) is sufficient to protect against the intrusion of toxic gases into the control room. Non-seismic pipe, ductwork for kitchen and sanitary facilities, and other nonessential components in the control building are designed to ensure that their failure during a safe shutdown earthquake will not adversely affect essential components. Potential sources of danger such as pressure vessels and carbon dioxide firefighting containers will be located outside the control room and the compartments containing control building life support systems. There are no high-energy lines near the control room; therefore, the habitability systems are protected against the dynamic effects that may result from possible failures of such lines. Section 3.6.1 of this report addresses protection of safety-related equipment from the effects of pipe breaks. The staff concludes that the control room habitability systems satisfy GDC 4.

By letter dated July 7, 1988, the staff requested additional information on a number of issues (e.g., makeup air inlet for control room emergency zone, locations of control room ventilation inlets relative to major potential plant release points, radiation protection instrumentation, and minimum positive pressure during the pressurization mode) regarding the design features provided for compliance with GDC 19. This was identified as Open Item 29 in the DSER (SECY-91-153). GE has provided the requested information. The staff has reviewed and finds it acceptable as discussed below. Therefore, DSER Open Item 29 is resolved.

The ABWR main control room (MCR) is physically integrated with the reactor building and turbine building

and is located between these two structures. During and following a LOCA, which is the controlling DBA for the radiological consequences to the control room operators, radiation exposure to the operators will consist of contributions from airborne fission products entrained in the control room ventilation system and direct gamma radiation from the surrounding buildings and process equipment.

GDC 19 requires that the control room be designed to permit personnel to occupy it under accident conditions. GE has proposed that this requirement be met by incorporating shielding and emergency ventilation systems into the control room design, and by two roof-mounted and automatically controlled room air intakes. GE has stated that the structure housing the control room is designed and will be constructed to meet seismic Category I criteria, as is the emergency ventilation system, which is also designed to meet the single-failure criterion.

GE has provided for the radiation protection of control room occupants by the use of shielding walls and by the installation of redundant safety grade emergency ventilation systems. The inhabited portions of the ABWR control room are located underground (4.3 m (14 ft) below grade). A distance of more than 11.3 m (37 ft) (including 2 m (6.5 ft) of concrete shielding and 0.46 m (1.5 ft) of floor) separates the control room and steamlines. GE stated that the expected radiation exposure rate in the control room resulting from direct gamma radiation during normal operation and after an accident is less than 0.006 mSv/hr (0.6 mR/hr). The staff accepted GE's estimate of direct gamma radiation of less than 0.006 mSv/hr (0.6 mR/hr) after an accident because of the decay of short-lived isotopes and the immediate closure of the main steam isolation valves that are upstream of the pipes above the MCR.

If a significant concentration of airborne radioactive materials is detected at the normal control room ventilation system air intake, the air intake will isolate automatically. Automatic control room pressurization will occur immediately, and filtered air will be taken in by either of two separate emergency ventilation systems. Each emergency ventilation system consists of, at a minimum, a 50 mm (2 in.)-thick charcoal adsorber for removal of iodines. The intakes for these systems are separated from each other and from the plant stack. It is not likely that both air intakes would simultaneously admit equivalent levels of radioactive contaminated air.

The viability of the dual-inlet concept depends on whether or not the placement of the inlets ensures that airborne radionuclide concentrations in one inlet will always be relatively low. The capability to ensure that this condition

exists at an inlet depends, in part, on building wake effects, site-specific terrain, and wind stagnation or reversal. For the ABWR design, the inlets are located at the extreme edges of the control building. However, the staff finds that it is possible, under certain low-probability conditions, for both inlets to be drawing air from the same source of radioactive materials, and that the location of the air inlets is less than ideal, since they are not on opposite sides (180°) of potential radiation release points.

In Amendment 8, GE claimed a factor of 4 reduction in the estimate of the atmospheric dispersion parameter for the control room to account for dilution effects associated with the ABWR air inlet configurations and the ABWR-specific building arrangement. SRP Section 6.4 allows a factor of 10 for reduction of the parameters for a dual air intake design with automatic selection control features and with the inlets placed on plant structures on opposite sides. Therefore, the staff stated in the DFSER that GE's proposed reduction factor of 4 (instead of 10) was reasonable and acceptable because the lower reduction factor consecutively accounts for air inlets not being on opposite (180°) sides of potential radiation.

In Amendment 24, GE changed the control room HVAC system design by deleting automatic selection features of the most favorable (less radioactive) air intake from either of two separate emergency air intakes to the control room. In addition, GE revised (1) the filtered emergency air intake flow rate into the control room, (2) the control room air recirculation flow rate, and (3) the control room limiting atmospheric dispersion values provided in Table 15.6-14 of the SSAR. Therefore, the staff disallowed a factor of 4 reduction given in the DFSER for estimating the atmospheric dispersion parameters for the control room, since GE had deleted the automatic selection features of the most favorable air intake from the ABWR design. Even though SRP Section 6.4 allows a factor of 2 for reduction of the parameters for a dual air intake design without manual or automatic selection features, the staff has not provided, and GE did not request in Amendment 24, any reduction factors because the emergency control room air intakes are not placed on opposite sides of potential radiation release points.

The staff recalculated the control room operator doses using the design-basis LOCA described in Section 15.3.1 of this report and revised control room design parameters provided in Amendment 24. The revised staff assumptions and dose estimates are listed in Table 15.10 of this report. The staff concludes that the calculated doses still meet the radiological consequences values of GDC 19 and TMI Action Plan Item III.D.3.4, "Control Room Habitability." The staff further concludes that the ABWR control room design still provides an acceptable means of maintaining

Engineered Safety Features

the control room in a safe and habitable condition by providing adequate protection under accident conditions in accordance with TMI requirement 10 CFR 50.34(f)(2) (xxviii).

GE identified an interface requirement described below for the applicants referencing the ABWR design to protect operators against the effects of the release of toxic substances. However, as a result of further review of all interface items, GE committed to revise the SSAR to reclassify this item as a COL action item.

By letter dated April 16, 1993, GE stated that the ABWR standard plant site design parameters (SSAR Table 2.0-1) did not include toxic gases in the site vicinity. A COL action item has been included in Section 6.4.7 of the SSAR requiring utility applicants referencing the ABWR design to demonstrate that control room operators are adequately protected against the effects of the release of toxic substances, either on or off the site, and that the plant can be safely operated or shut down under conditions created by a DBA. The need for site-specific toxic gas protection will be reviewed to ensure that the control room operators are protected against releases of hazardous material in accordance with TMI Action Plan Item III.3.D.4. The amounts and locations of any possible sources of toxic substances in each plant vicinity will be identified following the methods outlined in RGs 1.78 and 1.95. Specific detectors to permit automatic control room isolation will be provided, where necessary.

In the DFSE, the staff stated that the following were to be included in the ITAAC. The COL applicant will need to verify that the following are consistent with the licensing basis documentation: the asbuilt design; the operating, maintenance, and emergency procedures and training; the performance characteristics of the control room habitability system; and the TS and surveillance procedures. This was identified as Open Item 6.4-1 in the DFSE. GE provided a revised design description and ITAAC. The adequacy and acceptability of the design description and the ITAAC are evaluated in Section 14.3 of this report. On the basis of this evaluation, DFSE Open Item 6.4-1 is resolved.

The staff reviewed the ITAAC for the control room habitability area (CRHA) HVAC system (Table 2.15.5a, Design Commitments 1 through 10) and the control building (Table 2.15.12, Design Commitments 1 through 13) as they relate to control room habitability. In the DFSE, the staff found them to be acceptable with some exceptions. These were identified as DFSE Open Item 6.4.2. GE provided a revised design description and ITAAC. The adequacy and acceptability of the design description and the ITAAC are evaluated in Chapter 14.3

of this report. On the basis of this evaluation, DFSE Open Item 6.4-2 is resolved.

The staff concludes that the control room habitability systems meet the acceptance criteria of SRP Section 6.4 and are, therefore, acceptable.

6.5 Fission Product Removal and Control Systems

6.5.1 Engineered Safety Features Atmosphere Cleanup Systems

The staff reviewed the engineered safety feature (ESF) atmosphere cleanup systems in accordance with SRP Section 6.5.1. Conformance with the acceptance criteria formed the basis for the staff's evaluation of these systems with respect to the applicable regulations of 10 CFR Part 50.

The ABWR design has two ESF filter systems: the emergency air filtration system of the CRHA HVAC system and the standby gas treatment system (SGTS).

SRP Section 6.5.1 acceptance criteria applicable for the emergency air filtration system's high-radiation mode of the CRHA HVAC system are GDC 19 as it relates to the system being designed to ensure habitability of the control room under accident and LOCA conditions and GDC 61 as it relates to the design of the system for radioactivity control under normal and postulated accident conditions. As the staff concluded in Section 6.4 of this report, the emergency air filtration system, which is part of the CRHA HVAC system, complies with GDC 19 radiation exposure limits for the control room operators. The SRP states that the specified acceptance criteria are met by conforming with the guidelines of RG 1.52, "Design, Testing, and Maintenance Criteria for Post Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," Revision 2, and SRP Table 6.5.1-1 regarding the system instrumentation. Therefore, the staff has used system compliance with RG 1.52 and SRP Table 6.5.1-1 as the basis for concluding that the ESF filter system meets the applicable GDC and consequently the applicable SRP acceptance criteria. The above compliance evaluation is discussed in Section 9.4.1 of this report for the emergency air filtration system of the CRHA HVAC system and in this section for the SGTS.

SRP Section 6.5.1 acceptance criteria applicable for the SGTS are GDC 41 as it relates to the design of the system as it will be used for containment atmosphere cleanup following postulated accidents and for the control of releases to the environment, GDC 42 as it relates to the

inspection and testing of the SGTS, GDC 61 (see the above paragraph) and GDC 64 as they relate to monitoring radioactive releases via the SGTS under normal, anticipated operational occurrences and postulated accident conditions. As the staff concludes in Section 11.5 of this report, the system complies with GDC 64.

The staff's evaluation of the two ESF filter systems is given below.

Emergency Air Filtration System: The function of this system is to supply filtered outside air to the main control area envelope (MCAE) and to pressurize the control room after a design-basis accident (DBA). This system is evaluated in Section 9.4.1 of this report. In Section 9.4.1 of the SSAR and the DFSE, the emergency air filtration system of the CRHA HVAC system was erroneously identified as the control building outdoor air cleanup system. GE has corrected this error in SSAR Amendment 20, and SSAR Amendment 32 identified separate and independent discharge and return paths to and from the MCAE for the emergency filtration system.

Standby Gas Treatment System: The primary function of the SGTS is to filter and thereby reduce offsite airborne releases of radioiodine and particulates following a DBA (e.g., LOCA, fuel-handling accident, fuel cask drop accident). This will be accomplished by automatic isolation of the secondary containment from its normal HVAC air paths and automatic actuation of the SGTS on receipt of a LOCA signal, detection of high radiation levels in the secondary containment normal HVAC exhaust or in the refueling floor exhaust or loss of secondary containment HVAC supply/exhaust fans signal. The system can also be manually initiated from the control room (e.g., for de-inerting during power operation) to reduce airborne radioactive iodine and particulate releases. The SGTS will maintain the secondary containment at a negative pressure of 6.4 mm (1/4 in.) of water gauge or greater relative to the surrounding spaces within 20 minutes from the time the secondary containment is isolated and process the effluent gases through a filter train to remove airborne iodines and particulates. The system consists of two identical, parallel, physically separated, 100-percent-capacity subsystems, 6,800 m³/hr (4,000 cfm) each, with associated piping and ducts, valves, dampers, and controls. Each subsystem consists of a moisture separator, electric process heater, prefilter, pre-high-efficiency particulate air (HEPA) filter, charcoal adsorber (15-cm (6 in.) depth), post-HEPA filter, process fan (6,800 m³/hr (4,000 cfm) at 1 atmosphere and 21 °C (70 °F)) and cooling fan (700 m³/hr (412 cfm) at 1 atmosphere and 21 °C (70 °F)). IE Bulletin 80-03 has been met since GE has revised SSAR Section 6.5.1.3.3 to state that the charcoal tray and screen will be all welded con-

struction in accordance with the bulletin. Therefore, the SGTS filter trains will preclude the possible loss of charcoal from adsorber cells. The system is designed to seismic Category I requirements and will be housed in a seismic Category I structure.

In the DSER (SECY-91-153), the staff stated that Appendix 6A to Chapter 6 of the SSAR demonstrated the system's compliance with each of the regulatory positions in RG 1.52, (Rev. 2), except one relating to the requirement for redundancy in filter trains. Originally, the SGTS design included only a single filter train. In the DSER (SECY-91-153), the staff identified the use of a single filter train in the SGTS as Open Item 30. Before the DFSE was issued, GE submitted in a letter dated February 13, 1992, piping and instrumentation diagrams (P&ID's) for a number of systems including the SGTS. The SGTS P&ID showed two filter trains. Therefore, in the DFSE, DSER Open Item 30 was resolved. However, as a followup to GE's proposed design change, the staff required GE to confirm the design change by (1) submitting layout drawings showing the locations of the two filter trains and explaining the adequacy of the spatial separation between the trains, and (2) revising SSAR Section 6.5.1, Table 6.5-1, and Appendices 6A and 6B. The staff identified the above requirements as Confirmatory Item 6.5.1-1 in the DFSE. GE provided the required information and SSAR section, table, and appendices in an amendment to the SSAR. The staff has reviewed them and finds them acceptable. Therefore, DFSE Confirmatory Item 6.5.1-1 is resolved.

Appendix 6B to Chapter 6 of the SSAR addresses the compliance of the instrumentation for the SGTS with the requirements in Table 6.5.1-1 of SRP Section 6.5.1. Before the DFSE was issued, GE identified in Appendix 6B several deviations from the instrumentation requirements listed in SRP Table 6.5.1-1. GE attributed these deviations as being partly due to the single filter train proposed for the SGTS. In the DFSE, the staff stated that in addition to its concern regarding the deviations that stem from the use of a single filter train, it was concerned about deviations that cannot be attributed to a single filter train design. Therefore, as stated above, the staff required GE to revise Appendix 6B to reflect the redundant filter trains and additionally justify any remaining deviations from SRP Table 6.5.1-1. The staff identified its concern regarding the deviations as Open Item 6.5.1-1 in the DFSE. GE has amended Appendix 6B and has provided acceptable justifications for the deviations. The staff has reviewed GE's justifications for compliance of the SGTS with SRP Table 6.5.1-1 (minimum instrumentation, readout, recording and alarm provisions for ESF atmosphere cleanup systems) and finds that the proposed instrumentation and controls in the main control room

Engineered Safety Features

(MCR), as well as selected local panels, meet the above guidance, and are acceptable. Although it would prefer that GE provide all of the local instrumentations and controls described in the above guidance for operational efficiency and convenience, the staff finds that the SGTS instrumentation and controls meet the above guidance on the basis of the following:

- (1) The instrumentation and control (i.e., readout, recordings, and alarms) provisions for the related parameters (i.e., pressure drop, temperature, humidity, flow, etc.) are available and would be made accessible in the MCR.
- (2) Plant-specific procedures will be developed to respond to these instrumentation and controls.
- (3) The filtration train fully complies with the regulatory position of RG 1.52.
- (4) The acceptance, preoperational, and surveillance testing meet the intent of ASME N509 and N510.
- (5) Unfiltered inleakage will be controlled by the use of welded filtration housings (GE has stated that the advanced design of the filter housing and flow pattern would virtually eliminate any untreated filter bypass).
- (6) SGTS dose analysis is bounded by the maximum system flow.

Therefore, DFSER Open Item 6.5.1-1 is resolved.

During its meeting with the staff on May 5, 1992, GE committed to provide an analysis to demonstrate that the use of the system during the inerting, de-inerting, pressure control, or purging of the primary containment during normal plant operation will not impair its functional capability during a DBA. Therefore, the staff identified the submittal of the analysis as Open Item 6.5.1-2 in the DFSER. However, the staff considers that the analysis need not be performed, provided the use of the SGTS during power operation is limited to no more than 90 hours per year (approximately 1 percent of the time). BY SSAR Amendment 32, GE stated that normal operation of the SGTS would be much less than 90 hours per year for both trains combined during startup, power, and hot shutdown modes of operation. However, if 90 hours of operation per year for either train is to be exceeded, the COL applicant is required to provide functional damage analyses to demonstrate that the SGTS is capable of performing its intended function in the event of a LOCA, since this would be a change to SSAR certified design information. The

staff finds this acceptable. Therefore, DFSER Open Item 6.5.1-2 is resolved.

The staff concludes that the SGTS has a removal efficiency of 99 percent for all forms of radioiodine. It further concludes that the system meets the acceptance criteria of SRP Section 6.5.1 and is, therefore, acceptable.

6.5.2 Containment Spray System

SRP Section 6.5.2 does not apply to the ABWR plant.

6.5.3 Fission Product Control Systems and Structures

The staff reviewed fission product control systems and structures in accordance with SRP Section 6.5.3. The ABWR design includes two fission product control systems and structures. These are the primary containment, which includes the suppression pool with scrubbing and retention capability, and the secondary containment, which includes the SGTS with filtration capability.

Primary Containment: The primary containment is a cylindrical steel-lined reinforced-concrete structure that forms a limited leakage boundary for fission products released to the containment atmosphere following a LOCA or any other accident that releases lesser amounts of fission products. The structure is divided by a reinforced-concrete diaphragm floor and the reactor vessel pedestal into the upper and lower drywell, and a suppression chamber (wetwell). The diaphragm floor will be rigidly attached to the reactor pedestal and the containment wall. The diaphragm floor includes a liner to prevent steam bypass from the upper drywell to the suppression chamber airspace during any accident. The primary containment is totally enclosed within the reactor building, a portion of which forms the secondary containment. GE has assumed a design leak rate of 0.5 percent per day of the free containment volume for the duration of the accident (for further discussion on leak rate, see Section 15.4.4.1 of this report). A test program will be implemented to confirm leak integrity of the primary containment structure (see Section 6.2.6 of this report). The primary containment provides a passive barrier to limit leakage of airborne radioactive material following a LOCA by immediate closure of containment isolation valves except for those necessary for ECCS and ESF functions. Information on the primary containment design, its isolation methods, and isolation times are given in Sections 6.2.1 and 6.2.4 of this report. This information includes an evaluation of the suppression pool bypass leakage area of 0.0046 m² (0.05 ft²) assumed by GE. GE originally assumed a decontamination factor (DF) of 10 for the elemental and particulate forms of iodine resulting from scrubbing and retention in the suppression pool. In its radiological

consequence analysis, the staff conservatively assumed a decontamination factor of 2 (equivalent to suppression pool steam bypass of 50 percent). The staff finds that the assumed pool DF of 2 for elemental and particulate forms of radioactive iodine meets the regulatory position in SRP Section 6.5.5, "Pressure Suppression Pool as a Fission Product Cleanup System," and is acceptable as discussed in Section 15.3 of this report. Therefore, the design features of the primary containment discussed in this section are acceptable for controlling fission product release during an accident.

Secondary Containment: The secondary containment is a reinforced-concrete building that forms an envelope surrounding the primary containment above the basemat. It encloses the penetrations through the primary containment and all those systems external to the primary containment that could become a potential source of radioactive release after an accident. Following an accident, the secondary containment normal HVAC paths are secured. The SGTS will maintain the secondary containment at a negative pressure of 6.4 mm (1/4 in.) of water gauge or greater relative to the surrounding spaces within 20 minutes from the time the secondary containment is isolated and process the effluent gases through a filter train to remove airborne iodines and particulates. GE has assumed a draw-down time of 20 minutes for achieving this negative pressure in its LOCA dose analysis. GE states in SSAR Section 6.5.1.3.1(5) that the COL applicant will perform a secondary containment draw-down analysis to demonstrate the capability of the SGTS to maintain the design negative pressure following a LOCA, including inleakage from the open, nonisolated penetration lines identified during construction engineering and in the event of the worst single failure of a secondary isolation valve to close. The SGTS will filter airborne fission products (iodines and particulates) leaking from the primary containment before their release to the environs. GE assumed an inleakage rate of 50 percent of the secondary containment free volume per day at a differential pressure of 6.4 mm (1/4 in.) water gauge with respect to the surrounding spaces. The design of the secondary containment is discussed in Section 6.2.3 of this report. The design of the SGTS is discussed in Section 6.5.1 of this report.

On the basis of the above information and its evaluation of the SGTS in Section 6.5.1 of this report, the staff concludes that GE has demonstrated 99-percent removal efficiency by the SGTS filter train for all forms of iodine. The staff further concludes that the fission product control systems and structures provided in the ABWR design have the capability to reduce the DBA doses to within 10 CFR Part 100 limits (for further information, see Chapter 15 of this report).

The staff concludes that the fission product control systems and structures provided for the ABWR meet the acceptance criteria of SRP Section 6.5.3 and are, therefore, acceptable.

6.6 Inservice Inspection of Class 2 and 3 Components

A detailed evaluation of the inservice inspection (ISI) of the reactor coolant pressure boundary is given in Section 5.2.4 of this report.

The staff's evaluation of SSAR Section 6.6 was not complete in the DSER (SECY-91-153) and was therefore identified as Open Item 31. The staff reviewed SSAR Section 6.6, and Open Item 31 was resolved based on the following discussion.

SSAR Section 6.6 and Table 6.6-1 describe certain commitments and plans for the preservice inspection (PSI) and ISI programs for ASME Code, Class 2 and 3 components. GE discussed basic inspection concepts and general ASME Code provisions because the requirements of the NRC regulations might be controlled by the date of order of each specific component subject to examination. The staff review was performed in accordance with SRP Section 6.6, except as explained below.

Throughout the service life of the plant, the COL applicant has the overall responsibility for the ISI of the Class 2 and 3 components. However, other organizations also contribute to the examination activity, including the reactor vendor and the architect-engineer. The NRC staff and GE discussed the issue of ISI requirements on March 25 and 26, 1992, and decided that the COL applicant is responsible for the development of the psi and ISI programs for all ASME Code, Class 2 and 3 components. This was identified as DFSEER COL Action Item 6.6-1.

GE included this information in SSAR Section 6.6.9, which states that the COL applicant will develop psi and ISI program plans as outlined in Section 6.6 of the SSAR. This is acceptable.

6.6.1 Requirements of 10 CFR 50.55a, "Codes and Standards"

Pursuant to 10 CFR 50.55a(g)(3), ASME Code Class 2 and 3 components should be designed and provided with access to enable the performance of inservice examination of such components and should meet the preservice examination requirements in ASME Code, Section XI, applied to the construction of the particular component. The applicable construction code will be determined by paragraph NCA-1140 of ASME Code, Section III, but the

Engineered Safety Features

construction code that is selected must be incorporated by reference in 10 CFR 50.55a(b).

All items of a nuclear power plant may be constructed to a single edition of the ASME Code, or each item may be constructed to individually specified code editions and addenda. NCA-1140 of ASME Code, Section III, states, in part, that in no case should the ASME Code edition established in the design specification be earlier than 3 years before the date that the nuclear power plant construction permit application is docketed. In addition, NCA-1140 permits the use of later code editions by mutual consent of the owner and certificate holder, that is, an organization holding a valid N, NV, NPT, or NA certificate of authorization issued by ASME.

Throughout the service life of a boiling or pressurized water-cooled nuclear power facility, components (including supports) that are classified as ASME Code, Class 2 and 3, must meet the requirements, except design and access provisions and preservice examination requirements, set forth in ASME Code, Section XI, that become effective after the editions specified in 10 CFR 50.55a(g)(3) and are incorporated by reference in 10 CFR 50.55a(b), to the extent practical within the limitations of design, geometry, and materials of construction of the components. The inservice examination of components conducted during the initial 10-year interval must comply with the requirements of the latest edition and addenda of ASME Code, Section XI, incorporated by reference in 10 CFR 50.55a(b) on the date 12 months before the date of issuance of the operating license. The regulations also require that the ISI program be updated for each subsequent 10-year interval to comply with the ASME Code incorporated by reference in 10 CFR 50.55a.

6.6.2 Application of the Codes and Standards Rule to Standard Design Certification

Compliance with the requirements of the regulations for PSI and ISI must be based on commitments for or by the COL applicant. The applicable ASME Code(s) for both design and ISI are not known at the time of the design certification application. In addition, the regulation permits the COL applicant the option to change to a newer edition of ASME Code, Section XI, by component, during the construction of the plant.

The staff reviewed the information in the SSAR that emphasized design access and the use of effective nondestructive examination (NDE) methodology. Because PSI requirements are established and known at the time each component is ordered, 10 CFR 50.55a(g) does not have provisions for relief requests for impractical examination requirements. ASME Code, Section XI, does

have provisions to use certain shop and field examinations in lieu of the onsite preservice examination. Therefore, the COL applicant will incorporate plans for NDE during design and construction in order to meet all access requirements of the regulations. This was identified as DFSER COL Action Item 6.6.2-1.

GE included this information in SSAR Section 6.6.9, which states that the COL applicant will incorporate plans for NDE during design and construction in order to meet all access requirements of the regulations. This is acceptable.

ASME Code, Section XI, states that the preservice examination should be conducted with equipment and techniques equivalent to those that are expected to be used for subsequent inservice examinations. Improvements in the ultrasonic testing of Class 2 and 3 components will occur in the near future. ASME has published ASME Code, Section XI, Appendix VII, "Qualification of Nondestructive Examination Personnel for Ultrasonic Examination," and Appendix VIII, "Performance Demonstration for Ultrasonic Examination Systems." The NRC staff has published in the Federal Register its intent to reference in 10 CFR 50.55a(b) the edition of ASME Code, Section XI, that includes the published Appendix VII. In addition, the staff has established a technical contact to coordinate the implementation of Appendix VIII. Therefore, the PSI program for the ABWR should include provisions that ultrasonic testing be performed in accordance with Appendices VII and VIII pursuant to 10 CFR 50.55a(g)(3). However, the staff did not identify the above as a separate issue because it was covered by DFSER Open Item 5.2.4-1. See Section 5.2.4 of this report.

The requirements for the initial ISI program will be determined by the ASME Code in effect 1 year before the issuance of an operating license. The COL applicant is required to meet the ISI requirements "to the extent practical within the limitations of design, geometry and materials of construction of the components." The regulations have provisions for the staff evaluation, on written request, of new ASME Code requirements determined by a licensee to be impractical for its facility.

6.6.3 Staff Evaluation

A PSI program and an ISI program will be prepared for each ABWR plant. GE states that provisions have been made in the design and layout of ASME Code, Class 2 and 3 systems to allow for access for the examinations required in ASME Code, Section XI, Articles IWC-2000 and IWD-2000, and as defined in the ISI program. As a result of a meeting between GE and the NRC staff on March 25

and 26, 1992, it was determined that the COL applicant is responsible for the development of the PSI and ISI programs for all ASME Code, Class 1, 2, and 3 components. The NRC staff reviewed the above information regarding design access for the preservice examination and finds the information pertaining to planning adequate. The staff notes that the SSAR Section 6.6 states that the COL applicant will meet the

requirements of the NRC regulations defined in 10 CFR 50.55a.

The SSAR Section 6.6.3.2 states that the examination techniques to be used for ISI will include radiographic, ultrasonic, magnetic particle, liquid penetrant, eddy current, and visual examination methods. For all examinations, both remote and manual, specific procedures will be prepared describing the equipment, inspection technique, operator qualifications, calibration standards, flaw evaluation, and records. These techniques and procedures will meet the requirements of Articles IWC-2000 and IWD-2000 in the edition of ASME Code, Section XI, in effect as stated in 10 CFR 50.55a(g). GE has revised SSAR to state that personnel performing ultrasonic examinations will be qualified in accordance with ASME Code, Section XI, Appendix VII. Ultrasonic examination systems will be qualified in accordance with an industry-accepted program for implementing ASME Code, Section XI, Appendix VIII. GE considered the supplemental examinations recommended in GE service information letters (SILs) and rapid communication service information letters (RICSILs) for previous BWR designs as not applicable to the ABWR. In the ABWR design either of the following has been eliminated: components addressed by the SILs or RICSILs or the need for the examination by eliminating crevice designs and using materials resistant to the known degradation mechanisms, such as intergranular stress corrosion cracking, on which the SIL and RICSIL examinations were based.

The staff finds these commitments acceptable.

6.7 High-Pressure Nitrogen Gas Supply System

The ABWR design includes four compressed air systems: the instrument air system, the service air system, the high-pressure nitrogen (HPIN) gas supply system, and the atmospheric control system (ACS). In SSAR Section 6.7, GE supplies information on the design of the HPIN system. The staff reviewed the information in accordance with SRP Section 9.3.1.

The HPIN system consists of both a nonessential (i.e., non-safety-related) and an essential systems. A single nonessential system will provide a continuous nitrogen supply to all pneumatically operated components in the primary containment during normal operation. As noted in Section 6.5.2 of this report, during normal operation, the nitrogen gas evaporator/storage tank will supply the HPIN system via the makeup line to the ACS. The essential system has two independent divisions. Each division contains a safety-related emergency stored nitrogen supply capable of supplying 100 percent of the requirements of the division being serviced. Nitrogen gas for the essential system will be supplied from HPIN gas storage bottles. There are tielines between the nonessential system and each division of the essential system. Each tieline has a motor-operated shutoff valve.

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

6.8 Main Steam Isolation Valve Leakage Control System

An MSIV leakage control system will not be used in the ABWR, as discussed in Section 10.3 of this report.

7 INSTRUMENTATION AND CONTROLS

7.1 Introduction

The staff has reviewed the GE Nuclear Energy (GE) advanced boiling water reactor (ABWR) standard safety analysis report (SSAR) and associated reference material. A draft safety evaluation report (DSER) was issued on September 18, 1991, attached to SECY-91-294, "Draft Safety Evaluation Report on the General Electric Boiling Water Reactor Design Covering Chapter 7 of the Standard Safety Analysis Report." The applicant responded to the open issues identified in the DSER by letter dated February 3, 1992. A copy of the DSER was also provided to the Advisory Committee on Reactor Safeguards (ACRS).

A draft final safety evaluation report (DFSER) was issued on October 14, 1992, attached to SECY-92-349, "Draft Final Safety Evaluation Report on the General Electric Boiling Water Reactor Design Covering Chapter 7 of the Standard Safety Analysis Report." The DFSER has also been provided to the ACRS. The applicant responded to the open issues identified in the DFSER by several submittals described in this report. The following sections describe the staff's final safety evaluation findings. This report describes the resolution of the open items from the DSER and DFSER.

The staff has reviewed Chapter 7 of the SSAR submittal through Amendment 35. Chapter 7 contains the primary instrumentation and control (I&C) descriptions and commitments. Chapter 1 of the SSAR contains most of the associated instrumentation block diagrams. A few items from other chapters of the SSAR are included in this evaluation, as indicated. The staff has reviewed both proprietary and non-proprietary information from the SSAR and other docketed materials.

7.1.1 Acceptance Criteria

The acceptance criteria used as the basis for the staff's evaluation of the I&Cs for the ABWR are set forth in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition," and 10 CFR Part 52. The primary sections of NUREG-0800 used for this portion of the review are Section 7, "Instrumentation and Control," and Section 9.5.2, "Communications Systems." The Commission issued review guidance in staff requirements memoranda (SRM), dated June 26, 1990, and February 15, 1991, pertaining to SECY-90-016, "Evolutionary Light Water Reactor Certification Issues and Their Relationship to Current Regulatory Requirements," and SECY-90-377, "Requirements for Design Certification under 10 CFR Part 52," respectively.

The staff developed the necessary acceptance criteria to cover the certification of those systems because the SRP presently does not address design certification, or the newer digital technology used in these ABWR I&C systems (the SRP was last revised in 1984). The additional acceptance criteria and conformance to the SRP are discussed in the applicable sections of this report. This report also describes certain items which will be included in the certified design material (CDM) which includes the design description; inspections, tests, analyses and acceptance criteria (ITAAC), site parameters, and interface requirements for design certification. The description of the CDM development process, bases, and acceptability is primarily discussed in Section 14.3 of this report. This report discusses only those CDM areas which relate to the I&C systems' certified design process in addition to specific design characteristics.

The staff referenced its evaluation of previously-reviewed plant designs and topical reports, as stated.

7.1.2 Method of Review

In the DSER (SECY-91-294) the staff discussed the following items: systems which have had significant design changes compared to previously-reviewed and-accepted designs; the appropriate level of detail necessary for design certification; and design issues where more information was needed in order for the staff to make a finding of acceptability. Since issuing the DSER the staff has concentrated on resolving the open issues in the DSER and reviewing the submittals of the CDM and technical specifications (TS). The DFSER (SECY-92-349) described the remaining open issues and the acceptance of design process commitments in lieu of design details for certain rapidly developing technologies such as software design.

In addition to reviewing the SSAR, the staff also visited GE's offices and reviewed documents associated with the SSAR material. All documents relied upon for the safety evaluation that are not included in the SSAR have been docketed.

7.1.3 General Findings

GE has identified the I&C systems which are important to safety in accordance with Regulatory Guide (RG) 1.70, Revision 3, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," November 1978. Each of the safety systems identified in the SSAR also has an associated Design Description and ITAAC. Some aspects of the design which affect all of the I&C systems (such as the software design process) are addressed in an I&C CDM section rather than repeated for

Instrumentation and Controls

each individual I&C system CDM. Most of the non-safety systems are also described in a CDM.

The ABWR I&C systems are significantly different from I&C systems in previously-approved designs. The primary differences result from using digital microprocessor-based I&C systems with multiplexed data links in place of the analog electronics, relay logic, and hardwired systems previously approved by the staff. The staff previously reviewed designs using digital equipment similar to that proposed for the ABWR; however, the ABWR significantly extends the quantity and scope of system coverage of the digital microprocessor applications and, in conjunction with the 10 CFR Part 52 design certification process, requires a unique review which is not directly comparable with any past licensing review. The differences from past license reviews are addressed in greater detail in this chapter.

The comparison of the ABWR to the GESSAR II design in SSAR Table 7.1-1 is not a sufficient basis for accepting the ABWR I&C design because, as the table shows, of the significant differences in the implementation of the I&C system design when using the ABWR digital equipment.

7.1.3.1 Standard Review Plan Criteria

The acceptance criteria listed in Table 7.1 of this report are identified in the Commission regulations and Standard Review Plan (SRP) Chapter 7, "Instrumentation and Controls." The SRP also provides some review guidance and acceptance criteria not found in the specified standards, General Design Criteria (GDC) and other references. Table 7.1 does not consider the degree of conformance of the ABWR with the requirements or the method of implementation of the I&C system design, but simply lists the SSAR sections where the standards and criteria are mentioned in the SSAR. Many of the standards and criteria discussed in SSAR Chapter 7 and Appendices A, B, and C are applicable to the previous SSAR sections which describe the I&C systems without being specifically listed and discussed in those sections. The request for additional information (RAI) column refers to the staff RAI and the GE response to them which relate to specific standards and criteria. The RAIs and responses are provided in Chapter 20 of the SSAR. The conformance to and any exceptions from the standards and criteria are discussed in this chapter. SSAR Table 7.1-2, "Regulatory Requirements Applicability Matrix for I&C Systems," also provides a similar matrix of the SRP criteria to that in this report, but compares them with the I&C systems rather than the SSAR sections. In general, GE committed to meet the SRP requirements without exception. The few exceptions GE requested are noted in

SSAR Section 1.8 and the applicable sections of this chapter.

10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," contains the GDC. Chapter 3.1 of the SSAR discusses compliance with the GDC generally and references other SSAR chapters for specifics. Table 7.1-2 of the SSAR lists the GDC applicable to the I&C systems.

The SSAR Chapter 3 discussion of GDC 3, "Fire Protection;" GDC 26, "Reactivity Control System Redundancy and Capability;" GDC 27, "Combined Reactivity Control System Capability;" GDC 30, "Quality of Reactor Coolant Pressure Boundary;" GDC 37, "Testing of Emergency Core Cooling System;" GDC 42, "Inspection of Containment Atmosphere Cleanup Systems;" GDC 43, "Testing of Containment Atmosphere Cleanup Systems;" GDC 55, "Reactor Coolant Pressure Boundary Penetrating Containment;" and GDC 56, "Primary Containment Isolation;" referred to Chapter 7 of the SSAR for further discussion of each GDC. However, these GDC did not appear in the SSAR Chapter 7 GDC discussions. The topics generally are discussed in other sections of the SSAR.

GDC 2, "Design Bases for Protection Against Natural Phenomena;" GDC 15, "Reactor Coolant System Design;" GDC 16, "Containment Design;" GDC 38, "Containment Heat Removal;" and GDC 44, "Cooling Water," were listed in Chapter 7, but there was no corresponding reference to Chapter 7 in the Chapter 3 evaluations. The GDC discrepancies were listed as Open Item 7.1.3.1-1 in the DFSE. GE revised the SSAR to include references in Chapter 3 to Chapter 7 for GDC 2, 15 and 38. GE also clarified that the GDC discussed in Chapter 3 apply to the equipment included in Chapter 7, regardless of whether there is a specific discussion in the I&C sections in Chapter 7 or not. The staff concludes that this is acceptable; therefore, this open item is resolved.

The specific standards to which GE has committed are a significant consideration for the staff's safety findings. The most important aspects of those criteria, which are required to be certified by rulemaking, are included in the CDM and are discussed in the staff's evaluation.

7.1.3.2 Electric Power Research Institute Requirements Document

Electric Power Research Institute (EPRI) prepared a document of technical requirements, referred to as the advanced light water reactor (ALWR) Utility Requirements Document (ALWR RD), intended by EPRI to be applied to the design of ALWR power plants, including the ABWR.

Table 7.1 SRP Acceptance Criteria

SRP CRITERIA GDC	ABWR SSAR Chapter 7											
	7.1	7.2	7.3	7.4	7.5	7.6	7.7	7.8	7.A	7.B	7.C	RAI
10CFR50.55A	X	X	X	X	X	X			X			
10CFR50APPB	X	X	X	X	X	X			X	X		X
GDC(GENERIC)	X	X	X	X	X	X			X	X		
GDC 1	X	X	X	X	X	X			X	X		X
GDC 2	X	X	X	X	X	X			X			X
GDC 4	X	X	X	X	X	X			X			X
GDC 10	X	X				X			X			
GDC 13	X	X	X	X	X	X	X		X			
GDC 15	X	X	X	X		X	X		X			
GDC 16	X	X	X			X			X			
GDC 19	X	X	X	X	X	X	X		X			
GDC 20	X	X	X	X		X						
GDC 21	X	X	X			X						
GDC 22	X	X	X			X						
GDC 23	X	X	X			X						
GDC 24	X	X	X			X						
GDC 25	X	X		X		X						
GDC 28	X					X			X			
GDC 29	X	X	X			X			X			
GDC 33	X		X	X		X			X			
GDC 34	X		X	X		X						
GDC 35	X		X	X		X						
GDC 38	X		X			X						
GDC 41	X		X			X						
GDC 44	X		X			X						
IEEE STANDARDS IN SRP	ABWR SSAR Chapter 7											
	7.1	7.2	7.3	7.4	7.5	7.6	7.7	7.8	7.A	7.B	7.C	RAI
279-1971	X	X	X	X	X	X	X		X			
317-1972												X
323-1974	X	X							X			
336-1971												
338-1971	X	X	X	X	X	X			X			
379-1972	X	X	X									
384-1974	X	X	X	X	X	X	X		X			
OTHER GUIDANCE IN SRP	ABWR SSAR Chapter 7											
	7.1	7.2	7.3	7.4	7.5	7.6	7.7	7.8	7.A	7.B	7.C	RAI
ANSI/ANS 4.5					X							
ANSI N41.15	X	X	X	X	X				X			
ISA 67.02	X	X	X	X	X	X	X		X			
NUREG-0694			X		X							
NUREG-0696	X	X	X	X	X	X	X		X			
NUREG-0718	X	X	X	X	X							
NUREG-0737	X	X	X	X	X	X	X		X			

Table 7.1 SRP Acceptance Criteria (continued)

TMI ACTION PLAN ITEMS IN SRP	ABWR SSAR Chapter 7											
	7.1	7.2	7.3	7.4	7.5	7.6	7.7	7.8	7.A	7.B	7.C	RAI
II.D.3	X		X		X							
II.E.1.2	NOT	APPLICABLE										
II.E.4.2	X		X									
II.F.1	X				X	X						
II.F.3	X		X		X	X						
II.K.1	X	X	X									X
II.K.2	NOT	APPLICABLE										
II.K.3	X	X	X				X					
REGULATORY GUIDES IN SRP	ABWR SSAR Chapter 7											
	7.1	7.2	7.3	7.4	7.5	7.6	7.7	7.8	7.A	7.B	7.C	RAI
RG 1.22	X	X	X	X		X						
RG 1.47	X	X	X	X		X			X			
RG 1.53	X	X	X	X		X			X			
RG 1.62	X	X	X	X	X	X			X			
RG 1.75	X	X	X	X	X	X	X	X	X			
RG 1.89	X	X	X	X	X	X			X			
RG 1.97	X		X	X	X	X						X
RG 1.105	X	X	X	X	X	X						
RG 1.118	X	X	X	X	X	X						
RG 1.151	X	X	X				X					
BRANCH TECHNICAL POSITION IN SRP	ABWR SSAR Chapter 7											
	7.1	7.2	7.3	7.4	7.5	7.6	7.7	7.8	7.A	7.B	7.C	RAI
BTP ICSB 3	X		X	X		X						
BTP ICSB 4	NOT	APPLICABLE										
BTP ICSB 12	X	X										
BTP ICSB 13	NOT	APPLICABLE										
BTP ICSB 14	NOT	APPLICABLE										
BTP ICSB 16	DELETED											
BTP ICSB 20	X		X									
BTP ICSB 21	X	X	X			X						
BTP ICSB 22	X	X	X			X						
BTP ICSB 26	X	X										

Volume II, "Evolutionary Plant," specifically applies to the ABWR design. By letter dated June 12, 1990, GE provided a comparison of the ALWR RD, Volume II, and the ABWR SSAR. The EPRI requirements pertaining to the I&C systems are contained primarily in Chapter 10 of Volume II of the ALWR RD, "Man-Machine Interface Systems." GE concluded in their letter that a detailed comparison of the ABWR SSAR to the many ALWR RD requirements showed that the ABWR complied with all but a few of the EPRI ALWR requirements.

GE did not document its detailed comparison, nor did the staff compare the EPRI's requirements to the entire ABWR design for applicability. In addition, there are many requirements in the ALWR RD without any corresponding references to published standards. There are also many requirements in the ALWR RD which allow a vendor different options for implementation in a design, or provide a caveat that the requirement should be implemented to the extent practical. In August 1992, the staff issued the final safety evaluation report for the

Volume II of the ALWR RD, and requested that GE clearly document in detail how the ABWR design differs from that in the ALWR RD and also how it complies with EPRI's requirements. This was identified as DFSER Open Item 1.1-1. The resolution of this issue is described in Section 1.1 of this report.

7.1.3.3 Additional Criteria

Table 7.2 of this report lists criteria and standards discussed in the ABWR SSAR but are not included in the SRP criteria in Table 7.1 of this report. However, some of the standards in Table 7.2 are referenced in standards included in Table 7.1 and, therefore, could be considered as part of the SRP listings. They are listed in Table 7.2 for convenience and completeness. Table 7.2 also includes the standards and criteria which GE has committed to use since the DSER was issued to address the new technology issues not addressed in the SRP. Some of these items are included in SRP chapters other than Chapter 7. Additional comments regarding use of future standards to accommodate changing I&C digital technology are discussed elsewhere in this chapter.

ABWR SSAR Table 7.1-2, "Regulatory Requirements Applicability Matrix for I&C Systems," lists the ABWR I&C systems and the requirements which apply to each system. The safety system logic and control (SSLC) and essential multiplexing system (EMS) are not explicitly defined in SSAR Table 7.1-2. The staff noted that GE has defined the "reactor protection (trip) system (RPS)" as a system that provides sensor signals and reactor trip outputs, and that uses the SSLC as the means for implementing trip logic. The SSLC in turn uses the EMS as the means for interfacing with sensors and actuated equipment except where the devices are hardwired. Since the commitment to the SRP standards and criteria as shown in SSAR Table 7.1-2 includes the SSLC and EMS, the staff finds the commitment acceptable.

In the DSER, the staff determined that SSAR Table 7.1-2 provides commitments to the acceptance criteria prescribed by the SRP with one exception. The table in the SSAR did not include an explicit commitment to GDC 1, "Quality Standards and Records," of 10 CFR Part 50. Although GDC 1 was addressed in Chapter 3 of the SSAR, the staff believed that an explicit commitment should be provided in Chapter 7 and identified this as an open issue in the DSER (Open Item 5). Specifically, GDC 1 requires that systems and components important to safety be designed, fabricated, and tested to quality standards commensurate

with the importance of the safety functions to be performed. In addition, this criterion also requires that GE identify and evaluate applicability, adequacy, and sufficiency of generally recognized codes and standards.

In the DSER, the staff concluded that acceptance criteria for software-based digital I&C systems were not addressed in sufficient detail by the regulatory positions and the industry standards referenced by GE in the SSAR at the time the DSER was written. No standards were identified regarding electromagnetic compatibility (EMC), multiplexor architecture, communications protocols, and software design. GE has since committed to industry standards listed in SSAR Appendix 7A for these areas, and the staff concludes that the ABWR design conforms with GDC 1 in that GE has committed to use the appropriate industry standards for the type of equipment to be used in the implementing the design. Therefore, this open issue is resolved.

SRP Appendix 7-B, "General Agenda, Station Site Visits," includes the following statement:

An important part of the review at the operating license stage is a site visit. It is preferable to have the site visit sometime before the completion of the drawing review. The purpose of the site visit is to supplement the review of the design based on the drawings and to evaluate the actual implementation of the design as installed at the site. The NRC Regional Office having jurisdiction over the plant under consideration should be notified ahead of time of the visit so that the regional inspectors can become familiar on a first-hand basis with findings that may require followup action. Since proper implementation of design is the ultimate goal of the technical review process, the importance of a site visit is self-evident.

SRP Appendix 7-B provides a general agenda for the station site visit which includes verification of layouts, separation and isolation, test features and potential for damage due to fire and flooding, or other environmental effects. Because the design certification will be issued prior to the selection of a construction site, this SRP review item cannot be completed at this time. The method to be used to address the necessary inspection tasks for design certification will be addressed through the ITAAC process and commitments to preoperational tests described in SSAR Chapter 14. The review described in SRP Appendix 7-B will be accomplished as part of the testing done for the combined license (COL).

Table 7.2 Chapter 7 Additional Requirements

CRITERIA/ STANDARD	ABWR SSAR Chapter 7											
	7.1	7.2	7.3	7.4	7.5	7.6	7.7	7.8	7.A	7.B	7.C	RAI
GDC 12						X						
GDC 17												X
GDC 18												X
GDC 54					X							
GDC 55					X							
GDC 56					X							
GDC 57					X							
RG 1.26		X										
RG 1.28	X	X	X	X	X	X			X			
RG 1.30	X	X	X	X	X	X			X			
RG 1.32	X	X	X	X	X	X			X			
RG 1.38	X	X	X	X	X	X			X			
RG 1.58					X							
RG 1.64	X	X	X	X	X	X			X			
RG 1.70	X	X	X	X	X	X	X	X	X			X
RG 1.74					X							
RG 1.88	X	X	X	X	X	X			X			
RG 1.100	X	X	X	X	X	X			X			
RG 1.123	X	X	X	X	X	X			X			X
RG 1.144	X	X	X	X	X	X			X			
RG 1.146					X							
RG 1.152	X	X	X	X	X	X			X	X		X
RG 1.153	X	X	X	X	X	X			X			X
IEC 801-2	X	X	X	X	X	X			X			X
IEC 880										X		
IEEE 344	X	X	X	X	X	X						
IEEE 352	X	X										X
IEEE 472	X	X	X	X	X	X			X			X
IEEE 518		X										X
IEEE 603	X	X	X	X	X	X			X			X
IEEE 730										X		
IEEE 7-4.3.2	X	X	X	X	X	X			X	X		X
IEEE 802.2	X	X	X	X	X	X			X			X
IEEE 802.5	X	X	X	X	X	X			X			X
IEEE 828	X	X	X	X	X	X			X	X		X
IEEE 829	X	X	X	X	X	X			X	X		X
IEEE 830	X	X	X	X	X	X			X	X		X
IEEE 1012	X	X	X	X	X	X			X	X		X
IEEE 1033	X	X	X	X	X	X			X	X		X
IEEE 1042	X	X	X	X	X	X			X	X		
IEEE 1228	X	X	X	X	X	X			X	X		
NEMA 4		X	X									
ASME NQA 2.7										X		
NEDO 24708			X									
NEDC 31336		X										X
NEDO 31439												X

Table 7.2 Chapter 7 Additional Requirements (continued)

CRITERIA/ STANDARD	ABWR SSAR Chapter 7											
	7.1	7.2	7.3	7.4	7.5	7.6	7.7	7.8	7.A	7.B	7.C	RAI
NEDE 31906				X								X
ANS 4.5					X							
NUREG 0493	X	X	X	X	X	X	X	X	X	X	X	X
NUREG 0588					X							
NUREG CR3453									X			
NUREG -0308									X			
NUREG -0491									X			
ISA 67.04	X	X	X	X	X	X				X		X
10CFR21	X	X	X	X	X	X				X		X
MIL HDBK 251	X	X	X	X	X	X				X		X
MIL HDBK 217E	X	X	X	X	X	X				X		X
MIL STD 2167	X	X	X	X	X	X			X	X	X	X
MIL STD 461C	X	X	X	X	X	X				X		X
MIL STD 462	X	X	X	X	X	X				X		X
10CFR50.46	X		X									
10CFR50.49	X	X	X	X	X	X				X		X
10CFR50.62				X						X		X
GL 83-08										X		X
GL 84-23										X		X
ISO 7498		X							X	X		X
ANSI C37.90.2	X	X	X	X	X	X			X	X		X
ANSI C62.41	X	X	X	X	X	X			X	X		X
ANSI C62.45	X	X	X	X	X	X			X	X		X
ANSI C63.12	X	X	X	X	X	X			X	X		X
ANSI X3T9.5	X	X	X	X	X	X			X	X		X

Because the ABWR has been submitted for design certification, the requirements of 10 CFR Part 52 apply in addition to those of 10 CFR Part 50. 10 CFR Part 52 requires a level of design detail beyond a simple commitment to conformance with the existing requirements. 10 CFR 52.47(a)(2) requires that:

The application must contain a level of design information sufficient to enable the Commission to judge the applicant's proposed means of assuring that construction conforms to the design and to reach a final conclusion on all safety questions associated with the design before the certification is granted. The information submitted for a design certification must include performance requirements and design information sufficiently detailed to permit the preparation of acceptance and inspection requirements by the NRC, and procurement

specifications and construction and installation specifications by an applicant. The Commission will require, prior to design certification, that information normally contained in certain procurement specifications and construction and installation specifications be completed and available for audit if such information is necessary for the Commission to make its safety determination.

10 CFR 52.47(b)(1) also states that, "... this rule must provide an essentially complete nuclear power plant design except for site-specific elements." The following sections of this report describe the information provided by GE and the staff's conclusions concerning conformance with the SRP criteria, additional criteria necessary to address newer digital I&C technology, and the above requirements of 10 CFR Part 52.

Instrumentation and Controls

In the DSER, the staff concluded that GE had adequately committed to the SRP requirements but had not provided design information detailed enough for the staff to conclude that the plant will be built in accordance with those standards. This was identified as an open issue in the DSER. The resolution of this open issue was based on a digital I&C system design process and the use of a phased approach to design implementation through the ITAAC. In addition, certain restrictions on changing SSAR commitments have been incorporated to ensure staff agreement prior to any changes to key details of the design process. This approach formed a substantial part of the basis for staff's acceptance of the I&C systems for certification under 10 CFR Part 52.

The CDM describes the basic design process to be followed during the design of the I&C hardware and software and will include specific acceptance criteria. The design process CDM will be in the same format as the system CDMs. The I&C system design process is described below. Additional general information on this aspect of the review process is provided in SECY-92-287, "Form and Content for a Design Certification Rule."

CDM Process

The ABWR I&C systems described in the SSAR lack substantial design detail. Therefore, the following descriptions of the process in the SSAR and CDM for development of the detailed design, together with certain SSAR change restrictions are key features of the final safety determination for the I&C systems in this report and are included in the design certification rule for the ABWR. The staff's audit and approval function following design certification is also a critical feature of the acceptance of the CDM and SSAR descriptions for the I&C systems. The staff proposed in the DFSER that this design process be described in the appropriate CDM. The CDM has been revised to include the phased development process, however, the staff audit participation is only described in this report.

In the design description section of the CDM, a textual and graphical description of the fundamental characteristics of the various I&C systems is provided. The CDM includes a summary of the systems and design commitments which will be certified for the life of the plant. The items in the design description cannot be changed over the life of the plant without rulemaking and, therefore, the details of the design which can be changed are not described. The ITAAC section of the CDM provides a general description of the activities that will be performed by the COL applicant to verify that the I&C system has been built in accordance with the certified design description and commitments. The SSAR provides additional information

on the implementation of the design. The SSAR provides additional detail (unlike the CDM) by reference to specific standards for the designers (COL applicant) to follow. The staff will audit the COL applicant's activities at appropriate review points to ensure that the detailed design is in conformance with the design descriptions and ITAAC.

It is premature to complete the final design details for the microprocessor and digital control technology aspects of the I&C system design of the ABWR before the COL is issued. This is because the technology in this area is rapidly evolving and it is, therefore, important that the certified design description and ITAAC not "lock in" a design which could be obsolete at the time of construction. The approach that will be used is to "lock in" a design process and the specific acceptance criteria which, if met, would result in a design which is acceptable. At the same time, the functional system description, commitment to standards, and commitment to a structured design process in the CDM must be sufficient for the staff to make its final safety determination. This process was described in SECY-92-053, "Use of Design Acceptance Criteria During 10 CFR Part 52 Design Certification Reviews."

The CDM was not submitted by GE as part of the original design certification application. Subsequently, a small group of system design description and ITAAC were submitted as a pilot program in an effort to reach agreement on the general scope of the design description and ITAAC for all systems. The staff reviewed the RPS design description and ITAAC as representative of the I&C systems. The staff provided comments whose resolutions were incorporated into the documents. In general, the descriptions and ITAAC that were proposed by GE were not sufficiently detailed or complete for a staff safety finding. Therefore, the staff concluded that it was necessary to review both the SSAR and the CDM and to require more CDM information in order to reach a safety conclusion. Because the CDM was still under review when the DFSER was issued, the staff identified this as DFSER Open Item 7.1.3.3-1. The CDM has been submitted, reviewed, and found acceptable. The acceptance of the CDM is addressed in Section 14.3 of this report. Therefore, this item is resolved.

GE has not finalized the hardware and software design for the ABWR digital I&C systems. Therefore, the staff used the two-part approach described in SECY-92-053 to reach its safety finding for design certification. In reviewing the I&C systems, the first part of this approach involved a detailed functional review at the block diagram level to ensure the applicant has appropriately implemented the Commission's requirements related to postulated single failures, common mode failures, signal isolation, and other

aspects of the staff's review that are typical for any safety-related I&C system, including analog control systems such as those in current operating nuclear plants. This review confirmed that the detailed functional requirements for the I&C systems are established.

The second part of the staff's approach to reach certification safety finding addresses the adequacy of the digital control systems implementation with respect to the system functional requirements. This relies upon a formal design implementation process with a phased ITAAC program for design development within certain predefined constraints and limits.

This report documents the results of the staff's review of the ABWR I&C systems at the functional block diagram level as described above for the first part of the two-part approach documented in SECY-92-053.

Certified Design Material

Because the CDM and SSAR reviews are interrelated, it is necessary for the two parts of the review to be considered integrally rather than individually. The following section uses the computer development process to illustrate the implementation of the CDM and the NRC staff involvement. This process is also used for multiplexor design, setpoint methodology, and EMC and equipment qualification verification.

Computer Development

The primary function of the computer is to implement the functional I&C requirements described in the CDM and the SSAR for the ABWR systems. The decomposition of the functional system (SSLC, RPS, ARI, etc.) requirements to specific computer hardware and software components to accomplish the various tasks is accomplished using the structured design process described below.

The CDM includes the description of the design process to be followed for hardware and software development, design commitments, the ITAAC to be performed, and the appropriate acceptance criteria. The ITAAC for computer hardware and software (included in the I&C CDM) differs from the majority of the system ITAAC in that the acceptance criteria include acceptance criteria for the certified design process (referred to as design acceptance criteria in SECY-92-053). This ITAAC describes attributes of the process to be used to develop the software as well as attributes of the final software product. The ITAAC for software and hardware (included in the CDM Section 3.4, "Instrumentation and Control") describes several design stages.

The following design stages (the software lifecycle) are necessary for the development and operation of both safety-related and non-safety-related software.

- (1) planning
- (2) requirements
- (3) design
- (4) implementation
- (5) integration
- (6) validation
- (7) Installation
- (8) operation and maintenance

The staff considers these stages to be necessary for the development of a computer system of sufficient quality to adequately perform its design function. The stages are based on generic activities that represent software and system lifecycle models, and encompass existing national standards and input from ongoing efforts in the revisions of national and international standards for computer systems in nuclear power plants, and expert opinion.

GE has revised the CDM and SSAR to provide a similar lifecycle definition but selected the following categories:

- (1) planning
- (2) design definition
- (3) software design
- (4) software coding
- (5) integration
- (6) validation
- (7) change control

The CDM and SSAR contain criteria which describe the method to develop plans and procedures that will guide the design process throughout the lifecycle stages. The ITAAC provides the acceptance criteria for verifying the design through the stages, while the SSAR adds the set of guidelines and standards that will provide more detailed criteria for the development of the design process. The CDM has been written to encompass the most important aspects from the standards. The SSAR set of standards and criteria encompass the guidance that will be used to start the computer software and hardware design process by generating the plans for the computer design throughout the lifecycle. Therefore, the staff concludes that these stages are acceptable for the development and operation of safety-related and non-safety-related computer systems.

The software QA (SQA) plan describes the software-specific activities that are to be performed and controlled in addition to the approved QA plan (in accordance with Appendix B of 10 CFR Part 50, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants") for the total ABWR design. The SQA plan

Instrumentation and Controls

establishes the criteria under which the other software development plans will be generated. The DFSER identified the following documents which are typical of a software development program.

- The software management plan (SMP) establishes the organization and authority structure for the design, the procedures to be used, and the interrelationships between major activities.
- The software configuration management plan (CMP) provides the means to identify software products, control and implement changes, and record and report change implementation status.
- The software development plan describes a development process, tool documentation, and products developed according to the software lifecycle.
- The verification and validation (V&V) plan describes the method to ensure that the requirements of each phase or stage are fully and accurately implemented in the next phase.
- The software safety plan describes the safety and hazards analyses that will be performed.
- The software operation and maintenance plan (SOMP) includes the procedures required to ensure that the software will be operated correctly and that the quality of the software is maintained.

GE combined these plans into an SMP, a CMP, and a V&V plan. The NRC will perform audits and inspections of these plans and their implementation at each appropriate phase or stage of the lifecycle.

The ITAAC activities completed by the COL applicant will be inspected by the NRC to verify conformance with the requirements at several stages during the digital control system design process. The documents which demonstrate satisfactory implementation of the ITAAC will be available for inspection during the NRC audit at the completion of each of the above stages. Figure 7.1-1 (from Section 3.4 of the CDM) of this report shows the stages or phases described by GE. The stages described in the DFSER, including the Figure 7.1-2 shows the stages described in the DFSER, including the NRC audit and the COL applicant conformance review points. These stages correspond closely with the phases described by GE in the CDM. The actual stages, including the conformance review and audit points, will be determined for each of the

software products to be developed when the design implementation is scheduled to begin. The COL applicant is required to satisfactorily complete ITAAC activities at each stage prior to proceeding to the next stage of the design development process. Failure to successfully complete the ITAAC at a stage, as determined by the conformance review or the NRC audit, may require repeating an earlier stage ITAAC and/or changing the system design. The NRC staff will issue an inspection report for each stage ITAAC and will identify any open issues which require resolution. Significant open issues which are not resolved could result in the NRC staff concluding that the ITAAC has not been satisfactorily completed.

At each stage, the COL applicant must verify that the design development is in accordance with the certified design process and that the detailed design developed (through that stage) is in conformance with the certified design. Upon completion of ITAAC activities for each stage, the COL applicant will certify to the NRC that the stage has been completed and the design and construction completed up through that stage is in compliance with the certified design.

Certain ITAAC, such as those for software development, will be repeated for all the products which use software. All of the products to be verified by the ITAAC process must be complete before the ITAAC itself is complete. The COL applicant will also provide a description of the next stage of design development and associated testing, analysis, and acceptance criteria in enough detail that the NRC staff can determine whether or not the proposed design development and testing are consistent with the certified design process and the subsequent ITAAC. This phased process will continue until all ITAAC steps for all the safety-related software are complete.

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

Figure 7.1.1-1 Integrated hardware/software development process

Figure 7.1-2 Flow of document through the software life cycle

The commitments described in the SSAR and related (docketed) documents provide acceptable methods and descriptions of the implementation of the CDM. The determination that the plant has been constructed in accordance with the CDM will require the use of the more detailed information contained in the SSAR and the documents which will be produced during the design phases. The SSAR commitments are based on postulations of how the design will be implemented which may change when detailed design begins. Therefore, some of the SSAR commitments may be changed. The acceptance of the ABWR design is based upon assumptions by the staff that the criteria that have been accepted will still be the proper criteria when the design is implemented. If a significant change in design process is selected by the design implementer, it may be appropriate to select different standards. Changes to SSAR commitments that affect the technical design of the I&C systems, including the design process, design implementation, and the NRC staff review as described in this report, must be submitted to the NRC for review prior to implementation. The majority of the SSAR material may be revised through a "50.59-like process" as described in SECY-92-287. This report specifically identifies those areas which must be reviewed by the staff prior to change by a licensee. The areas that the staff has determined must be submitted for review of proposed changes include computer design (hardware and software), multiplexor design, setpoint methodology, and EMC. These areas are designated with the following statement:

Any changes to this commitment would involve an unreviewed safety question and, therefore, require NRC review and acceptance prior to implementation. Any requested changes to this commitment shall either be specifically described in the COL application or submitted for license amendment after COL issuance.

The process to be used for software development and implementation is in compliance with the specified industry standards governing those activities (Tables 7.1-1 and 7.1-2), the requirements of 10 CFR Part 50, Appendix B, QA program for I&C systems and the requirements listed in CDM 3.4. In particular, the vendor implementation and the NRC evaluation of the safety-related software QA program is accomplished under a 10 CFR Part 50, Appendix B, QA program.

Based on the additional commitments to standards in the SSAR, increased level of design detail in the design description submitted, and the use of ITAACs (including NRC inspections at various stages) to allow design detail to be developed after design certification, the DSER open

issue on level of detail for digital I&C system design is resolved.

Prototyping

In discussing the level-of-detail open issue in the DSER (Open Item 1), the staff stated that it expected prototype testing of new technology to confirm expected safety performance would be required to confirm potential systems interactions, and to allow the staff to reach its safety determination on systems which may not have extensive operating experience. Based on the limited design information available for the interconnected RPS, ESF, EMS, and SSLC systems, the staff concluded that prototypes would be needed to demonstrate acceptable performance of these systems.

The term "prototype" has been used in several different ways. The staff identified several types of feasibility or demonstration testing that have at times been referred to by various parties, including the NRC, as prototype testing or prototyping. There are probably types other than the ones listed below that could be referred to as prototyping, but the staff believes that the following discussion will serve to clarify the issue of prototyping in the level-of-detail issue. The first example of prototyping, which occurs prior to design certification, is the need for a demonstration of the basic technology, or a demonstration of an example of a possible implementation of the proposed system. An example of this is the GE NUMARC digital equipment product line as a demonstration that microprocessors can be used to perform a function such as neutron monitoring which requires data transmission, comparison to setpoints, and logic decisions similar to the functions that will be required for the ABWR design. The staff considered the previous demonstrations of digital technology and does not require further demonstration of the feasibility of digital implementation of the I&C design. The staff also does not believe that there is a need to put acceptance criteria into the ITAAC for this prototype point. The design process description and the assessment of proven technology required in the equipment selection process is adequate.

The second example of prototyping occurs during the selection of the equipment vendors. There may be some prototyping to demonstrate capabilities or compatibility of specific hardware selected with the rest of the design. The staff does not believe that this demonstration is necessary to make a safety determination based on the system integration incorporated into the digital system design process.

The third prototyping example (probably at many places along the design implementation process) could include the development of specific system- aspect prototyping. For

Instrumentation and Controls

example, this may include breadboarding and testing specific components and different software languages to effectively implement the neutron monitoring algorithms. The staff does not believe that this type of prototyping needs to be described in the ITAAC because the final selected process or component will be assessed under the design process without a need to assess methods not selected for the design.

The fourth prototyping example could be identified on a typical design process as "perform hardware/software integration testing." The staff has concluded that this is an appropriate point to audit in the design process, and it will be part of the verification of the ITAAC at that stage to demonstrate that the design has been implemented correctly. This point includes a verification of the real-time performance capabilities of the system.

The items incorporated in the ITAAC include the development of the test and evaluation procedures as the design commitment, inspections to verify that the procedures are being used correctly as the inspection activity, and the completion of the testing, documentation of the results, and the feedback of the results to the design process as the acceptance criteria. The staff believes that the inclusion of this prototyping (hardware/software integration testing) in the ITAAC serves two primary purposes. The first is to verify that the design process is being implemented as certified and that the previous requirements have been met. The second is to establish the more detailed acceptance criteria for the next stage of the design process. The staff believes that this prototype needs, as a minimum, one full channel of the SSLC using hardware and software similar to the intended final product to adequately demonstrate that the design process stage has been completed.

The fifth example of prototyping (not usually referred to as prototype) is the final factory acceptance test. Factory acceptance testing will be covered by the ITAAC and has similar requirements to the previous prototype example, except that the test will be with the final software and hardware to be installed in the plant and would include all channels of the I&C system.

The last prototype example (also not usually referred to as prototype) is the final test and inspection, prior to fuel load, as installed in the plant. This test will be in the ITAAC and the SSAR Chapter 14 preoperational test program. The acceptance criteria will be a successful demonstration that all the original criteria in the design certification have been met.

GE agreed with the use of prototypes as described above and included prototyping in the ITAAC, as previously

stated. The required I&C system testing is described in the ITAAC and SSAR. The COL applicant will also perform a series of tests of the I&C system following fuel load under 10 CFR Part 50 guidelines that are not included in the ITAAC. Therefore, the issue of prototyping in the level-of-detail open issue (Open Item 1) is resolved.

7.1.4 Specific Findings

In DSER Section 7.4.1.2, the staff stated that the standby liquid control system (SLCS) lacked automatic initiation capability. Subsequently, GE added automatic initiation capability to this system and revised SSAR Section 7.4.1.2 to reflect this addition. However, SSAR Section 7.1.1.4.2, SLCS, was not revised to include the automatic initiation feature. Therefore, GE's commitment to revise SSAR Section 7.1.1.4.2 was identified as DFSER Confirmatory Item 7.1.4-1. The automatic initiation feature is also included in the CDM and will be verified during the ITAAC process. GE revised SSAR Section 7.1.1.4.2 to reflect automatic initiation capability for the SLCS. GE has also included this information in the SSAR and the staff finds it to be acceptable. This item is resolved.

In response to NRC Bulletin 88-07 (June 15, 1988), "Power Oscillations in Boiling Water Reactors (BWRs)," GE committed to implement the BWR Owners Group (BWROG) resolution which includes the installation of an Oscillation Power Range Monitor (OPRM) system. This was identified as DFSER Confirmatory Item 7.2.1-2. The SSAR and CDM have been revised to reflect this change and, therefore, this item is resolved. The resolution is discussed further in Sections 7.2.1 and 7.6 of this report.

The CDM for the neutron monitoring system (NMS) (including the OPRM) had not been submitted for review in time for the DFSER and, therefore, the OPRM CDM was DFSER Open Item 7.1.4-1. GE revised the SSAR to incorporate this information; therefore, this item is resolved.

The OPRM had not been included in the draft RPS ABWR TS and, therefore, this was listed as DFSER technical specification (TS) Item 7.1.4-1. The OPRM has since been added to the TS, therefore, this TS item is resolved. The TS is described in Chapter 16 of the SSAR.

SSAR Section 7.1.2.1.6, "Protection System Inservice Testability," described an integrated self-test provision built into the SSLC microprocessors. The SSAR described this feature as "safety associated" rather than "safety-related" consistent with the SSLC. The DSER identified this as an open issue (Open Item 7). GE committed to qualify the self-test features as Class 1E (safety-related) for

microprocessors performing safety functions and has revised the SSAR to reflect this commitment. This DSER open issue is, therefore, resolved.

Based on the above discussion, the staff concludes that GE has proposed an acceptable approach for design certification of the I&C systems under 10 CFR Part 52. All DSER and DFSER open items specific to the I&C systems design are resolved.

7.2 Reactor Protection System

7.2.1 General System Description

The reactor protection system (RPS) is described in Section 7.2 of the SSAR. Certain aspects of the RPS design are also discussed in SSAR Appendices 7A, 7B, and 7C. The RPS is also described in the CDM and is included in the TS. As with the other I&C systems, the RPS CDM provides the design description and ITAAC for the functional requirements of the RPS. Other CDM sections that also apply include the general I&C and EMS descriptions. The RPS includes those power sources, sensors, communication links, software/firmware, initiation circuits, logic matrices, bypasses, interlocks, racks, panels and control boards, and actuation and actuated devices that are required to initiate a reactor trip. The RPS is designed to automatically initiate the rapid insertion of the control rods of the reactivity control system to ensure that the specified acceptable fuel design limits are not exceeded. Manual initiation is also provided. The RPS also provides status information to the operator and status and control signals to other systems and annunciators. The RPS is qualified as a Class 1E safety system and will be environmentally and seismically qualified. The alternate rod insertion (ARI) capability via the fine motion rod control drive is discussed in Section 7.4 of this report.

The RPS is an open loop system with no feedback control other than operator initiation in response to display indication. The RPS performs four major functions: sense, command, execute, and display. At this highest level functional description, the ABWR RPS is similar to that previously reviewed and accepted for the GESSAR II design. The primary differences in the I&C portion of the ABWR design are: (1) the technology of the digital microprocessor-based logic and multiplexed data communications systems which is now used for most of the I&C design, and (2) the sharing of equipment for logic and display functions with other safety systems. A representation of the RPS configuration is provided in Figure 7.2-1 of this report for reference.

The RPS is implemented in the ABWR using the SSLC system which is also shared with the main steamline

(MSL) isolation valves (MSIV) and the engineered safety feature (ESF) systems. ESF systems which will share the SSLC are:

- (1) neutron monitoring system
- (2) process radiation monitoring (PRM) system
- (3) nuclear boiler system (NBS)
- (4) leak detection and isolation system (LDS)
- (5) residual heat removal (RHR) system
- (6) reactor core isolation cooling (RCIC) system
- (7) high-pressure core flooders system
- (8) reactor building cooling water
- (9) reactor service water (RSW)
- (10) HVAC emergency cooling water
- (11) diesel generator
- (12) electrical power distribution system
- (13) standby gas treatment system (SGTS)
- (14) atmospheric control system
- (15) safety-related heating, ventilation, and air conditioning (HVAC)
- (16) suppression pool temperature monitoring (SPTM) system
- (17) automatic depressurization subsystem
- (18) high-pressure nitrogen gas supply
- (19) fuel pool cooling and cleanup
- (20) radioactive drain transfer system
- (21) flammability control system

The following descriptions generally apply to the above ESF systems and the RPS. Differences are discussed in the ESF system Section 7.3 and other applicable sections of this report.

The SSLC is included in the I&C CDM. Figure 7.2-2 of this report is a representation of the SSLC logic and control block diagram (from the CDM) which shows the interconnection of the RPS and ESF systems. The figure also shows the inherent interrelationship with the EMS. Specific components and attributes of the SSLC are discussed in the following sections of this report.

The first primary function of the RPS is to sense the condition of certain parameters of the plant and provide accurate information to the command (comparison and calculation) section of the RPS, which in turn will initiate the execution of a reactor trip (scram) when the predetermined conditions have been met. The RPS will initiate a reactor scram when any one of the following conditions occur:

- (1) neutron monitoring system conditions exceed acceptable limits
- (2) high reactor pressure
- (3) low reactor water level (level 3)
- (4) high drywell pressure

Figure 7.2-1 Safety system logic and control block diagram

- (5) MSL isolation
- (6) low control rod drive (CRD) charging header pressure
- (7) high MSL radiation
- (8) turbine stop valve closed
- (9) turbine control valve fast closure
- (10) operator initiated manual scram
- (11) high suppression pool temperature

The specific setpoints at which a reactor trip decision is reached are discussed in the setpoint (Section 7.2.7) and TS (Section 7.11) sections of this report. The setpoint methodology is also included as a part of the I&C CDM and SSAR Appendix 7B. The specific setpoints to be used for plant operation will be determined and verified using a structured process described in the CDM after design certification but prior to reactor fuel load.

SSAR Section 7.2.1.1.4.2 lists the initiating RPS circuits. When the DFSER was prepared, the SSAR list included most of the above listed items plus a reactor scram on high seismic activity. GE has stated that this latter item was inadvertently included and would be deleted. This is acceptable to the staff and was identified as DFSER Confirmatory Item 7.2.1-1. The SSAR figures and several additional textual sections throughout SSAR Section 7.2 have been revised to reflect the removal of the seismic trip. This confirmatory item is resolved.

The high suppression pool temperature trip was added as an RPS signal after the DFSER was issued.

The NMS will provide separate, isolated, bistable startup range neutron monitor (SRNM) trip and average power range monitor (APRM) trip signals to all four divisions of RPS trip logics. The NMS provides trip signals directly to the divisional trip logic units (TLU). The NMS does not use the EMS to transmit its data. The NMS will also not use the digital trip modules (DTMs) for the determination of tripped or non-tripped status, but will perform that function within the NMS modules themselves.

The NMS provides trip signals and information from the SRNM and power range neutron monitor (PRNM). The PRNM includes both the local power range monitor (LPRM) and the APRM. The NMS monitors neutron flux from a source range of $1.E+3$ nv to 125 percent of rated power. Trip signals will be provided for SRNM upscale, SRNM short period, SRNM and APRM inoperative, APRM rapid flow decrease (reactor internal pump trip), and APRM upscale (flux and thermal power).

The staff determined that the SSAR includes commitments for compliance with IEEE 279-1971 for the NMS. In addition, in response to NRC Bulletin 88-07, "Power

Oscillations in Boiling Water Reactors," GE committed to implementing the BWROG resolution by incorporating an OPRM system in the ABWR design. This was DFSER Confirmatory Item 7.2.1-2. The OPRM has been added to the ABWR SSAR, and therefore, this item is resolved. The NMS (and the OPRM) is described in more detail in Section 7.6 of this report.

The second primary functions of the RPS is the command function, which is implemented using the logic of the SSLC. The RPS automatically initiates rapid insertion of the control rods to scram the reactor when warranted by any one of the predetermined conditions (listed above). The scram is initiated by means of four redundant divisions of sensor channels, trip logic, and trip actuators, and two divisions of manual scram controls and scram logic circuitry. In most instances, the EMS (described in Section 7.2.2.1 of this report) encodes the analog sensor channel output signal or contact position into a digital message, then transmits the message through an optical data link to a central decoder, which presents the signal to the DTM. In this manner, the EMS transmits data to the DTM (described in Section 7.2.2.2 of this report) from the reactor pressure transducer, reactor water level (Level 3) transducer, drywell pressure transducer, and CRD charging header pressure transducer. One exception to this EMS-to-DTM pathway is the manual scram, which bypasses both the EMS and DTM, since it is hardwired from the main control panel directly to the output logic unit (OLU). Other exceptions include the main steam isolation valve position switches, which are hardwired to the DTM, as are the MSL radiation monitors, turbine stop valve closure sensors, and turbine control valve fast closure sensor. Signals from these sensors are transmitted directly to the DTM, without passing through the EMS. However, the SSAR text and figures originally stated that the turbine inputs were multiplexed. This was DFSER Confirmatory Item 7.2.1-3. GE revised the SSAR to state which sensor inputs use the EMS and which are hardwired. The revised SSAR states that the turbine inputs are hardwired. This item is, therefore, resolved.

The DTM compares the sensor data (either received directly from the sensor or through the EMS) to a preestablished setpoint and determines if the individual sensor is in a tripped or non-tripped state. The data link between the sensors and the DTM is a fiber distributed data interface (FDDI), a bi-directional, fiber optic link in which two redundant token ring networks provide the data path protocols. Each sensor input is multiplexed through each of the two links so that the loss of a single direction of an EMS link will not prevent the SSLC from receiving the sensor information.

Figure 7.2-2 Safety system logic and control (SSLC) control interface diagram

The RPS is a four division system where each parameter is monitored by four sensor channels, one in each division. GE defined "sensor channel" in the TS as sensor, data acquisition, and data transmission hardware and software to the input of a bistable or voter in the TLU or safety system logic unit (SLU). A division of sensor channels is the total set of sensor channels which are input to the SSLC for the RPS, MSIV actuation and ESF. Each division of sensor channels is powered from the respective Class 1E power supply. Unless otherwise specified, a "division" of RPS (or MSIV or ESF) is the set of sensor channels, the logic channel, and output channel which are powered from the same division of electrical power. This definition of division also specifies physical and electrical separation. To ensure consistency, a "logic channel" is defined as the hardware and software that process the sensor channel inputs to produce an identifiable actuation signal within a division.

Certain sensed parameters have more than the four individual division sensors which would provide one signal per logic per channel input. The SRNM, for example, has ten channels which feed into the four RPS logic channels (three each to Divisions I and III and two each to Divisions II and IV). The MSIV position and turbine control valve fast closure each provide eight logic channel inputs.

Trip signals from the four channels of DTMs (and the NMS input) form four two-out-of-four coincidence logic matrices, one for each division of trip logic. A trip output from a DTM will also indicate the parameter which has exceeded its setpoint. If the parameter is the NMS, the indication to the operator will be that the trip was caused by the NMS without specifying the particular aspect within the NMS which caused the trip. The two-out-of-four comparison is accomplished in the TLU and the voting is done by local coincidence logic which requires the two tripped signal inputs to be from the same parameter. Because the RPS is designed to be fail-safe, the actual implementation of the logic is three-out-of-four channels not tripped. The TLU is described in Section 7.2.2.3 of this report. Trip signals from each division TLU go to the RPS OLU where a hard logic (not software/firmware) unit (similar in concept to the GESSAR II design) interconnects the divisional two-out-of-four TLU trip signals to the control rod scram pilot valves. The OLU is described in Section 7.2.2.4 of this report. Each final scram logic provides four output signals, each of which operates a load driver. A scram occurs when two or more divisions of TLUs are tripped. The two-out-of-four vote will occur twice -- once in a vote of sensor parameters in each TLU and again in a vote of RPS division output logic at the load drivers.

The RPS interfaces with the CRDs through its own pair of solenoid scram pilot valves to perform the "execute" function. There are two scram valves for each CRD. A rod scram is initiated when both (two-out-of-two) solenoids of a scram pilot valve are de-energized. When de-energized, the scram pilot valve vents the air that holds the scram valve closed. Opening of the scram valve allows the pressurized (at greater than RCS pressure) control rod water to act on the CRD piston resulting in the rapid insertion of the rods. With this arrangement, a scram of all rods is initiated if any two (or more) of the four divisional logics (TLUs) are tripped.

The OLU output is used as an input to two backup scram logic circuits. The backup scram is an energize-to-scram logic arrangement. The backup scram logic is 125 Vdc powered rather than the de-energize to scram 120 Vac power of the scram pilot valves discussed above. When the relays for the backup logic are tripped, the relay contacts will energize the air header dump valve solenoids and initiate the scram. This is a diverse means of reactor scram.

The ARI can also automatically (and manually) insert the control rods by means of the fine motion control rods or the ARI scram valves in case of off-normal conditions and is discussed in Section 7.4 of this report. The SLCS can also automatically (and manually) shut down the reactor and is discussed in Section 7.4.1.2 of this report.

For manual scram, four push-button switches are provided, one for each divisional trip logic. Additionally, hardwired manual scram is provided by two pushbuttons in the power supply circuits for the pilot scram solenoids. Actuation of any two (or more) of the four divisional switches or both hardwired manual scram pushbuttons will scram the reactor by means of the load drivers for automatic scram. The reactor can also be scrambled by setting the mode switch to the "shutdown" position. These provisions in the RPS design for manually scrambling the reactor conform with the requirements of IEEE-279 and the guidelines of RG 1.62 on manual initiation of protective actions.

The display function of the RPS will be accomplished with a combination of Class 1E fixed mimic displays and Class 1E divisional visual display units (VDUs). SSAR Table 18F-1 (also 18F-2 and 3), "Inventory of Controls (displays and alarms) based upon the ABWR EPGs and PRA," provides a listing of the inventory of controls and displays required to execute all emergency procedure guideline (EPG) steps, the alarms required to alert the operator to perform the steps, and the displays to judge that the actions have been initiated or accomplished. The human factors aspects of the controls and displays, and the staff conclusions are addressed in SSAR Chapter 18 and

Instrumentation and Controls

the corresponding section of this report. There are also RPS inputs to the non-Class 1E plant computer through isolation devices.

The RPS will be periodically tested during plant operation as defined by the TS discussed in Section 7.11 of this report. In addition to the standard operator initiated surveillance, the safety system logic portion of the RPS is tested by the self-test subsystem (STS). The STS is a software-based self diagnostic system which continuously and automatically tests the SSLC components and interconnections. A more complete description of this system is discussed in Section 7.2.2.5 of this report. The provisions for testing in the design of the RPS and the periodic testing of the RPS, as described in the SSAR, conform with guidelines of RG 1.22, "Periodic Testing of Protection System Actuation Functions" dated February 1972 and IEEE Standard 338, "Criteria for the Periodic Testing of Nuclear Power Generating Station Safety Systems" as supplemented by RG 1.118, "Periodic Testing of Electric Power and Protection Systems" Revision 2. The commitments in the SSAR to the design basis requirements for the capabilities for sensor checks and test and calibration are consistent with the requirements of GDC 21, "Protection System Reliability and Testability."

Complete electrical and physical separation must be maintained between the four RPS divisions in order to meet the criteria of IEEE Standard 279-1971. This requirement is included in the CDM and will be verified during the implementation of the ITAAC. The staff concluded in the DSER that isolation of information (error handling) must also be addressed because of the extensive use of multiplexors and software. This aspect of the multiplexors is discussed in Section 7.2.2.1 of this report.

A trip of any sensor or logic channel will be annunciated and will cause that channel to lock in the trip mode until manually reset. The RPS will be fail-safe in that a loss of power to a channel will result in that channel going to the tripped condition. Other failures such as a break in a communications link will be detected and the self diagnostic of the individual microprocessor will put the output to the tripped state.

The design of the ABWR RPS is significantly different than in previous BWRs in the method the high level functional design is implemented. The staff reviewed the SSAR and found that conceptual design description to be a functional block diagram level of detail. The staff found that the hardware design documents for the SSLC did not state details such as bus protocol, bus data capacity, provisions for hardware level interrupts, the size of the memory, the speed and size of the microprocessor, the format of the status panel, hardware based interlocks, and

type of display media. The means of addressing the above design details for design certification is described in Section 7.1 of this report and is discussed further below.

The functional level block diagram of the safety-related signal paths is simple and direct. However, in a microprocessor-based system, the software implied in the blocks of the system diagram can mask much of the safety system's design complexity. Several of the significant issues involving the use of microprocessor-based systems and safety functions have been presented to the Commission in SECY-91-292 (September 16, 1991), "Digital Computer Systems for Advanced Light Water Reactors."

The staff concluded in the DSER that the SSLC design was not "essentially complete" as required by 10 CFR Part 52 because of the complexity of the ABWR I&C system design and the lack of design details for certification. The staff found that the design description presented for the SSLC was ambiguous because the design material was dispersed in the submittal and often contradictory. For example, data about the TLU trip (RPS division trip) status was not originally listed in the SSAR as an output to the operator.

The staff reviewed the SSLC system design specification, which was described by GE as a procurement level document. The staff considered the documentation for the SSLC to be inadequate for design evaluation and not in conformance with the design certification requirements for level of detail. The staff, therefore, concluded that the software design and development aspects of the SSLC were not described to a sufficient level of detail. GE did not adequately disclose the details of the design to enable the staff to reach a final conclusion on the acceptability of the design for certification, and this was an open issue in the DSER.

Since the issuance of the DSER, GE has provided additional detail, committed to industry standards appropriate for the digital system design, and submitted the CDM which includes the I&C system design process. The staff concludes that the combination of the improved level of detail provided, the commitments to standards, and the incorporation of the design process into ITAAC with extensive NRC auditing during design development and implementation provides the basis for a final safety determination for certification, and is sufficient to resolve this issue. In Section 7.1, the staff discusses the issue of level of design detail. This open issue from the DSER is, therefore, resolved.

The staff also reviewed the SSAR for conformance to IEEE Standard 279-1971 and ancillary requirements for

meeting the single-failure criterion, independence, control and protection system interaction (isolation), testing, bypass and bypass indication (including removal of bypass), and manual initiation. To meet the single-failure criterion described in Section 4.2 of IEEE-279 and IEEE-379, "Standard Application of the Single-Failure Criterion to Nuclear Power Generating Station Class 1E Systems," GE committed in the SSAR to compliance with RG 1.53, "Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems." The SSAR states that the RPS will comply with GDC 22 on protection system independence, the requirements in IEEE-279 on channel independence, and the guidance for physical and electrical independence of the instrumentation system in RG 1.75, which endorses IEEE Standard 384. Conformance to RG 1.47, "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems," is described in the SSAR and discussed further in Section 7.2.3 of this report. The RPS design includes provisions to meet the requirements of IEEE-279 on channel and operating bypasses. The RPS performs no control functions, interlocks with control systems are to be through isolation devices, and the channels of the RPS are to be electrically isolated and physically separated in order to meet the criteria of GDC 24 on separation of protection and control systems and the requirements of IEEE-279 for control and protection system interaction. The RPS description and drawings in the SSAR describe a clear commitment to the above requirements. The CDM describes the GE commitments to the most significant requirements from the various standards and criteria (though not the specific standards themselves). These commitments will be verified during the ITAAC implementation. The standards themselves and other less significant criteria referenced in the SSAR and this report are expected to change over time and are not "locked in" by the CDM.

In the DSER, the staff noted that the SSAR did not present a detailed failure modes and effects analysis (FMEA) to ensure that all postulated failures result in a known safe state if the RPS experiences conditions such as disconnection of the system, loss of power, or exposure to a postulated adverse environment. Therefore, the staff could not evaluate conformance to GDC 23, "Protection System Failure Modes." This was Open Issue 3 in the DSER. Open Issue 3 has since been resolved by the FMEA submitted by GE and included in the SSAR, and additional studies performed by the staff. As a result of these studies, the staff determined that some aspects of this issue relate to the potential for common-mode failure of the I&C system. Consequently, the DSER issue concerning the lack of a FMEA was resolved, but the potential common-mode failure and the required redundancy and diversity issues were identified as DFSE

Open Items 7.2.6-1 and 7.2.6-2. These open items are resolved in Section 7.2.6 of this report. The staff concludes that the RPS design meets the requirements of GDC 23, "Protection System Failure Modes."

When the staff issued the DFSE, the figures (interface electrical diagrams (IEDs)) and interface block diagrams in SSAR Section 7.2 lacked many details to be submitted later. This was DFSE Confirmatory Item 7.2.1-4. The figures have been corrected in the SSAR and this item is, therefore, resolved.

7.2.2 Safety System Logic and Control and Specific Subsystem Descriptions

The SRP and the SSAR format are established along functional system boundaries. The ABWR I&C system is significantly different from the I&C system designs in plants when the SRP was last revised in 1984 in that the safety functions are combined the SSLC, into a common computer-based logic, control, display system. Consequently, the ABWR I&C system does not follow the discrete system boundaries assumed in the SRP. The SSLC serves both the RPS/MSIV and ESF systems. The following sections describe common elements of the SSLC which are used for the RPS. Although GE does not consider the EMS to be part of the SSLC, it is inherently interconnected with the SSLC and, therefore, is included in this section of this report.

The staff reviewed some of the I&C system documents referred to as master parts list (MPL) documents. These documents are referenced as supporting documents in the IED figures in the SSAR. The MPL documents reviewed and discussed throughout this report were prepared for the Tokyo Electric Power Company, Inc., Kashiwazaki Kariwa Nuclear Power Generation Station, Units Nos. 6 and 7. These documents list applicable Japanese laws, regulations and standards. In addition, the Japanese standards list the U.S. standards which are either specifically accepted in the MPL documents or are referenced as guidelines. For example, MPL C71-4010 (Rev. 1, July 2, 1990), "Reactor Protection System Design Specification," states acceptance of 10 CFR Part 50, Appendix A (GDC) but refers to IEEE 279, IEEE 603, and RG 1.152 as standards that form a part of the document or are listed as references without a specific statement of acceptance or conformance. Several items in the MPLs are specific to the Japanese design and, therefore, will require revisions to refer to U.S. criteria if these documents are used in the ABWR design. For example, the Japanese plant RPS is specified to use a 50 HZ, 120 Vac power supply instead of a 60 HZ supply as specified in the ABWR SSAR. The MPL document also includes the seismic scram function which is not

Instrumentation and Controls

included in the ABWR design. The staff has not listed all such discrepancies in this report. The verification process in the ITAAC will be used to confirm that the MPL design documents, when developed and used by the COL applicant, will be in accordance with the SSAR and CDM. Where necessary, the staff has requested GE to clarify the standards commitments in the SSAR for the ABWR so reliance in the SSAR is sufficient for the safety finding. Standards commitments are discussed throughout this report.

MPL ABBE-4080 (Rev. 0, April 25, 1990), "EMS/SSLC Interface Requirements, Requirements Specification," states that system timing shall be a synchronous between the EMS and the SSLC. The interface between any one channel of the EMS and the corresponding channel of the SSLC may be either by hardwired or by fiber optic cable. The SSLC will determine from which of the dual EMS channels (per division) to select the information. The selection will be based upon data quality checks. The data quality checks may include checksums, parity checks, rate limiting checks, and the de-bouncing of digital inputs. The DTM will perform these functions except for the display functions which are taken directly to the displays from the control room multiplexing unit (CMU). Typical data formats are presented with options from which to select, dependent on the final design input and output parameters. The above information is consistent with the SSAR description.

The staff reviewed MPL A32-4080 (Rev. 1, April 24, 1990), "Safety System Logic and Control Design Specification." The document is in general conformance with the SSAR and CDM but does not add a significant amount of design information about the SSLC. Many of the design procedures are described as items to be developed later. This document lists topics to be considered when the design is developed and is similar to the SSLC CDM. The MPL document is, therefore, appropriate for the intended purpose.

7.2.2.1 Essential Multiplexing System

As described in SSAR Chapter 7.2, the EMS portion of the RPS system transmits data to the SSLC from the sensors that are not hardwired or part of the NMS. SSAR Appendix 7A discusses the EMS system. Additional information is included in SSAR Appendix 19N, "Analysis of Common-Cause Failure of Multiplexing Equipment," SSAR Chapter 7 Appendix C, and the MPL documents discussed below.

Data multiplexing is an integral part of the ABWR I&C system design. Multiplexing systems will transfer data between sensors and actuation control devices distributed

throughout the plant, and the logical processing units in the control room. The multiplexing system applications procedure (MPL Document No. A11-4120) and the SSAR define two multiplexor systems. The multiplexing system tasks are divided between the EMS and the Non-Essential Multiplexing System (NEMS).

The EMS receives inputs only from safety systems. Any outputs to non-safety systems are electrically isolated. The EMS is not listed as a separate safety system in GE Table 7.1-2 of the SSAR, but is considered part of each safety system for which it provides data communications. The EMS interfaces with the RPS/MSIV and ESF systems, and is qualified to the same quality standards as the RPS, ESF, and SSLC systems. The EMS is powered from the Class 1E 125 Vdc buses. The EMS transmits approximately 1500 signals either from the sensors to the SSLC or from the SSLC to the actuated equipment (for the ESF functions only). The EMS has a separate CDM description in addition to those aspects of the EMS included in the functional requirements of the RPS CDM and functional and design process requirements of the SSLC (I&C) CDM.

The SSAR states that the NEMS is used in the non-safety-related control systems and is not considered a safety system. The NEMS is discussed further in Section 7.7 of this report.

The EMS is comprised of four independent divisional multiplexing systems, each of which has redundant data links within the channel. In the DSER, the staff concluded that GE had not committed to appropriate industry standards for certain aspects of the EMS design, including the multiplexors themselves. GE has since committed to a deterministic, dual redundant, fiber optic ring structure which will follow the guidelines of industry accepted Standard American National Standards Institute (ANSI) ASC X3T9.5 (1988), "Fiber Distributed Data Interface (FDDI)." As noted in previous GE responses to staff requests for additional information, the design is not specifically constrained to this particular format and configuration. Specifically, the CDM does not require a specific design at this time. Rather, the CDM provides functional requirements with the SSAR providing more specific information at the functional level of detail. The SSAR also refers to IEEE 802.5 (1985), "Token Ring Access Method and Physical Layer Specifications," if a lower throughput of information is incorporated in the EMS design. Both options are consistent with International Standard Organization (ISO) 7498 (1984), "Open Systems Interconnection - Basic Reference Model," as the data link layer and physical layer. IEEE 802.2 (1985), "Standard for Local Area Networks: Logical Link Control," will define the protocols necessary to move data

to the higher levels of the ISO model. Any changes to these standards commitments would involve an unreviewed safety question and, therefore, require NRC review and acceptance prior to implementation. Any requested changes to this commitment shall either be specifically described in the COL application or submitted for license amendment after COL issuance. For example, if a new ANSI FDDI standard is developed which uses a shorter wavelength than the current standard, the COL applicant or holder will need to verify to the staff that the dispersion will still be small enough to meet the FDDI's bit error rate.

The data link and physical layers are part of the general seven-layer hierarchical model of the open systems interconnection which the FDDI protocol conforms to. The seven layers are application, presentation, session, transport, network, datalink (including logical link control and media-access control), and physical (including physical and physical medium dependent). The FDDI standard (ANSI ASC X3T9.5) also requires a station management function at each station to supervise sub-layer and ring management operations. The descriptions that GE has presented, with the caveat that this is but one of a number of possible implementations, are in general conformance to the above standard to the level of detail provided. These standards are supported by the industry.

Vendors currently offer chip sets to implement the individual FDDI sublayers, as well as complete systems that conform with the above standards. FDDI controllers are available with several different computer buses. The staff concurs with GE that no additional prototyping is required to demonstrate the feasibility of the design because the EMS will use proven technology. The capacity of the FDDI should be capable of handling the quantity of inputs/outputs at the rates necessary for proper operation of the SSLC. This will be verified by testing during the ITAAC implementation. The prototype requirements described in Section 7.1 of this report still apply.

The staff concludes that the ABWR SSAR references and contains commitments to appropriate standards for the EMS design at this level of design detail and, therefore, this open issue (Open Issue 5 regarding adequate commitments to industry standards) from the DSER is resolved. The implementation of the design in accordance with the standards referenced above and other ABWR SSAR statements will be verified during the ITAAC. The multiplexor design will also be verified through a staged ITAAC as discussed in Section 7.1.3.3 of this report. When the final design development is initiated, the COL applicant will submit the design plans for staff review and for development of the ITAAC verification audit points.

As a data highway, the EMS multiplexes data from the sensors to the control room logic units, and multiplexes the commands from the control room computers to the appropriate actuation control devices (for the ESF actuation only). The RPS (and the MSIV initiation) does not use the EMS for the output of the SSLC to the scram pilot valve initiation. With the exception of hardwired input connections (turbine stop valve closed, turbine control valve fast closure, MSIV position, manual scram, and the neutron monitor systems input) all of the RPS (and most of the ESF) sensor inputs will be processed by the EMS.

The hardware architecture of the EMS described in the SSAR uses fiber optics for the communications medium and microprocessors for the node controllers. The SSAR also states that the ABWR EMS may use coaxial cable or twisted-pair connections. If copper wire is used in place of the fiber-optic cable, electrical isolation will be maintained and provided where necessary (and demonstrated by testing of the devices in accordance with the Licensing Review Bases (LBB), Appendix B) thus resulting in potentially additional isolation devices. Fiber-optic cable is an inherently excellent electrical isolator, and its use to the extent practical will reduce the need for additional isolation devices.

The major components of the EMS are the remote multiplexing units (RMU), the CMU and the fiber-optic (or copper) cables. The RMU and CMU will contain the following items: transmission line interface circuits, processors, memories, signal conditioning circuitry, analog to digital conversion, and watchdog timers. The specific number of RMUs has not been determined. The only specific EMS requirement identified at this time is that within each division of EMS, the wide-range and narrow-range reactor water level sensors will be processed by different RMUs. This will provide an additional level of defense-in-depth such that any single RMU failure will only disable one of the water level indications and SSLC inputs. All of the components of the EMS are identified as Class 1E. At the RMU, an input sensor interfacing is provided to switch contacts, 4-20 ma current loops, thermocouples, RTD devices, pulse inputs, and voltage inputs. The RMU output interface (which is not used for RPS functions) is to relays, solenoids, voltage to current converters, panel meters, and indicator lights. All equipment will be rack mounted and operated from the divisional 125 VDC Class 1E power sources.

The RMU will be located in mild environment areas throughout the plant near the sensors and the actuation control devices with which they interface. The RMU and CMU are connected to the fiber-optic cable, and together control transmission to assign which RMU/CMU transmits or receives signals. The RMU multiplexes and sends data

Instrumentation and Controls

messages to the control room via the fiber-optic cable. In the control room, the CMU demultiplexes the message and sends it to the appropriate DTM of the SSLC. The DTM will select which of the redundant CMU signals it will use. Control signals from the SLU for the ESF functions of the SSLC are sent to the CMU which in turn transmits these control signals to the appropriate RMU.

The EMS design concept identifies the RMU and CMU as sharing a modular, microprocessor-based, bus-oriented architecture, using similar modules. The RMU and CMU are configured somewhat differently with the CMU having additional communications modules instead of input and output modules. The following functional modules will be in a typical RMU/CMU:

- a. Input - acquires analog and digital data from the sensors.
- b. Output - transmits control signals to equipment actuation control devices (ESF function only).
- c. CPU - processes the signal data, coordinates I/O and communications, and performs calibration and diagnostics.
- d. Memory - contains the stored program executed by the microprocessor and stores intermediate data.
- e. Communications - formats and multiplexes data that is sent through the serial optical link.
- f. Front Panel Interface - permits technician access to calibration and diagnostic functions.

GE stated in SSAR Chapter 20, that the RMU failure mode was dependent on what sensors were connected to it, either fail-safe (negative, low) or fail-as-is (last reading). The particular failure state for the RPS will be fail-safe and will be verified during the ITAAC implementation.

GE stated in SSAR Chapter 20, that the EMS includes self-test software that detects malfunctions in the hardware modules and provides alarms in the control room. GE has not presented specific information on the types of malfunctions for which testing is being provided for or the EMS reaction to each malfunction (e.g., restart, alarm, fail-safe, or fail-as-is). However, GE has provided a listing of typical parameters that are monitored. These include the status of the central processing unit, parity checks, data plausibility checks, watchdog timer checks, memory checks, voltage level checks, and data range and boundary checks. The self tests also provide the capability for internal testing and error checking within the EMS. In addition, the EMS is subject to routine periodic operator

initiated surveillance checks required by the TS. Also, as described in the simplified FMEA in Chapter 15.B.4 of the SSAR, the EMS will reject corrupted signals in the transmission. Further, the DTM, which receives inputs from the EMS, uses a data quality check to determine which of the CMUs it will accept data from.

The selection of the EMS hardware was not identified in the SSAR or supporting material. However, the top level specification documents referenced the design control and hardware reliability standards and criteria required to be met. In response to staff questions (Q) (Q420.92, Chapter 20 of the SSAR), GE committed to MIL-HDBK-251 and to MIL-STD-217E for EMS hardware reliability and thermal effects.

The staff identified an open issue in the DSER based on a concern that common-mode failure from effects such as electromagnetic interference (EMI/EMC) or design error could cause the loss of the EMS in more than one division. In response, GE performed an assessment of the effect on the ABWR I&C systems for a loss of all four divisions of the EMS. The GE study (SSAR Section 19n) concluded that under such a condition, the plant could be safely shut down from the remote shutdown system. This response was insufficient to fully resolve the common-mode failure concern, and further information was requested from GE. The concern about EMI/EMC, reliability and thermal effects was eventually resolved as discussed in Section 7.2.8 of this report. The concern with potential common-mode failure of the EMS (and other equipment) was also eventually resolved as addressed in Section 7.2.6 of this report.

The SSAR defines the top level hardware architecture for the EMS but presents limited information for the software that runs in the EMS microprocessors. GE stated the following high level design goals for the EMS software:

- a. real time executive kernel
- b. hierarchical task structure
- c. simple modules
- d. on-line calibration with bypass
- e. self-test as background process
- f. automatic recovery

The ABWR EMS uses microprocessors to control the movement of data from the sensors to the SSLC, between the different components in the EMS itself, and from the SSLC logic to the actuated devices (for the ESF systems). The high level block diagrams of the data signal paths are simple and direct. However, in a microprocessor-based system, the software implied in the blocks of the system diagram can mask much of the safety system's design complexity. In a microprocessor-based data transport

AIM

Association for Information and Image Management

1100 Wayne Avenue, Suite 1100
Silver Spring, Maryland 20910
301/587-8202

Centimeter

Inches

MANUFACTURED TO AIM STANDARDS
BY APPLIED IMAGE, INC.

4 of 10

system, the software is an essential line element in the execution of the safety system functions. For this reason, the staff concluded in the DSER that the design of the EMS did not provide the appropriate level of detail essential to the staff's review. This was identified as Open Issue 1 in the DSER. Since the DSER, GE has committed to the use of several additional software development standards as stated in the SSAR. This commitment, in conjunction with the design process, corresponding ITAAC steps, and staff audit verification, resolved this issue. The topic of hardware and software qualification is addressed further in Section 7.2.8 of this report.

The SSAR initially did not present an architecture for the software design. GE had not demonstrated how the decision logic, a parallel process in an analog system, would be implemented by the software, which is usually a serial process. The staff considered this an open issue in the DSER related to level of detail. GE has responded to this issue by providing the timing requirements for each stage of the control process from sensor to scram initiation, and demonstrating that the scram time requirements (50 msec) can be met with the EMS and the SSLC. The decision logic will be accomplished using a deterministic serial process. The timing requirements will also be demonstrated in the ITAAC implementation. This is acceptable to resolve the issue of serial vs. parallel processing.

The following is a list of software elements that process a signal as it moves from the sensor through the EMS to the SSLC control logic for the RPS.

a. Remote multiplexer unit - input

1. operating system/executive
2. signal acquisition
3. signal conditioning
4. analog to digital conversion
5. digitized signal data preparation for transmission
6. digitized data transmission on fiber optic link

b. Control room multiplexer unit - input

1. operating system/executive
2. digitized data acquisition from fiber optic link
3. data preparation for SSLC/DTM
4. data transmission to SSLC/DTM
5. multiplexed data transfer control
6. display information transmission

c. DTM processing by SSLC software modules.

A basic design parameter of a local area network is whether synchronous or asynchronous transmission is used. The staff identified concerns about this issue in the DSER, and in particular was concerned about the use of synchronization across channels which could violate the single-failure requirements of IEEE-279 and IEEE-603. GE responded by stating in the SSAR and CDM that the EMS will run asynchronously. There will be no common clock shared between channels. The inherent time skew in the logic processing that will result by not synchronizing the channels will be controlled by maintaining a significantly high sample rate and storing the last four validated variables in each TLU for each vote. There is also no common clock within a division, however, the RMUs will append timing information to the sensor information to be multiplexed to the CMUs. The staff finds this acceptable, and the DSER issue of synchronization is, therefore, resolved.

Although the detailed design of the EMS depends on the hardware that is selected, the functional requirements for the EMS as part of the ABWR safety systems are not hardware dependent. Because the detailed EMS hardware design was not provided, a concern regarding level of detail was identified as part of Open Item 1 in the DSER. In addition, Open Item 2 identified the need for design information about the isolation of corrupted data in multiplexors. In response to these issues, GE has defined the high level functional requirements of the EMS, the major parameters that define the data transmission attributes, and the criteria for selecting the data transmission hardware.

GE stated several functional requirements for the EMS as follows:

1. The EMS is to be implemented such that any single failure or any single channel removal within the system shall not prevent proper action of the RPS at the system level.
2. The EMS in each division will be independent, and physically and electrically separated from the other divisions to preclude failures in one EMS from propagating to another division.
3. Non-safety system failures are to be isolated from the safety systems.
4. The capability for testing during power operations is to be provided.

Instrumentation and Controls

5. The EMS will satisfy all the environmental operability requirements (seismic, temperature, humidity, radiation level).
6. The EMS will have adequate capability to withstand surges in accordance with IEEE-472.
7. The EMS will have a designated temperature range of 10 to 40 °C. (50 to 104 °F), for relative humidity of 10 to 60 percent for the CMU and of 10 to 90 percent for the RMU.

The staff concludes that the combination of additional information on EMS functional requirements, commitment to appropriate additional standards and the ITAAC process provide the necessary information on the EMS design and, therefore, the related part of DSER (SECY-91-294) Open Items 1 and 2 are resolved.

The two loops of the network are designated "master" and "standby" by the receiving fiber optic interface. The designation of which loop is "master" is on the basis of transmission errors, checksum errors, and other tests in the self diagnostics. The diagrams that the staff has reviewed showed that each DTM in the SSLC has two fiber optic interfaces. The "master" designation is also applicable at the RMU level where ESF equipment actuation commands are received. It was not initially clear to the staff how these two fiber optic units arbitrate between themselves to determine which is to be the "master" loop. GE stated that the decision will be based upon a data validity check at the DTM. The specifics of what will be valid data for each parameter was not provided. This was Open Item 6 in the DSER. GE has committed to the FDDI standard for data validation guidance which the staff finds acceptable to resolve this issue. The correct implementation and demonstration by test of the master/slave switch will be included in the ITAAC for multiplexors. DSER Open Item 6 is, therefore, resolved.

MPL DMH-4270 (Rev. 2, February 3, 1989), "Essential Multiplexing System Design Specification," provides various EMS design details. It states, for example, that the bit error rate shall be under $10E-9$ "theoretically." This is consistent with the FDDI guidelines. The CDM commitment is to a functional demonstration of capability without specifying particular throughput or error rates. The MPL document also states that the RMU and CMU shall have the capability to transmit the data at the rate of one megabit/sec (Mbps). The FDDI standard is written for essentially an 100 Mbps system. The difference in data transmission rate between the GE MPL and the FDDI standard is an example of an MPL revision that will be required and verified during the ITAAC implementation in order to ensure consistency with the CDM. The EMS will

be capable of sampling a sensor with a 10-msec sampling rate, but the actual sampling rate for each sensor has not yet been determined. The sampling rate will be established as part of the detailed design by the COL applicant. The maximum transmission distance for any single multiplexing station is specified in the MPL as 500 m (approx. 1640 ft). A maximum distance verification was to be included in the ITAAC. This was DFSER Confirmatory Item 7.2.2.1-1. Although the CDM has not been revised to specifically reflect the distance verification, the functional requirements in the CDM and an SSAR change restriction are acceptable to the staff and this item is, therefore, resolved.

The need to incorporate consideration of instrument channel inaccuracies due to A/D converter in the setpoint methodology was DFSER Confirmatory Item 7.2.2.1-2. GE revised CDM Section 3.4, "Instrumentation and Controls" to include setpoint methodology and specifically includes consideration of the A/D converter accuracy. This item is, therefore, resolved.

MPL A32-4080 (Rev. 0, April 25, 1990), "EMS/SSLC Interface Requirements, Requirements Specification," specifies that the four channels of the EMS be asynchronous. This is consistent with the SSAR and the CDM. It also requires a fixed format for both the messages and the sequence of the messages. This is consistent with the general GE commitment to use a deterministic EMS. The MPL notes that the RMUs shall provide appropriate filtering without providing a specification of what is appropriate. Providing appropriate filters will be part of the design development and implementation process which is the responsibility of the COL applicant.

The EMS design of the ABWR uses microprocessors and related software throughout the safety systems. While promising improvements in performance and reliability, this advanced technology also can introduce problems and failure modes that had not been addressed before in reactor safety system design. Several of the communications control functions performed within the EMS are implemented by software. RG 1.152, which endorses ANSI/IEEE ANS Standard 7-4.3.2 (1982), has been promulgated as the standard for developing software for safety systems. However, GE's commitment to this standard was ambiguous. There was no evidence in the SSAR that the software development for the EMS would be in accordance with IEEE Standard 7-4.3.2. This was an open issue in the DSER (SECY-91-294). GE subsequently removed the ambiguity from the SSAR commitment to IEEE Std. 7-4.3.2 and added additional software requirements to the SSAR which are discussed in Section 7.2.8 of this report. This issue is, therefore, resolved.

In the DFSER, the staff stated that the sensor data transmitted through the EMS should be clarified in Chapter 7.2 of the SSAR. For example, the turbine stop valve closure input was described in Section 7.2.1.1.4.2 (6) as using the EMS, which was not consistent with other information provided by GE. This was DFSER Confirmatory Item 7.2.2.1-3. GE has revised the SSAR to specifically state which types of data are transmitted through the EMS and which types one not. This item is, therefore, resolved.

GE has committed in the SSAR that the EMS design will meet the relevant SRP criteria and will meet current industry standards for multiplexing. GE also committed to additional hardware and software criteria beyond those contained in the SRP. Based on the above, the staff concludes that the EMS will be capable of accomplishing the RPS performance requirements. Verification of the final design against the above criteria will be performed during the ITAAC implementation. Any changes to the commitments described above concerning the performance specifications and architecture of the EMS would involve an unreviewed safety question and, therefore, require NRC review and acceptance prior to implementation. Any requested changes to commitments involving the EMS performance specification or architecture shall either be specifically described in the COL application or submitted as a license amendment after COL issuance.

7.2.2.2 Digital Trip Module

The digital trip module (DTM) is a microprocessor-based unit which performs the basic comparison of the sensors to the preestablished setpoint, and provides a trip or no-trip output for that parameter. For the RPS, there is one DTM for each set of sensor channels (a division of sensors, refer to Figures 7.2-1 and 7.2-2 in this report). The DTMs for the ESF functions are separate and are not shared as was the case for the EMS. The DTM, as with all of the SSLC, is Class 1E. The hardware and software requirements for the DTM are addressed in Section 7.2.8 of this report. The RPS DTM is designed to be fail-safe. A loss of power to the DTM will result in a trip signal to each of the TLUs until that division is placed in bypass by the operator.

The DTM will receive input from both the EMS and directly hardwired sources. The hardwired sources include both the traditional hardwired inputs (discussed previously in this section) and the process PRM which provides the MSL high radiation input. The only RPS scram signals not processed through the DTM are from the manual operator controls and the NMS.

The DTM will send a trip or no-trip signal for each of the sensed parameters to each of the four TLUs. The signal to the TLUs which are not in the same electrical division will be electrically isolated and separated. The DTM will also provide data to the Class 1E divisional display units, the Class 1E mimic board, and isolated outputs to the plant computer and non-Class 1E alarms and annunciators.

GE described the following elements of the DTM:

1. Operating system/executive
2. Setpoint check
3. Trip decision transmission if setpoint exceeded
4. Serial data transmission to all divisions

For the items that are multiplexed, both multiplexor loops within a division and the identical sensor information contained on both is available to the DTM. As described in the previous section on the EMS design, the DTM will perform a data validity check and select from which CMU it will receive information.

As part of the data checking, the DTM will also verify that it is receiving the correct parameter and will ignore any other signals, such as those for the ESF DTMs. The DTM will also perform a data validity check on the non-EMS inputs and provide alarms if the checks fail.

Figure 7.A.2-1 in SSAR Appendix 7A shows all the sensor signals sent via the EMS. This drawing was listed in the DFSER as needing revision to be consistent with the distribution of the sensor inputs to the DTM as described above. This was DFSER Confirmatory Item 7.2.2.2-1. The SSAR has been revised to properly show the DTM sensor inputs, and this item is, therefore, resolved.

MPL A32-4080 (Rev. 0, April 25, 1990), "EMS/SSLC Interface Requirements, Requirements Specification," requires that the DTM permit manual trip of individual plant data inputs when an inoperable condition is detected in the incoming EMS data. This is consistent with the SRP criteria for a manual trip.

MPL C71-4010 (Rev. 0, May 18, 1990), "Reactor Protection System, Hardware/Software System Specification," lists the 11 sensor inputs and 34 outputs to the TLUs (if seismic inputs/outputs are not used) for each DTM. In addition to the sensor tripped/not-tripped status, the DTM output will include the mode switch MSL isolation trip bypass permissive to the TLU in its own channel. The DTMs are essentially identical between the four RPS divisions. There will be sensor identification and small timing differences that provide for some small differences, however, most of the device's hardware and software is identical between divisions. The DTMs are

Instrumentation and Controls

also similar to the DTMs of the ESF functions. The software algorithms will be significantly different between the RPS/MSIV and ESF functions but the operating system, the self diagnostics, and the input/output will be identical. The staff was concerned that the DTMs could be vulnerable to a potential common-mode failure of all divisions and between the RPS/MSIV and ESF due to a hardware or software design error. This was an open issue in the DFSER and the resolution is discussed in Section 7.2.8 of this report.

7.2.2.3 Trip Logic Unit

The trip logic unit (TLU) is a microprocessor-based unit which performs a two-out-of-four coincidence logic calculation based upon the signals it receives from the four DTMs and the four divisions of NMS. There is one TLU for each of the RPS logic channels and it does not share any functions with the ESF system with the exception of the ability to be bypassed by the operator. The TLU also receives the manual operator division scram inputs and the sensor channel bypass command. The primary TLU output is to the MSIV and RPS OLU's. The TLU also has outputs to the Class 1E displays and isolated outputs to the non-Class 1E displays and annunciators. The TLU is Class 1E.

MPL C71-4010 (Rev. 0, May 18, 1990), "Reactor Protection System, Hardware/Software System Specification," lists the approximately 106 interfaces (inputs and outputs) for each TLU.

MPL A32-4080 (Rev. 0, April 25, 1990), "EMS/SSLC Interface Requirements, Requirements Specification," requires that the TLU include a watchdog timer function on all outputs to ensure constant updating. Time-out shall cause outputs to assume predetermined safe states. These predetermined states were not stated in this MPL and, therefore, the DFSER stated that information on this matter is to be included in the ITAAC verifications. This was DFSER Open Item 7.2.2.3-1. In response to this issue, GE stated that tests have been added to the ITAAC to demonstrate that the failure state is fail-safe for the RPS. In addition, the MPL for the SSLC design specification provides the system failure modes and specifies that the RPS is fail-safe. Therefore, this open item is resolved.

GE identified the operating system/executive and the functional software as the two elements of the TLU software. The TLU configuration is characterized by the nine I/O interface boards connected to the central bus. There are a total of 21 input channels and nine output channels physically connected to each TLU. The details of the configuration are as follows:

a. There are four fiber optic communications interface boards that handle the following data channels:

1. bypass control inputs - 4 channels
2. DTM inputs (other divisions) - 3 channels
3. NMS inputs (other divisions) - 3 channels
4. recirculating pump trip output - 1 channel
5. data output to plant computer - 1 channel

b. There are two signal interface boards that handle the following data channels:

1. DTM input (own division)
2. NMS input (own division)
3. operator control input
4. TLU control output (adjacent division)

c. There are two logic level interface boards that handle the following data channels:

1. MSIV input - 4 channels
2. auto reactor trip output - 1 channel
3. auto MSIV closure output - 1 channel
4. MSIV test close output - 4 channels

d. There is one contact input interface board that handles the following data channels from the SSLC panel switches:

1. non-coincidental trip disable input
2. main condenser vacuum bypass input
3. auto trip test input
4. auto isolation test input

The TLUs are essentially identical between the four RPS divisions. There will be identification and small timing differences in some TLUs, however, most of the devices are identical. The TLUs are also similar to the SLUs of the ESF functions. The TLU software algorithms will be significantly different but the operating system, the self diagnostics, and the input/output will be identical. The staff was concerned that the TLUs could be vulnerable to a potential common-mode failure similar to the DTMs. This was an open issue in the DFSER and the resolution is discussed in Section 7.2.8 of this report.

7.2.2.4 Output Logic Unit

MPL C71-4010 (Rev. 0, May 18, 1990), "Reactor Protection System, Hardware/Software System Specification," lists the 6 inputs and 12 outputs for each OLU. The OLU is a Class 1E solid state electronics device (non-microprocessor-based) which receives signals from the TLU within its own electrical division. There are

separate units for the MSIV and RPS initiation functions. The OLU also receives the division bypass commands, the manual division trip input for the RPS OLU and manual division isolation input for the MSIV OLU. The output of the OLU units is to the MSIV and RPS load drivers which provide a division level two-out-of-four vote. This is in addition to the separate two-out-of-four sensor input vote which occurs in the TLU.

7.2.2.5 Self-Test System

The self-test system (STS) feature of the SSLC is described in Section 7.1.2.6 of the SSAR. The STS performs testing on both the RPS and ESF functions of the SSLC. The STS was originally presented in the SSAR as an overlay testing and surveillance system which continually and automatically performs end-to-end testing of all active circuitry in the SSLC using short test pulses similar to the Clinton nuclear system protection system design. GE has substantially revised the SSAR description since the DSER was issued to tailor the STS for a multiplexed microprocessor-based design.

GE originally classified the STS as safety-associated and stated that Class 1E equipment would be used wherever the STS interfaced with safety-related equipment. The staff was concerned that the test features were not classified as Class 1E and could potentially degrade the safety function. This issue was identified as an open item in the DSER (SECY-91-294). The staff was also concerned about the potential interaction between the STS and the master/slave configuration in the SSLC causing problems with the SSLC operation and possibly violating the separation requirements of the SRP. The design of the STS as described in the SSAR is now substantially integrated into the SSLC, and GE has committed that all STS equipment will be qualified as part of the SSLC (Class 1E). The revised description and Class 1E designation have also resolved the configuration and separation issues identified in the DSER. This is acceptable and the DSER open issue is, therefore, resolved.

The STS for safety-related systems and the SSLC are required to be Class 1E in accordance with the criteria of the SRP. This had not been explicitly stated in the SSAR and was identified as DSER Confirmatory Item 7.2.2.5-1. The SSAR has been revised to clarify that the Class 1E requirement will apply for all safety-related self-test or self-diagnostic features and, therefore, this item is resolved. The Class 1E requirement applies to self-test features imbedded in the software of safety-related systems. However, it is acceptable to use non-Class 1E testing equipment in typical configurations where a safety-related I&C channel or system under test is taken out of service to perform manual tests.

The protection system in-service testability requirement comprises a set of six separate tests, which together constitute a complete system test:

1. A manual scram test will de-energize one set of scram pilot valve solenoids at a time. This test will also verify the indications in the main control room and the plant computer input.
2. A calibration check of the NMS will verify calibration of setpoints.
3. The single rod scram test will include a physics review performed before insertion of each rod.
4. A calibration check of the analog sensor inputs at the inputs to the RMUs will verify, by injecting calibrated sensor signals in place of the normal sensor inputs and monitoring the SSLC control room panels, linearity, accuracy, fault response, and downscale and upscale trip responses. This test is accomplished by placing a division of sensors in the bypass position.
5. A check of sensor operation will verify sensor inputs by cross comparison of other channels, by varying the monitored variable, or by substituting a test source. If the test requires disconnecting the sensor from the system, an out-of-service alarm will be given in the main control room.
6. A self-diagnostics test of the SSLC equipment will be run to detect and determine the location of a failure in the functional SSLC system. The self-test provision within each division of the SSLC system consists of an on-line, continuously operating, self-diagnostic monitoring network, and an off-line semi-automatic, end-to-end surveillance program. The self-diagnostic software within each logic processing unit will monitor critical circuit nodes and timing functions, as well as states of registers, memory locations, and program flow and timing. These monitoring functions are designed to detect both internal problems and certain external problems such as corrupted data input. The self-diagnostics will include hardware techniques such as watchdog timers.

The capability to perform on-line protection system testing is included in the CDM. The specific implementation is described in the SSAR. Specifics of the system design features to permit testing are not available. The ITAAC process will verify that the above tests can be accomplished and includes use of the test equipment indicated for the above tests itself to demonstrate some of the other I&C functional requirements in the ITAAC.

Instrumentation and Controls

One significant feature of the testing that has been described by GE is the elimination of the need to lift leads and install jumpers to perform testing. Such actions have been a significant problem area at currently operating plants. The DFSER stated that this feature should be included as a CDM requirement, which was identified as DFSER Confirmatory Item 7.2.2.5-2. The SSAR has been revised to include the ABWR design testing features that eliminate jumpers and lifting leads, and the TS have been prepared based on these design features. The CDM has been revised to reflect the technician interfaces for testing, and the staff finds it to be acceptable. This confirmatory item is resolved.

Any changes to the design testing features and commitments identified in the SSAR would involve an unreviewed safety question and, therefore, require NRC review and acceptance prior to implementation. Any requested changes to these commitments shall either be specifically described in the COL application or submitted for license amendment after COL issuance.

7.2.3 Indication of Bypassed and Inoperable Status

MPL C71-5030 (Rev. 0, April 23, 1990), "Reactor Protection System Verification and Validation Criteria Design Specification," requires that only one sensor channel be bypassed at a time. However, the ABWR design does not include the capability to bypass an individual sensor. The sensor bypass referred to in this MPL is the same as that which the staff and GE have defined as a division of sensors bypass. The ITAAC in CDM 3.4, Instrumentation and Control, includes a verification that only one division of sensors may be bypassed at a time.

MPL C71-4010 (Rev. 1, July 2, 1990), "Reactor Protection System Design Specification," states that one sensor channel may be bypassed and the coincidence logic will go to a two-out-of-three vote at the TLU. This bypass is implemented at the input to the TLU. Bypass of one division of output logic at the OLU will result in a two-out-of-three coincidence logic. This bypass is implemented at the input to the OLU. The above MPL document also requires that bypass status be readily apparent and under direct control of the operator.

When any part of the RPS or supporting systems is bypassed or made inoperable, a continuously displayed status of this condition is required in the main control room. SSAR Section 7.2.2.2.1 states that this requirement is met with individual indicator lights grouped near the affected equipment. The following alarms and annunciators are provided for the RPS bypass and inoperable status and are qualified as Class 1E:

- RPV level 3 scram bypassed
- RPV pressure high scram bypassed
- Drywell pressure high scram bypassed
- Neutron flux high scram bypassed
- MSIV closure scram bypassed
- CRD charging water pressure low scram bypassed
- MSL radiation high scram bypassed

The following alarm is provided for the RPS inoperable function as non-Class 1E:

Indicated RPV water level invalid

There are also additional commitments for all bypasses to be annunciated, however, the alarm and annunciator list from which this information was extracted did not list all of the non-Class 1E alarms and annunciators but only those specifically identified in the Emergency Planning Guidelines (EPGs). To address this issue, the staff reviewed the ABWR SSAR for commitment to RG 1.47, "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems," Position C.2, which requires that the bypassed and inoperable status of RPS auxiliary or supporting systems be automatically indicated in the main control room. The description in the SSAR indicates that adequate indication and annunciation for these systems will be provided in the main control room. The indication is automatic at the system level when the system loses power or when it is out of service. A switch will be provided for manual initiation of bypass indication for out-of-service conditions which could not be automatically annunciated. The staff concludes that GE has provided adequate commitment to RG 1.47. The DFSER listed the verification that all bypasses are appropriately annunciated as DFSER Confirmatory Item 7.2.3-1. Bypass testing has been included in the CDM, and this item is, therefore, resolved.

7.2.4 Alarms and Annunciators

For the ABWR, the most significant change in alarm and annunciator design from previous designs is that many of the alarms are now Class 1E and as such will be powered from the divisional Class 1E power sources. These Class 1E alarms are distributed between the fixed display (fixed alarm tiles above the mimic section) and the VDU. As a part of the ABWR design, the RPS channel trips will annunciate and identify the variable which caused the trip. Loss of annunciator events at currently operating plants have demonstrated that this is important information for the operator. SSAR Chapter 18 and the corresponding section of this report for the human factors review provide a discussion and evaluation of the displays.

The ABWR has committed to using non-Class 1E "uninterruptable power supplies" for the non-Class 1E annunciators. Details on the distribution of the non-Class 1E power loads were not provided by GE in the SSAR. However, independence of Class 1E and non-Class 1E power supplies and isolation of non-Class 1E alarms from Class 1E alarms will ensure that the Class 1E alarms would be available in the event of loss of the uninterruptable power supplies.

At the staff's request, GE provided an inventory of control room instrumentation, including a list of Class 1E alarms, based upon the EPGs. Because the inventory is emergency mitigation based, it does not include a list of the non-Class 1E alarms that are not required for entry conditions into the EPGs. Additional discussion on the inventory is provided in Section 18.0 of this report.

The number of fixed position alarm tiles in the ABWR control room has been significantly reduced compared to current operating plants by employing alarm prioritization and filtering. In the ABWR, the fixed alarm tiles are only used for important plant-level alarm conditions that potentially could affect plant availability and plant safety, or indicate the need for immediate operator action. Less critical alarms are presented on the operator console VDUs. In addition, the large control room overview display panel will include important safety-related (Class 1E) system-level alarms and some non-Class 1E system-level alarms.

The following is a list of Class 1E alarms for the ABWR. The fixed displays will be primarily alarm tiles near the large overhead mimic. The term "VDU" refers to the CRTs or plasma screens on the control room consoles. (There are several VDUs, and this list is not intended to imply that all the alarms are on one screen.)

The Class 1E alarms that indicate RPS information include:

- (1) RPV water level (fixed)
- (2) drywell pressure high (fixed)
- (3) drywell pressure high (VDU)
- (4) RPV water level 3 (fixed)
- (5) RPV water level 3 (VDU)
- (6) RPV pressure high (fixed)
- (7) RPV pressure high (VDU)
- (8) neutron flux high (fixed)
- (9) neutron flux high (VDU)
- (10) MSIV closure (fixed)
- (11) MSIV closure (VDU)
- (12) CRD charging water pressure low (fixed)
- (13) CRD charging pressure low (VDU)
- (14) main turbine stop valve closure (VDU)
- (15) main turbine control valve fast closure (VDU)
- (16) MSL radiation high (fixed)
- (17) MSL radiation high (VDU)
- (18) RPS Div I trip (VDU)
- (19) RPS Div II trip (VDU)
- (20) RPS Div III trip (VDU)
- (21) RPS Div IV trip (VDU)
- (22) RPS Div I manual trip (VDU)
- (23) RPS Div II manual trip (VDU)
- (24) RPS Div III manual trip (VDU)
- (25) RPS Div IV manual trip (VDU)
- (26) manual scram (A) initiated (VDU)
- (27) manual scram (B) initiated (VDU)
- (28) reactor scram (fixed)
- (29) reactor period short (fixed)
- (30) SRNM neutron flux upscale rod block (VDU)
- (31) SRNM neutron flux upscale reactor trip (VDU)

The non-Class 1E alarms that annunciate RPS information include:

- (1) drywell pressure high (fixed)
- (2) RPV pressure high (fixed)

In addition to the above displays of alarm conditions, access to the plant computer will allow the operator in the control room to identify the specific sensor or sensors that caused a channel trip. The plant computer will also maintain the sequence of events log. This log will contain both nuclear steam supply system and balance of plant inputs. The plant computer is not part of the safety systems and no credit is taken for this system in meeting SRP acceptance criteria. The plant computer is described in Section 7.7.1.5 of this report.

The ABWR design provides for the distribution of alarms between the fixed displays, the VDUs, and the large overview panel; the independence of divisionalized Class 1E and uninterruptible non-Class 1E power supplies that provide power to the alarm systems; and the isolation of non-Class 1E alarms from the Class 1E alarms. These provisions in the design minimizes the potential for total loss of annunciators due to single failures as experienced by operating nuclear plants and meet the requirements of redundancy, independence, and separation of alarm systems of the Commission's position on control room annunciator reliability in SECY-93-087.

7.2.5 Support Systems

The support systems discussed in this section include the line power supplies and the HVAC systems.

SSAR Chapter 8 and the corresponding section of this report discuss the plant station power. For the SSLC, the

Instrumentation and Controls

power is supplied by four independent and separated Class 1E 120 Vac sources, each of which is backed up with a Class 1E 125 Vdc battery source through an inverter. Divisions I and III of the 120 Vac sources are each supplied power from the corresponding division 480 Vac power source; 120 Vac Divisions II and IV are supplied from the Division III 480 Vac source as indicated by SSAR Figure 8.3-3. SSAR Figure 7.2-1 initially incorrectly showed a fourth division of 480 Vac power and, therefore, needed to be revised to reflect the power supply design. This was DFSER Confirmatory Item 7.2.5-1. SSAR Figure 7.2-1 was subsequently corrected and this item is, therefore, resolved.

MPL C71-4010 (Rev. 1, July 2, 1990), "Reactor Protection System Design Specification," is consistent with the SSAR and the CDM in that it calls for four Class 1E 120 Vac systems. However, the specifications in the MPL identified only two 125 Vdc systems whereas the ABWR CDM and SSAR stated that there are four 125 Vdc systems provided. GE indicated in the SSAR that the four 125 Vdc system configuration is the one for the ABWR design, and this is acceptable to the staff.

The scram pilot valve solenoids are powered from the Divisions II and III 120 Vac SSLC buses.

A potential Division II 6.9 KV/480 Vac transformer or 480 Vac switchgear failure could disable the 480 Vac power sources to the inverters associated with Divisions II and IV 120 Vac SSLC buses. Such a failure could also affect the inverters and result in disabling the capability to use the backup 125 Vdc sources. The failure of the inverters would lead to degradation or loss of power to the two 120 Vac SSLC buses. This issue was addressed in two ways. The first is by incorporating a TS requirement that limits the bypass of SSLC channels. This was an open item in Chapter 16 (Item 16-2) of the DFSER and the resolution is discussed further in Section 7.11 of this report. The second way is by providing electrical protection assemblies (EPAs) for the SSLC buses in the ABWR electrical distribution system design. EPAs provide an additional level of protection to the SSLC power supplies. Each assembly consists of a circuit breaker with a trip coil driven by logic circuitry which senses line voltage and frequency, and trips the circuit breaker open on a condition of overvoltage, undervoltage, or underfrequency. The EPAs will detect a spectrum of degraded conditions of the SSLC bus power supply and open the power supply line in time to prevent serious damage to connected equipment. Another area of benefit by using EPAs is reduction in the possibility of the scram pilot solenoid valves sticking as a result of damage caused by insufficient voltage supply to the solenoid coils. The ABWR design requires the coils of both scram pilot solenoid valves of each CRD to disengage

when power is removed, to initiate a scram. The EPAs will detect an undervoltage condition and remove power prior to possible damage to the coils. Normal voltage drop between the location of the EPAs and the solenoid coils will be included in the design considerations. The ITAAC for the EPAs will include a verification that the wiring to the solenoid valves is sized so that the voltage drop in the cables will not result in insufficient voltage being supplied to the solenoid coils. The ITAAC will also include a verification that the neutral leads of the scram pilot solenoid valve coil windings are configured such that credible faults (e.g., hot shorts) will not prevent the valves from performing their safety function. The above ITAAC considerations were identified as DFSER Confirmatory Item 7.2.5-2. GE has included this information in the SSAR and the staff finds it to be acceptable. Therefore, this item is resolved.

The four primary SSLC cabinets (or set of cabinets) are installed in the main control room. Portions of the RPS and SSLC (in particular, the RMUs) are located in other mild environment areas such as the control building equipment rooms. The RPS I&C equipment which is designed for harsh environments and will be required to meet the electric equipment environmental qualification requirements of 10 CFR Part 50.49 are certain sensors, sensor lines, transmitters, and associated cabling. None of the microprocessor-based equipment, multiplexor units, or fiber optic links will be in areas designated as a harsh environment area. All of the SSLC equipment is qualified to the environment of the room in which it is located. The support system is the HVAC system for the specific area. Section 7.2.8 of this report describes the hardware and software qualification.

7.2.6 Defense-in-Depth Analysis

The I&C systems for the ABWR help ensure that the plant operates safely and reliably by monitoring, controlling, and protecting critical plant equipment and processes. Both safety and non-safety I&C systems for the ABWR are primarily digital-based and differ significantly from the primarily analog systems used in currently licensed operating plants. The digital I&C system shares more data transmission functions and process equipment than was the practice with the analog systems. The ABWR I&C systems use the same software and processing equipment (hardware) across the safety divisions, and therefore, a hardware design error, a software design error, a software programming error, or a maintenance error may affect all of the I&C system divisions and result in a common-mode or common-cause failure of redundant equipment. The staff was concerned that the use of digital computer technology in I&C systems could result in safety-significant common-mode failures. Because of these

concerns, which were expressed early in the preliminary reviews of the ABWR, the staff requested that GE prepare an analysis of ABWR I&C system defense-in-depth based upon NUREG-0493, "A Defense-in-Depth and Diversity Assessment of the RESAR-414 Integrated Protection System" (1979). The staff believes that a level of diversity is necessary to provide defense-in-depth against potential common-mode software errors so that necessary safety functions can be performed reliably. The staff considers common-mode software errors to be a special case of single failure and, therefore, protection against such errors is to be part of the design bases.

To complicate the concern with regard to common-mode failure, the initial SSAR did not include the design details that would allow the staff to independently assess the diversity and defense-in-depth of the design and conformance with the guidelines of NUREG-0493. Therefore, the staff concluded that the common-mode failure potential of software had not been adequately addressed, and this was identified as an open item in the DSER. Subsequently, GE provided the results of analyses done to address the defense-in-depth issue as amendments to the SSAR. Resolution of this item is discussed further in this section of the report.

BACKGROUND

The first design reviewed by the staff specifically to address defense against potential common-mode failures in digital systems was the Westinghouse RESAR-414 design. The results of this study were published in NUREG-0493, "A Defense-in-Depth and Diversity Assessment of the RESAR-414 Integrated Protection System," (March 1979). NUREG-0493 discussed common-mode failures and different types of diversity, and presented a method for assessing the defense-in-depth of the design.

The staff described concerns with common-mode failures and other digital system design issues in SECY-91-292, "Digital Computer Systems for Advanced Light Water Reactors." SECY-91-292 describes how common-mode failures could defeat not only the redundancy achieved by the hardware architectural structure, but also could result in the loss of more than one echelon of defense-in-depth provided by the monitoring, control, reactor protection, and engineered safety functions performed by the digital I&C systems. The two principle factors for defense against common-mode/common-cause failures are quality and diversity. Maintaining high quality will increase the reliability of both individual components and complete systems. Diversity in assigned functions for both equipment and human activities, equipment, hardware and software, can reduce the probability that a common-mode failure will propagate.

The modules in the ABWR SSLC are to be implemented by microprocessor-based designs with identical or similar hardware and software used in all four divisions. Because of this similarity the concerns expressed in NUREG-0493 and SECY-91-292 apply directly to the SSLC.

The staff reviewed the ABWR I&C system design and concludes that common-mode failure concerns were valid for several reasons.

1. The commonality of the timing between channels is such that an error in one channel is expected to occur in all identical channels and equipment within a few milliseconds of each other.
2. The possibility exists that an initiating transient or accident itself creates the set of circumstances that reveal a software error. The staff considers that the models of the systems used to develop the test sets may not contain sufficient inputs for all transient and accident situations, and therefore, certain situations may not be adequately tested. In most software applications, including the ABWR design, it is not possible to include a 100-percent test of all software inputs due to the very large number of possible input combinations.
3. There is presently no general consensus as to a method to demonstrate a quantitative measurement of software reliability. Some assumptions, such as the GE ABWR PRA assumption of 4.25×10^{-8} failures per demand for the EMS, cannot be demonstrated with the current software metrics or existing data bases, and are likely unrealistically low.
4. Redundancy in software does not necessarily increase the reliability and availability of the overall system to an acceptable level as it can in analog systems. Some of the failure modes of a software-based system (particularly the common-mode failures mentioned above) are fundamentally different from those of an analog system.
5. Self-diagnostics and periodic testing provide a significant safety improvement by reducing the possibility of undetected failures during plant operation, but they do not prevent the failures from occurring. The improved self-diagnostics of digital systems do not resolve the common-mode failure issue. The analysis described subsequently in this report explains the common-mode vulnerabilities of the ABWR design, and the attributes which mitigate those failures.

Instrumentation and Controls

Several NRC regulations and industry standards address the need for defense against potential common-mode failures. GDC 22 requires that "design techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protection function." IEEE Standard 279-1971 requires that "equipment, not subject to failure caused by the same credible event, shall be provided to detect the event" IEEE Standard 603-1980 has the same requirement as IEEE-279. IEEE Standard 379-1968 states that "certain common-cause failures shall be treated as single failures when conducting the single failure analysis. Such failures can be in dissimilar components and can have dissimilar failure modes. Failures resulting from cascaded failures and from design bases events have already been discussed and are those which shall be included in the analysis. Common-cause failures not subject to single-failure analysis include those that can result from external environmental effects, design deficiencies, manufacturing errors, and operator errors. Design qualification and quality assurance programs are intended to afford protection from external environmental effects, design deficiencies, and manufacturing errors. Personnel training, proper control room design, and operating and maintenance procedures are intended to afford protection from maintenance and operator errors." Common-mode failure issues are also addressed in the requirements of 10 CFR 50.62 concerning mitigation of anticipated transients without scram (ATWS).

There are several different types of diversity, each of which offers certain protection against the common-mode failures. Various forms of diversity include signal diversity, equipment diversity, aspect diversity, and people diversity. Signal diversity includes the use of different signals (sensors) to initiate an action, such as neutron flux and reactor pressure as diverse signals for initiation of reactor scram. Equipment diversity includes using different kinds of equipment to perform a function. An example of equipment diversity described in NUREG-0493 is the use of relay vs solid-state logic in the I&C system. The ABWR design employs equipment diversity in that the remote shutdown station (RSS) is a hardwired analog system and is, therefore, diverse from the microprocessor-based SSLC. Included in the equipment diversity category is the use of different software languages. Aspect diversity involves using different logic levels. An example for the ABWR is the functional algorithm diversity between the DTM functions within a channel. Another example of aspect diversity in the ABWR design is the de-energize to actuate (fail-safe) aspect for the RPS actuation vs. the energize to actuate (fail as-is) aspect of the ARI. People diversity refers to using different groups of people to design, verify, validate, and maintain an I&C system. Examples of people diversity required in the ABWR design

are the different groups performing the I&C system design vs. those performing the QA function, and the independence between the software verifier and the software designer. It is difficult to define how much improvement in safety results from a given kind or degree of diversity. For microprocessor design, this is especially difficult because there is no industry consensus on a method to quantify software reliability and/or availability.

DISCUSSION OF THE RESOLUTION

GE's initial analysis of common-mode failure was provided in Appendix 7A of the SSAR. In the SSAR, GE stated that the use of shared sensors in the design may increase the effects of potential common-mode failures. Therefore, the SSLC system architecture is designed to provide maximum segregation of system functions by using separate DTMs and TLUs within each of the four I&C divisions. In the analysis it was also noted that for the reactor shutdown function there are five different methods for controlling reactivity including the RPS hydraulic scram, the air header dump valves of the ARI system, the fine motion control rod drive (FMCRD) insert function of the ARI system, the SLCS, and the CRD system. The reactor core cooling function can be accomplished as described in the SSAR by four different systems including the motor driven (FDWC) system, the motor driven high-pressure core flooders system (HPCF), the turbine driven RCIC system, and the low-pressure mode of the RHR system. Appendix 7A also described the RSS which provides diverse (hardwired) core cooling control functions. GE concluded that the ABWR meets the intent of NUREG-0493 for I&C system diversity. The staff, however, concluded in the DSER that the GE analysis did not adequately address potential loss of safety functions due to postulated common-mode failures, and that this was an open issue.

The staff considers the two principle factors for defense against common-mode failures to be quality and diversity. The quality aspects of the ABWR I&C systems are addressed in other sections of this report. Quality, in part, is achieved by the use of quality design standards for the hardware and software, and the I&C system testing to be performed. With few exceptions, the staff concluded in the DSER that the quality issues had been substantially resolved.

Though there were some studies performed by GE in the design process, no analyses were presented to the staff which adequately demonstrated how the SSLC design complied with NUREG-0493, and thereby provided adequate defense against potential common-mode failures. The staff concluded in the DSER that this concern had

not yet been addressed adequately by GE. The staff identified the defense-in-depth issue as DFSE Open Items 7.2.6-1, 7.2.6-2, and 7.2.6-3.

Because the staff determined that the ABWR SSAR and related documentation had not adequately addressed the common-mode failure and defense-in-depth concern, the staff performed its own common-mode failure assessment of the ABWR based upon the guidance of NUREG-0493. Some additional considerations were added to the original NUREG-0493 approach, such as evaluation of information available to the operator, common-mode failures during accidents as well as transients, the time available for systems actuation, and the use of non-Class 1E systems to provide a diverse means of accomplishing safety functions if the first-line safety systems failed. The staff's defense-in-depth and diversity assessment of the ABWR design was performed by the Lawrence Livermore National Laboratory (LLNL) (under contract to the staff - referred to as the LLNL diversity study). The results were made available to GE for determination of any factual errors in the design assessment. GE informed the staff of some errors in the study which have been corrected.

The LLNL diversity study evaluated I&C system function for all of the SSAR Chapter 15 events. This set of events was judged by the staff to be sufficiently complete to bound those situations requiring initiation of safety systems based on the NUREG-0493 methodology. The study evaluated each event in conjunction with a set of postulated common-mode failures. Two specific events were selected for detailed study in the preliminary stages of the review. Those events were generator load rejection with normal bypass, and steam system piping break outside containment. Assumptions made by LLNL in the analysis during the review were documented in the LLNL diversity study. One aspect that was not specifically evaluated in the study was anticipated operator actions. As described later in this report, operator actions have now been considered.

In the LLNL diversity study several areas of concern were identified. The use of the EMS for the RPS, ESF and information to the operator was a particular vulnerability, along with the common elements shared by the DTMs and TLUs. In the study it was concluded that, in general, there was information and system controls to mitigate each of the transients investigated, however, there may not be sufficient time and information to complete all necessary activities manually to maintain safety functions, especially if the actions required the use of the RSS.

Due to the significance of the EMS to the proper functioning of both the RPS and ESF, a common-mode failure of the EMS was a significant concern of the staff in

the resolution of the defense-in-depth and diversity issue. The modules identified in the EMS are to be implemented by microprocessor based designs with similar hardware in all four divisions. MPL A11-4121 (Rev. a (preliminary), January 19, 1988), "Multiplexing System Application Procedure, Design Procedure," specifically states that the individual multiplexing systems shall be designed with a high degree of standardization. The DTM and TLU/SLU components of the SSLC also share common software design features which would result in a failure of all four channels if there is a software error.

In reviewing the results of the study, the staff found diverse I&C system features in the ABWR design, several of which GE had previously presented as solutions to the potential common-mode software error concern. In the study, credit was given to non-Class 1E systems to mitigate an event if the non-safety systems were reasonably expected to be available. Thus, from the study the staff determined that the ABWR design has a number of attributes which provide defense-in-depth and protection against a potential software error as follows:

1. The turbine inputs to the RPS are hardwired (do not use the EMS). Further, the NMS and the PRM system RPS inputs use microprocessors but are directly wired to the SSLC and do not use the EMS. Therefore, an EMS common-mode failure would not disable these inputs.
2. Manual scram functions and manual MSIV actuation are hardwired from the control room, do not use the EMS, and are not dependent upon microprocessors.
3. The ARI function, that is part of the ATWS system, is independent of the EMS. The ARI system is a two-out-of-three logic initiation non-Class 1E system that is separate and independent from the RPS. By letter dated June 2, 1993 (on important features identified by the ABWR PRA), GE identified this feature as providing a significant factor in the reduction in the probability of an ATWS.
4. The NEMS is diverse in both hardware and software from the EMS. The NEMS is a non-Class 1E system.
5. The RSS is conventionally hardwired from the station itself to the actuation devices (does not use the EMS) and does not use microprocessors. The RSS is outside of the main control room as required for its primary design function of shutting down the reactor upon abandoning the main control room. The RSS is a two channel control and indication station which contains most of the ESF capabilities.

Instrumentation and Controls

6. Final display information for the operator is provided by a separate, diverse means -- the Class 1E fixed mimic display and the Class 1E divisional VDUs. Alarms and parameter information are also available on the non-Class 1E VDUs and from the plant computer.
7. The NMS bypasses the DTM and is input directly to the TLU. A common DTM failure will not fail the NMS scram function. The ARI system controls also bypass the DTM because the ARI system is not part of the SSLC.
8. The DTMs and TLU/SLUs have a significant level of functional diversity between the RPS and ESF functions and between portions of the ESF. As shown on Figures 7.3-1 and 7.3-2 of this report, there are three DTMs, one TLU and two SLUs per channel of the SSLC. The software algorithms within the SSLC are functionally diverse between the RPS and ESF functions as a result of the equipment to be tripped/actuated, i.e., deenergize-to-operate function for RPS and MSIV, and energize-to-operate function for ESF.
9. GE stated in the SSAR that the SSLC software will be relatively simple which will result in a high degree of assurance that the required testing will reveal virtually all of the software errors. The EMS software is even simpler. However, no supporting analysis has been provided to support these conclusions. All the safety-related software will be verified and validated. The staff concludes that software, even that which has been verified and validated with a high-quality program may still have undetected errors. This was DFSE Open Item 7.2.6-4. In a letter dated April 30, 1993, GE provided a description of the software development program, a commitment to provide a set of hardwired backups, and a revised common-mode failure analysis. In the letter, GE concluded that a software simplicity analysis was no longer relevant since the quality and diversity issues for software will address the staff's concerns regarding software reliability and digital system defense-in-depth. The staff agrees that this analysis is no longer necessary as the basis for resolving the defense-in-depth questions. These concerns are addressed in the computer ITAAC and overall digital system diversity, and, therefore, Open Item 7.2.6-4 is resolved.

Based on the two examples reviewed in detail in the LLNL diversity study, the staff and GE concluded that some postulated failures in the SSLC (and EMS) would disable the displays and controls of the ESF systems in the main control room. Some postulated failures would also result in the loss of a significant amount of information to the

operator in the control room. Another conclusion was that it would be necessary to access the RSS in order to initiate the ESF equipment and mitigate the events under consideration. For the events reviewed, there was display and mitigation capability available for all postulated failures. The mitigation control function was occasionally at the RSS or was provided by a non-Class 1E system. Normally available control systems, such as the FDWC system, were credited in the analysis, while systems not normally used, such as the fire water system for reactor water injection, were not.

The results of the LLNL diversity study were presented to GE. The staff identified the following four primary concerns arising out of the study:

1. The response to the initiating event needed to be confined to the main control room.
2. Consideration of the time available for manual operator actions was incomplete in the analysis.
3. System level actuation capability from the control room for the ESF functions was lacking.
4. Display of the necessary Class 1E variables in the main control room was lacking.

The staff requested GE to complete its review of the LLNL diversity study and respond to the above concerns. The above results were also provided to the Commission for approval in SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs," April 2, 1993, as the staff's generic position for digital system defense against common-mode failures for advanced light-water reactors.

The staff position on I&C system diversity for ALWRs as approved by the Commission in a SRM dated July 21, 1993, is as follows:

1. The applicant shall assess the defense-in-depth and diversity of the proposed I&C system to demonstrate that vulnerabilities to common-mode failures have been adequately addressed.
2. In performing the assessment, the vendor or applicant shall analyze each postulated common-mode failure for each event that is evaluated in the accident analysis section of the safety analysis report (SAR) using best-estimate methods. The vendor or applicant shall demonstrate adequate diversity within the design for each of these events.

Figure 7.3-1 Safety system logic and control communications path for engineered safety systems

Figure 7.3-2 Assignment of interfacing safety system logic to safety system logic and control controllers

3. If a postulated common-mode failure could disable a safety function, then a diverse means, with a documented basis that the diverse means is unlikely to be subject to the same common-mode failure, shall be required to perform either the same function or a different function that provides adequate protection. The diverse or different function may be performed by a non-safety system if the system is of sufficient quality to perform the necessary function under the associated event conditions.
4. A set of displays and controls located in the main control room shall be provided for manual system-level actuation of critical safety functions and monitoring of parameters that support the safety functions. The displays and controls shall be independent and diverse from the safety computer system identified in Items 1 and 3 above.

The staff's proposed applicable regulation for digital system defense against common-mode failures for the ABWR design is, as follows:

Digital instrumentation and control systems provided for the standard design must include:

- (1) an assessment of the defense-in-depth and diversity of instrumentation and control systems,
- (2) a demonstration of adequate defense against common-mode failures, and
- (3) provision for independent backup manual controls and displays for critical safety functions in the control room.

GE completed its review using events that GE believes envelope the Chapter 15 events. GE disagreed with the staff's position and indicated that adequate defense-in-depth and diversity should consider the likelihood of the postulated events in conjunction with the assumed common-mode failures. Under these circumstances, credit for the RSS to mitigate the event should be permitted.

In response to the GE position, the staff prepared a list of the set of equipment which it believed was necessary to bring the ABWR design into compliance with the proposed staff position and which were necessary for a demonstration of appropriate diversity. The list was based on a functional, symptom-based approach to accident mitigation to assure that adequate reactivity control, core cooling, reactor coolant system integrity, and primary containment integrity are maintained for all events. This list was also based upon a review of the control functions

identified in the EPGs and the post-accident monitoring indications identified in RG 1.97.

GE completed three studies related to furthering the issue of defense against common-mode/common-cause failures in the ABWR I&C systems. The first was an analysis of the common-mode failure of ABWR multiplex equipment (SSAR Appendix 19N, Amendment 22). This study identified the following potential common-mode failure mechanisms:

- (1) Earthquake
- (2) Loss of dc power
- (3) Loss of cooling
- (4) Sensor miscalibration
- (5) RMU miscalibration
- (6) Set point drift
- (7) Maintenance/test error
- (8) Manufacturing error
- (9) EMI
- (10) Fire
- (11) Software fault

In the study, GE addressed each of the above issues. Most were evaluated as not being credible causes due to the qualification of the I&C equipment, physical separation, or administrative controls. The study contained the conclusion that common-mode software fault is a credible, although unlikely, possibility. GE committed to administrative controls to minimize errors, TS requirements to assure failure detection, and symptom-based procedures to assure that adequate core cooling is maintained in the event of a common-mode EMS failure.

The second GE study relating to the issue of defense against common-mode failures was the preparation of an inventory of controls, displays, and alarms relied upon for accident mitigation based on the EPGs, including their locations. This information was submitted as SSAR Appendix 18F. Based on the information provided in the SSAR, GE demonstrated that the displays will have separation and some diversity. The diversity for the safety-related displays is primarily provided between the fixed mimic panel and the safety channel VDUs.

The third study prepared by GE was an event-based common-mode failure evaluation. (The result of the study was presented to the staff at a meeting held at the GE offices in San Jose, California on August 26, 1992, and was documented as part of the minutes of that meeting). In performing the study, GE evaluated 14 events from the SSAR Chapter 15 transients and accidents, considering the emergency operating procedure entry conditions and a postulated common-mode failure in the ABWR I&C system. The study addressed the automatic actions that

Instrumentation and Controls

would occur following each event coincident with the postulated common-mode failure. The common-mode failure postulated for each of the events was an undiscovered fault in the EMS resulting in all valid and correct control and monitoring data transmissions being lost. The emergency operating procedure (EOP) entry conditions were then considered. In several events, more than one monitored parameter is expected to indicate the need to enter the EOPs. Indication of reactor pressure vessel water level low was the entry condition for many of the events; drywell pressure high indication was the entry condition for several others. The study considered operator actions described in the EOPs, with the equipment which was not disabled by the postulated common-mode failure. The reactor and containment response was evaluated for each event and was compared to the SSAR Chapter 15 analysis (which does not assume common-mode failures of I&C equipment). A summary of each event was provided which concluded that, for one hour or more, sufficient water was available for decay heat removal.

A significant factor considered in this analysis was the function of the feedwater system during the postulated transient or accident. For each of the events (except for feedwater line breaks or failures) the feedwater system was assumed to remain operational. The feedwater system was also assumed to function properly in response to the specific event scenario. For example, a loss-of-coolant accident (LOCA) inside containment, a loss of condenser vacuum, or a loss of the auxiliary power transformer requires a reduction in feedwater flow to the reactor vessel. GE stated that reduced flow was a normal expected function of the feedwater system under the above event condition. If the feedwater system continues to run at 100 percent flow, the available water inventory would be exhausted in approximately 4 minutes. If the feedwater system responds as assumed in this study, water inventory would be available for over an hour. Based on this study, GE concluded that there was adequate capability in the ABWR design to mitigate each event for sufficient time until the RSS capabilities could be used.

The staff reviewed the GE study as documented in the SSAR and agreed with the conclusions. However, the staff believes that GE relied too heavily upon the feedwater system as a backup to safety-related RPV inventory means; therefore, additional backup capability should be provided in the main control room. This was discussed further in the DFSE.

CONCLUSION

The staff concludes in the DFSE that the following additions to the ABWR design are needed in order to

provide acceptable defense-in-depth against potential common-mode failures of the I&C systems:

1. The following control capability will be added in the control room:
 - a. clean up water line isolation valve (inboard) manual initiation. This is required by both Items 3 and 4 of the above staff diversity position.
 - b. RCIC steamline isolation valve (inboard) manual initiation. This item is required by both Items 3 and 4 of the staff position listed above.
 - c. HPCF system manual initiation. In the diversity analyses, many of the events rely on the feedwater system to mitigate the event in combination with the postulated common-mode failure. Experience with feedwater systems at operating plants raises a concern that they may not be reliable enough to satisfy the Item 3 requirement that the backup system be of sufficient quality to perform the necessary function under the associated event conditions. The ABWR feedwater system uses a triplicated I&C system and electric motor-driven pumps which should provide a higher degree of reliability than in currently operating plants. However, the reliability of the I&C system, especially responding to transients and accidents for which it was not specifically designed, cannot be determined. By letter dated June 2, 1993, GE provided a description of the important features identified by the ABWR PRA. This report identified the HPCF logic and control as an important item and stated that "although the probability of a common cause failure of the SSLC is very low, an independent and diverse means of HPCF operation further reduces the risk associated with system operation through the multiplexed digital SSLC." Therefore, the staff concluded that it is prudent to require HPCF backup capability to provide water to the reactor vessel. HPCF backup control in the main control room provides additional assurance that long term core cooling will continue to permit additional time for implementation of the emergency action plan, manning of the RSS, or other long term actions. The staff concluded that only one channel of HPCF capability was needed, and that system-level actuation be provided at the lowest level in the safety computer system architecture. The controls may be hardwired either to analog components or to simple, dedicated, and diverse software-based digital equipment that performs the system-level actuation logic. The above staff conclusions are consistent with the Commission approved diversity position.

2. The following display capability will be added in the control room:

- a. RPV water level
- b. RPV water level (level 3) alarm
- c. Drywell pressure
- d. Drywell pressure (high) alarm
- e. CUW isolation valve status
- f. RCIC steamline isolation valve status
- g. HPCF flow.

The displays may be analog components or simple, dedicated, and diverse software-based digital displays.

3. The RSS displays will be operable during normal operations. This will permit an operator to assess the status of the displayed parameters without transferring control from the main control room.
4. The FDWC system will be designed and tested to verify its availability consistent with the event analyses described above. Testing is required to provide an appropriate level of assurance that the feedwater system can respond to the conditions assumed in the analyses which are beyond the system's normal design basis. The inclusion of only a single I&C channel of high-pressure water injection capability in the main control room is predicated upon satisfactory operation of the feedwater system during the analyzed events as demonstrated by testing. For each event analyses of SSAR Chapter 15 (the GE and LLNL analyses referenced above) which shows that feedwater provides mitigation following the postulated common-mode failure of the safety-related I&C systems function, the FDWC system shall be tested using simulated inputs to demonstrate that it will perform as assumed in the analyses. This requirement is added to the ITAAC.

The above staff conclusions were provided to GE. GE has completed the last of the analyses demonstrating adequate core cooling. The analysis technique was recently revised when GE discovered that the code being used to model the events was not adequate for the steam cooling mode. Using a model that provided steam cooling analysis, GE concluded that the CRD pumps did not provide adequate inventory to mitigate a feedwater line break and, therefore, GE agreed to add hardwired HPCF pump initiation control and flow indication as diverse backup capability. This was the last remaining area of disagreement between the staff and GE, and this issue is now resolved. GE has provided an Appendix C to SSAR Chapter 7 which includes the above commitments to provide adequate defense against potential common-mode failures. Appendix C addresses the issues identified above and specifies the diversity and set of equipment that will not be subject to potential

common-mode failures. Therefore, DFSE Open Items 7.2.6-1, 7.2.6-2, and 7.2.6-3 are resolved and the design meets the staff's proposed applicable regulation for digital instrumentation and control systems.

REACTOR VESSEL WATER LEVEL INSTRUMENTATION

One issue associated with defense-in-depth and the diversity of design required to provide adequate protection against common-mode failures remains open. This issue concerns the RPV water level measurement. The issue was DFSE Open Item 20.3.8. The primary issue concerns the use of identical measurement techniques using condensing chambers and differential pressure transmitters to measure the water level and provide signals to I&C logic. Resolution of this item is discussed in Chapter 20 of this report.

7.2.7 Setpoints

The RPS setpoints will be listed in the RPS TS. The actual setpoints will not be established at this time because the design has not been completed, and the equipment has not been selected. The COL applicant will provide the specific setpoints based on the I&C system design and equipment prior to fuel load. The general requirement is that the setpoints be established high enough to preclude inadvertent actuation, but low enough to assure that proper margin is maintained in the setpoint determination. The TS (SSAR Chapter 16) have been submitted for review and are addressed in Section 16 of this report. Additional TS discussion is also included in Section 7.11 of this report.

GE has committed in the SSAR to meet the guidelines of RG 1.105, which govern instrument setpoints. RG 1.105 endorses ISA S67.04-1982, "Setpoints for Nuclear Safety-Related Instrumentation Used In Power Plants," with some exceptions. This is an acceptable commitment since ISA S67.04 defines a structured analysis acceptable to the staff to determine specific setpoints that adequately considers inaccuracies. This standard is currently undergoing revision and is expected to be issued in the near future. This is an example of a SSAR commitment which may be changed at some later time. The ITAAC for setpoints (which are included in the I&C CDM) requires a plant specific setpoint analysis which details the procedure for establishing specific setpoints. This plant specific analysis will be audited by the NRC during ITAAC implementation. Any changes to the setpoint commitments in the SSAR would involve an unreviewed safety question and, therefore, require NRC review and acceptance prior to implementation. Any requested changes to this commitment shall either be specifically described in the

Instrumentation and Controls

COL application or submitted for license amendment after COL issuance.

7.2.8 Hardware and Software Qualification

The SSLC will be qualified as a Class 1E system. All SSLC components associated with protection systems are Class 1E and will be qualified to the same standards as the protection systems. All programmable digital equipment used for safety-related functions will be qualified in accordance with the safety system design basis with which they interface. This includes environmental and seismic qualification. The SSAR describes commitments to the SRP qualification criteria. The staff reviewed the commitments and has concluded that, for the topics that are addressed by the SRP, the commitments are adequate. The staff identified three additional areas of qualification that are not fully addressed in the SRP criteria. These issues are discussed in SSAR Appendix 7A, and are software qualification, electro-magnetic susceptibility, and mild environmental qualification. These issues were identified in the DSER as open issues.

The ABWR SSLC is dependent upon the proper functioning of the software to perform its safety functions. The standard for software which has been formally endorsed by the staff to specifically address software qualification is ANSI/IEEE ANS-7-4.3.2 (1982), "Application Criteria for Programmable Digital Computer Systems in Safety Systems of Nuclear Power Generating Stations," which was endorsed by RG 1.152 in 1985. The ABWR SSAR stated that, software development will, in general follow this standard. The staff considered this statement unclear and requested GE to provide a clear commitment to the software development process to be followed for the ABWR I&C system. This issue is also related to the GDC 1 issue previously discussed in this report concerning commitments to industry standards, and the issue concerning level of detail identified in the DSER. Because the software products and the software development plans for the SSLC have not been developed yet, there was no method available to the staff to independently verify that the software was in conformance with ANSI/IEEE ANS 7-4.3.2.

ANSI/IEEE ANS 7-4.3.2 is high level description of a software V&V process. Since it was published in 1982 there have been several other standards issued which provide additional guidance for V&V. In addition, there are other aspects to software qualification in addition to V&V. ANSI/IEEE ANS 7-4.3.2 has undergone significant revision and was reissued in November 1993.

To resolve the above issue on software qualification, GE has committed in the SSAR to software development,

documentation and verification, in accordance with the 10 CFR Part 50, Appendix B requirements for quality assurance for safety-related systems and the following standards:

1. ASME NQA2a, Part 2.7, "Quality Assurance Requirements of Computer Software for Nuclear Facility Applications."
2. ANSI/IEEE ANS 7-4.3.2-1982, "Application Criteria for Digital Computers in Safety Systems for Nuclear Facilities." GE has also committed to the revised version of this standard.
3. IEEE 730-1984, "IEEE Standard for Software Quality Assurance Plans."
4. IEEE 828-1983, "IEEE Standard for Software Configuration Management Plans."
5. IEEE 829-1983, "IEEE Standard for Software Test Documentation."
6. IEEE 830-1984, "IEEE Standard for Software Requirements Specifications."
7. IEC 880-1986, "Software for Computers in the Safety Systems of Nuclear Power Stations."
8. IEEE 1012-1986, "IEEE Standard for Software Verification and Validation Plans."
9. IEEE 1033-1985, "IEEE Recommended Practice of Application of IEEE Standard 828 to Nuclear Power Generation Stations."
10. IEEE 1228 (draft), "Standard for Software Safety Plans."
11. IEEE 1042-1987, "Guide to Software Configuration Management."
12. Standard 2167A-1988, "Defense System Software Development."

GE has provided a discussion of software development in Table 7B.1 of the SSAR. In addition to a description of the process, and the commitment to standards, GE has added the following caveat:

Note that the documents listed above may differ regarding specific methods and criteria applicable to the SMP. In situations where such differences exist, all of the methods and criteria presented within those documents are considered to be equally

appropriate and valid and, therefore, any of the above listed documents may be selected as the basis for elements of the SMP.

The staff finds the list of standards acceptable. The software development for the ABWR is included in the I&C CDM. The CDM outlines a software development plan which follows the guidance of the standards included in the SSAR. The standards listed in the SSAR are expected to change over the lifetime of the design certification. The staff expects the COL applicant to use the current version of standards which are available when the software implementation is started. Section 7.1 of this report provides a discussion of the approach to implementation of the ITAAC process for the I&C systems. The staff considers the first step in the staged ITAAC involving development of software design plans to be critical to successful completion of the ABWR I&C system because of the rapidly changing technology in digital I&C systems, and the many different vendors that may be involved in implementation of the ABWR design.

GE has committed to several additional software development methods and features as discussed below. GE committed to use formal methods for the SSLC when formal methods are developed sufficiently. Formal methods are typically defined as entailing a mathematical or occasionally a graphical software specification rather than natural language specification. The outputs would be able to be verified mathematically. Use of formal methods is in a relatively early stage of development at this time. The staff agrees with GE that it is premature to specifically commit to a particular formal method in the SSAR and, therefore, the staff finds the commitment by GE to use the accepted industry software design practices at the time that the software is developed to be appropriate.

GE has committed to provide a safety and hazards analysis, a sneak circuit analysis, and a timing analysis for the digital I&C systems. GE was requested to provide a description of the specifics of the program in these areas. This was DFSER Open Item 7.2.8-1. GE has provided additional details on these items in the CDM and the SSAR, and this item is, therefore, resolved.

GE has committed to the use of software metrics to track error rates during software development. GE has not specified a particular software metric as this will probably be selected by the final software vendor, and is an area that may change over the lifetime of the design certification. This was DFSER Confirmatory Item 7.2.8-1. The SSAR and the CDM have been revised to include the selection of software metrics during the development process. This item is, therefore, resolved. The staff also identified COL Action Item 7.2.8-1 for this

issue since a possible resolution considered when the DFSER was written was for the COL applicant to select the software metrics. The selection of software metrics is now included in the CDM, SSAR and NRC audit of the staged ITAAC. Therefore, this COL action item is no longer applicable and is resolved.

GE stated that "proven technology" will be employed in the design and development of the ABWR I&C systems. The staff requested GE to clarify what was meant by proven technology. In response, GE stated that the definition of proven technology provided in the EPRI RD was appropriate for the ABWR. This EPRI requirement states that three years of successful experience in an application (nuclear or non-nuclear) very similar to the nuclear power plant application is adequate to demonstrate proven technology. The staff agrees with this definition and finds this acceptable. The stated goal, to which both the staff and GE agrees, is to use the best available technology without using unproven designs.

GE has committed to simple modular software programs that follow the guidance of DOD-STD-2167. The staff agrees that safety systems should have simple modular programs and finds this acceptable. GE has not specifically committed to follow DOD-STD-2167 for any other design requirements, though there are similar requirements for a structured design process between DOD-STD-2167 and the GE ABWR SSAR commitments.

At the time of issuance of the DFSER, the issue of commercial dedication of software for use in safety systems had not been adequately addressed by GE. GE subsequently made several commitments regarding commercial dedication of software, and the specific wording to be included in the SSAR. These commitments were identified as DFSER Confirmatory Item 7.2.8-2.

The first aspect of the commercial dedication issue is the use of well-developed operating systems in the development of a plant specific digital system, such as the SSLC. The staff agrees with GE that it is not necessary for the SSLC developer to perform a formal V&V of the operating system. However, it is essential that the SSLC developer assure that the operating system was developed under strict guidelines and has the quality necessary for a safety system.

The second aspect of the commercial dedication issue is the use of a complete component, such as a programmable logic controller, where most of the software has been developed prior to the decision to use it in a nuclear application. As with the operating systems described above, it is necessary for the developer to verify that the equipment selected is of sufficiently high quality for use in

Instrumentation and Controls

a safety system. It is not necessary for the final developer to repeat the V&V activities, but it is necessary for the developer to verify that the original equipment designer has followed equivalent criteria. This concern was resolved with SSAR commitments which were identified as DFSER Confirmatory Item 7.2.8-2.

Included in the commercial dedication issue is the qualification of the automated tools and design support software. It is necessary for the I&C system developer to verify that the tools are accurate. The staff expects the developer to verify the quality of the tools used in the design.

Also related to the issue of commercial dedication is the staff concern regarding communication by the suppliers of errors discovered in the suppliers' tools or software to the end user. This is similar to the 10 CFR Part 21 defect reporting required for Class 1E vendors. This was identified as DFSER Confirmatory Item 7.2.8-3. The CDM and the SSAR have been revised to include the selection criteria for commercial software, accuracy of tools, and notification of the end user by the developer of changes. Therefore, DFSER Confirmatory Items 7.2.8-2 and 7.2.8-3 are resolved.

Any changes to the hardware and software development commitments described in this section of this report would involve an unreviewed safety question and, therefore, require NRC review and acceptance prior to implementation. Any requested changes to this commitment shall be either specifically described in the COL application or submitted for license amendment after COL issuance.

The MPL documents do not add substantially to the information concerning hardware and software qualification. MPL C71-4010 (Rev. 0, May 18, 1990), "Reactor Protection System, Hardware/Software System Specification," lists the inputs and outputs for the DTM, TLU, and OLU but does not provide any additional hardware or software descriptions or specifications.

The second area of digital system qualification not addressed in the SRP criteria and which was an open issue in the DSER, concerns the qualification of the RPS and the other digital I&C equipment for the electromagnetic environment to which it will be exposed. This issue includes EMI, surge withstand capability, electrostatic discharge (ESD), radio frequency interference, and EMC. GE noted in the SSAR that one of the effective means of protection against EMI effects is the redundancy and separation of the divisions of the SSLC. GE committed in the SSAR to the following standards for electromagnetic environmental considerations:

1. ANSI IEEE C63.12-1987, "American National Standard for Electromagnetic Comparability Limits - Recommended Practice."
2. ANSI/IEEE C37.90.2-1987, "IEEE Trial - Use Standard, Withstand Capability of Relay Systems to Radiated Electromagnetic Interference from Transceivers."
3. ANSI/IEEE C62.41-1980, "Guide for Surge Voltages in Low-Voltage AC Power Circuits."
4. ANSI/IEEE C62.45-1987, "Guide on Surge Testing for Equipment Connected to Low-Voltage AC Power Circuits."
5. MIL-STD 461C-1987, "Electromagnetic Emission and Susceptibility Requirements for the Control of Electromagnetic Interference."
6. MIL-STD 462-1987, "Measurement of Electromagnetic Interference Characteristics."
7. IEC 801-2, "Electromagnetic Comparability for Industrial-Process Measurement and Control Equipment, Part 2: Electrostatic Discharge Requirements."
8. IEEE 518-1982, "Guide for the Installation of Electrical Equipment to Minimize Electrical Noise Inputs to Controllers from External Sources."

Commitment to the above standards resolved the DSER open issue. The standards listed above require the selection of specific test categories. Verification of the appropriate selection will be performed during the ITAAC implementation. The selection at the planning stages includes consideration of installation techniques (such as indicated in IEEE-1050 for shielding and grounding), and verification at the site that the installed condition is enveloped by the qualification testing.

EMI protection is included in the I&Cs CDM. One specific feature that the staff requested be included in the CDM is that the digital equipment be tested for the low range of the EMI spectrum as well as the mid to upper ranges. This was DFSER Confirmatory Item 7.2.8-4. The CDM was revised to include the selection of the EMI testing ranges when the equipment has been selected. This is acceptable to the staff and, therefore, this item is resolved. Any changes to these EMI commitments would involve an unreviewed safety question and, therefore, require NRC review and acceptance prior to implementation. Any requested changes to this commitment shall

either be specifically described in the COL application or submitted for license amendment after COL issuance.

The third area of digital system qualification concerns the qualification of the SSLC equipment for the mild environment temperature profiles to which the equipment could be subjected during plant operation. One issue in particular concerned the possibility of local hot spots as a result of higher current densities when using digital chip designs. GE committed to qualification of the SSLC internal panel components to a temperature rise of 15 °C (27 °F) above the normal ambient operating conditions. The electronic equipment panel cooling to maintain qualification is achieved by natural convection of the room air in the panels. Fans may be used to improve long term reliability of electronic equipment, but no credit is taken for forced air circulation for thermal qualification purposes. MPL DMH-4270 (Rev. 2, February 3, 1989), "Essential Multiplexing System Design Specification," lists the CMU and RMU environmental qualification requirements as 10 - 40 °C (50 - 104 °F) temperature, 10 - 60 percent relative humidity, and seismic Category I.

GE stated in the SSAR that the SSLC will be constructed from electronic components purchased to military specifications to the extent practical, and the components will be qualified by testing to higher temperatures than specified in the SSAR for a given room environment. The staff agrees that it is desirable to have this additional margin built into the design. This was DFSEER Confirmatory Item 7.2.8-5. The SSAR has been revised to include this information and, therefore, this item is resolved.

Based on the above evaluation of the RPS qualification, the staff concludes that the commitments in the SSAR meet the requirements of IEEE-279 for equipment qualification and quality of components and modules.

7.2.9 RPS Findings and Conclusions

The design description of the RPS in the SSAR was evaluated to confirm commitments to the SRP and the applicable regulatory guides and industry codes and standards. This review was concerned with the trip parameter sensors, EMS, SSLC and the protection actuation circuits. Based on staff review of the information provided for the sensor and protection actuation circuits, the staff concludes that the SSAR provides acceptable commitments to the appropriate SRP criteria.

The RPS includes systems and components that GE has committed to be designed to survive the effects of

earthquakes, other natural phenomena, abnormal environments and missiles. The staff, therefore, concludes that the GE commitments meet the requirements of GDC 2, "Design Bases for Protection Against Natural Phenomena," and GDC 4, "Environmental and Missile Design Bases," for the RPS.

Based on the review, the staff concludes that the design and the design process for the RPS as described by GE in the SSAR and CDM provides instrumentation to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems. It appears that appropriate controls have been provided to maintain the variables and systems within prescribed operating ranges. Therefore, the staff finds that the RPS design satisfies the requirements of GDC 13.

The staff also concludes, as discussed in Section 7.2.1 of this report, that the GE commitments to the requirements of 10 CFR 50.55a(h) (IEEE-279) and the requirements of GDC 20, "Protection System Functions," for the functional requirements of the RPS are acceptable.

The staff concludes that periodic testing of the RPS as described in the SSAR, and as discussed in Section 7.2.1 of this report, conforms with the criteria of RG 1.22 and IEEE Standard 338 as supplemented by RG 1.118 and is, therefore, acceptable. The staff further concludes that GE commitments to IEEE-279 with regard to system reliability and testability are consistent with the requirements of GDC 21, "Protection System Reliability and Testability," and are acceptable.

The staff concludes, as discussed in Section 7.2.1 of this report, that the RPS as defined by GE meets the criteria of IEEE-384 as supplemented by RG 1.75 for protection system independence with the exceptions noted in SSAR Table 1.8-7, "Summary of Differences from SRP Section 7," and discussed in the SSAR. The staff finds the identified exceptions acceptable, and therefore, the staff concludes that the RPS meets the requirements of GDC 22, "Protection System Independence."

Based on the staff review of results of the FMEA of the RPS in conjunction with the results of the studies of the RPS design for defense against common-mode failures, the staff concludes, as discussed in Section 7.2.1 of this report, that the RPS design adequately meets the requirements of GDC 23, "Protection System Failure Modes."

Instrumentation and Controls

Based on the review of the reactor protection system design, the staff finds that the system is designed to meet the requirements of IEEE-279 regarding control and protection system interaction meets the requirements of GDC 24, "Separation of Protection and Control Systems."

Based on the review of the reactor protection system, the staff concludes that the system satisfies the protection system requirements for malfunctions of the reactivity control system such as accidental withdrawal of control rods. Therefore, the staff finds that the RPS satisfies the requirements of GDC 25.

Based on the review of the protection and reactivity control systems, the staff finds that these systems are designed to assure a high probability of accomplishing their safety functions in the event of anticipated operational occurrences. The staff, therefore, concludes that the design meets the requirements of GDC 29.

The staff review also included consideration of the RPS quality and diversity, as discussed previously in Section 7.2.1 of this report. Based on this review, the staff concludes that GE has specified the appropriate quality requirements, and has provided adequate defense-in-depth and diversity for postulated common-mode failures. The staff has determined that changes to commitments involving (1) computer hardware and software development and quality standards, (2) essential multiplexor design and standards criteria, (3) electromagnetic environment protection criteria and standards, (4) design features and commitments of the SSLC self-test system, and (5) setpoint methodology, would involve an unreviewed safety question and, therefore, require NRC review and acceptance prior to implementation. Any requested changes to these commitments shall either be specifically described in the COL application or submitted for license amendment after COL issuance.

The staff also concludes that the staged audit approach of the ITAAC implementation discussed in Section 7.1.3.3 of this report is an important aspect in the final acceptance of the RPS.

Based on the above discussions and findings, the staff concludes that the design and the design process of the RPS as described by GE in the SSAR and CDM meets the requirements of GDC 2, 4, 13, 20, 21, 22, 23, 24, 25, and 29 and 10 CFR 50.55a(h) (IEEE-279). The RPS design is, therefore, acceptable.

7.3 Engineered Safety Features Systems

7.3.1 System Description

This section describes the I&Cs for equipment in the various ESF systems. ESF system descriptions are provided in SSAR Chapter 6. The ESF systems for the ABWR utilize the SSLC system which is shared with the RPS. Figures 7.3-1 and 7.3-2 of this report provide an overview of the ESF implementation with the SSLC. Section 7.2 of this report discusses the components of the SSLC. Differences between the RPS design implementation and the ESF design are discussed in the following section.

This section describes application to the ESF system of the design bases information identified in IEEE-279, and the various new technology criteria identified by RG 1.152, NUREG-0493 and the ABWR CDM. The additional RPS criteria discussed in Section 7.2 for software, EMI and mild environment qualification also apply to the ESF systems.

The ESF systems are:

- (1) emergency core cooling systems
- (2) LDS
- (3) wetwell and drywell spray mode of RHR
- (4) suppression pool cooling mode of RHR
- (5) SGTS
- (6) emergency diesel generator support systems
- (7) reactor building cooling water system
- (8) essential HVAC emergency cooling water system (HECW)
- (9) high-pressure nitrogen gas supply system

The systems that provide the ESF functions for the ABWR are similar to those of operating BWR designs previously reviewed by the staff. As with the RPS, the primary differences are in the method of implementation of the I&C aspects of the design. The focus of this review was on the use of the EMS and the SSLC system in place of the relay, solid state logic, and copper wire cable of previous I&C system designs.

7.3.1.1 Emergency Core Cooling Systems

The ECCS consists of the HPCF system, the automatic depressurization (ADS), including the safety/relief valve (SRV) electrical actuation logic system, RCIC system, and low-pressure floodler (LPFL) mode of the RHR system. The ECCS I&C (ESF actuation) systems sense the need for ECCS action and initiate appropriate equipment as required. Though the ESF systems share the SSLC and EMS with the RPS, the ESF systems are described in the

SSAR and CDM as separate I&C systems which use the SSLC and EMS.

7.3.1.1.1 High-Pressure Core Flooder System

Four reactor water level and drywell pressure transmitters (one for each division) provide inputs to the SSLC logic, either of which logic signal activates the HPCF system upon a two-out-of-four coincidence of the parameter setpoint. The logic arrangement permits on-line testing of the electronics. After activation, the two HPCF pumps reach rated flow within 36 seconds. The HPCF pumps automatically stop on reactor high water level, or can be manually shutdown. The HPCF pumps are interlocked to prevent starting if an open suction path from the condensate storage tank or suppression pool is not available.

Separation within the HPCF actuation logic system is such that no single failure can prevent system actuation. The logic system is also designed so that no single failure results in a spurious actuation. While the initiating logic is a four-channel system, the actuated system is a two-channel system. The HPCF control is powered by Divisions II and III of the SSLC as shown in Figure 7.3-2 of this report. Divisional separation is maintained between the four sensor inputs (Divisions I, II, III, and IV sensor channels) to the logics, the Divisions II and III logic channels, and the Divisions II and III output controls (output channels).

7.3.1.1.2 Automatic Depressurization System

The MSL inside the drywell have a total of 18 SRV which discharge to the suppression pool. Eight of these valves are designated for use as the ADS. ADS consists of redundant trip channels in two separate logics that control two separate solenoid-operated pilot valves on each ADS valve. Either pilot valve can operate its associated ADS valve. ADS initiation signal is either reactor low water level (L1) and high drywell pressure or a sustained reactor low water level (sustained for 8 minutes). Both parameter setpoints must be reached before ADS is initiated. The valves are interlocked with the HPCF and RHR pump discharge pressure sensors to assure that an HPCF or RHR pump is running prior to depressurization. There is also a time delay (29 seconds) between the completion of the logic voting and the initiation signal. This time delay allows the HPCF or RCIC systems to restore reactor vessel water level if they are available before depressurizing the reactor by actuating ADS. Manual initiation of ADS is also available.

Sensors provide inputs to the four-division SSLC. The ADS actuation output signal to the valves is via Divisions I

and II output channels (electrical divisions) of the SSLC. The SSLC uses eight reactor vessel water level sensors; four different sensors for Divisions I and II. Both electrical divisions are routed to each of the eight ADS valves. The ADS Division I actuation output energizes a solenoid pilot valve on each ADS valve. Similarly, the ADS Division II actuation output energizes the second pilot valve on each ADS valve. Actuation of either solenoid pilot valve opens the ADS valve.

7.3.1.1.3 Reactor Core Isolation Cooling

The RCIC system is a high-pressure injection system which uses a steam turbine-driven pump. The actuated equipment is a single train system. This system is initiated when either high drywell pressure or low reactor water level (L2) setpoints are met. Each parameter has four sensors which provide input via the EMS to the SSLC. The output is via electrical Division I of the SSLC and the EMS.

The RCIC pump turbine is automatically shutdown on turbine overspeed, high turbine exhaust pressure, RCIC auto-isolation signal, low pump suction pressure, reactor water level high (L8), or manual trip if the initiation signal is not present. The RCIC system fails-as-is upon loss of power to the SSLC or loss of input signals. Automatic and manual isolation capability for the RCIC system is provided as part of the leak detection and isolation system.

The RCIC system itself is not redundant because the HPCF system provides functionally redundant capability. Some RCIC system valves are Division II components and appropriate separation of these devices and circuits from the Division I equipment is maintained. System tests are accomplished with a division-of-sensors bypass in place, as discussed in Section 7.2 of this report. On-line signal verification is accomplished by the SSLC. System status annunciation and performance indicators are provided in the control room.

7.3.1.1.4 RHR/Low-Pressure Flooder

The LPFL is an operating mode of the RHR system which is designed to provide water to the reactor vessel following a design basis LOCA. The LPFL is initiated automatically on reactor low water level (L1) signals from the eight water level transmitters of the NBS. The LPFL injection valve actuation logic requires a reactor low-pressure permissive for automatic actuation. These transmitters are separated into four divisions, as discussed in Section 7.2 of this report. Four transmitters provide signals (one from each division) to RHR Divisions I and III, while the other four supply similar signals to RHR Division II. The LPFL system is also initiated on high drywell pressure as sensed

Instrumentation and Controls

by four transmitters (one from each division) of the NBS. The signals from each parameter are combined, through fiber optic isolators, in two-out-of-four logic for each division of LPFL in order to meet the single failure criterion. The RHR/LPFL flow paths are redundant with functional diversity provided by the HPCF and RCIC systems. The SSLC incorporates automatic testing of the instrumentation and verification of the output signals. System status annunciation and performance indicators are provided in the control room. The equipment will be environmentally qualified for the location in which it is to be installed.

7.3.1.2 Leak Detection and Isolation System

The LDS I&C consist of temperature, pressure, radiation, and flow sensors to detect, indicate, and alarm leakage from the reactor primary pressure boundary, and, in certain cases, also close isolation valves to shut off leakage outside the containment. Manual control is provided in the control room for system level isolation of leakage. Each power-operated isolation valve is also provided with a separate manual control switch in the control room independent of the automatic and system level manual logic. All LDS isolation valves are actuated with de-energize to isolate logic.

The LDS system has several isolation capabilities. Containment isolation is initiated on high drywell pressure, low reactor water level (L1, L1.5, L2, L3), manual operator action, and high radiation from the PRM system. Direct operator action, via manual logic reset control, is required to reset the trip condition, provided the initiating signal is cleared. Reactor water cleanup system isolation is initiated on high differential flow and high equipment area temperature. RHR system shutdown cooling suction lines are isolated on low reactor water level (L3) and high ambient temperature. RCIC is isolated on high ambient temperature and high turbine exhaust pressure. The MSL is isolated on low reactor water level (L1.5), high MSL differential pressure, high MSL radiation, high MSL tunnel ambient temperature, high MSL tunnel area temperature in the turbine building, low main condenser vacuum, and low MSL pressure.

7.3.1.3 RHR/Wetwell and Drywell Spray Cooling

The wetwell and drywell spray cooling is an operational mode of the RHR system. This mode uses RHR pumps B and C. The wetwell and drywell spray cooling is manually initiated from the control room, with drywell pressure providing permissive interlocks from NBS sensors and EMS and SSLC system functions for the drywell cooling

mode. The sensor circuits and logic are provided with separation, redundancy and testability consistent with other ESF circuits described in this section and in Section 7.2 of this report. The manual initiating sequence begins with an LPFL initiation signal (low reactor water level). If high drywell pressure and/or high wetwell pressure is also present, the operator will manually close the reactor injection valves, and manually open the spray valves. If low water level occurs again, the system will automatically realign to the reactor injection mode.

7.3.1.4 RHR/Suppression Pool Cooling Mode

The suppression pool cooling mode of RHR uses the same I&C as the LPFL previously described. Redundancy is provided by three separate logic divisions. However, unlike the LPFL, no functional or equipment diversity is identified. This system is automatically initiated upon receipt of a high temperature signal from the SPTM system. The suppression pool cooling mode is also initiated manually. Annunciators and indicators of the RHR systems' operation status in the suppression pool cooling mode are non-safety and available in the control room.

7.3.1.5 Standby Gas Treatment System

The SGTS is initiated automatically upon the detection of high drywell pressure, low reactor water level, high radiation in the fuel handling area or secondary containment HVAC exhaust air. Manual initiation is also available. Two logic divisions are powered from separate ESF buses. The SGTS I&C are supplied power from the Divisions II and III emergency power supplies. Both electrical isolation and physical separation of the divisions are maintained. The system electronics are tested by signal insertion. The SSAR states that SGTS electrical equipment is conformed to the environmental conditions for the area in which it is to be installed. Non-safety-related system status indicators and annunciators are provided in the control room.

7.3.1.6 Emergency Diesel Generator Support Systems

The three emergency diesel generators provide power to and are controlled by Divisions I, II, and III of the Class 1E power supplies. The EDG support systems are described in SSAR Chapter 9 and the associated section of this report. The support systems include the jacket water system, the starting air system, the lube oil system, and the fuel transfer system. Though not specifically mentioned in the SSAR, the support systems also include the EDG HVAC system. The I&C for the support systems are designed to the same criteria as the primary system.

7.3.1.7 Reactor Building Cooling Water System

The I&C system for the reactor building cooling water system consists of two-out-of-four logic with system initiation occurring on low reactor water level or high drywell pressure signals. The instrumentation system's output to the actuated equipment is separated into three divisions such that no single failure can disable this system. Annunciators and indicators of system status are non-safety-related.

7.3.1.8 Essential HVAC Emergency Cooling Water System

The HECW system supplies demineralized chilled water to the cooling coils of the control building safety-related electrical equipment rooms and main control room coolers, and the diesel generator zone air conditioning systems. The HECW system is composed of three divisions, each of two divisions containing two 50-percent capacity refrigerators and chilled water pumps and one division containing one refrigerator and chilled water pump. The systems' I&C output to the actuated equipment are supplied from Divisions I, II, and III power buses.

The HECW system divisions are mechanically and electrically separate. The system is designed to operate during both accident conditions and normal plant operation and during all modes of operation for the control building and diesel generator zone cooling systems. The HECW system operation is initiated automatically when the controls in the main control room are set for automatic operation and any of the HVAC systems located in the control building or diesel generator areas are started. The HECW system can also be started manually from the control room. The HECW system I&C will be tested manually.

The HECW system I&C equipment will be qualified for the particular environment in the area in which it is located as described in Section 7.2 of this report. The environmental qualification of the electrical equipment is also verified via the ITAAC. DFSER COL Item 7.3.1.11-1 discussed the testing and temperature verification that the COL applicant is to include in its pre-operational test procedures in order to confirm electrical equipment environmental qualification. GE revised SSAR Section 7.3.3.1 to add the requirement that the COL applicant include temperature profiles for racks containing Class 1E microprocessor equipment (for various loss of HVAC conditions) in the pre-operational test procedures. This is acceptable to the staff. The COL applicant items associated with I&C equipment cooling are discussed in Section 7.8 of this report.

7.3.1.9 High-Pressure Nitrogen Gas Supply System

The high-pressure nitrogen gas supply system provides compressed nitrogen to the ADS SRV, the MSIVs (for testing only), and other instruments and valves. This system supports both safety- and non-safety-related portions of the plant. The safety-related portion of the system consists of two redundant banks of high-pressure nitrogen bottles and associated piping, valves, and controls powered from separate essential power supplies (Divisions I and II). Upon detection of low nitrogen pressure to the ADS accumulators, this system will automatically isolate the safety-related portion of the system from the non-safety-related portion by isolation valves which automatically terminate the normal nitrogen supply and open the emergency nitrogen gas bottle supply to the ADS accumulators.

7.3.2 Safety System Logic and Control and Specific Subsystem Descriptions

The SSLC is discussed in detail in Section 7.2 of this report. This section discusses the differences between the RPS and ESF portions of the SSLC. The primary difference is the use of safety SLUs in place of the RPS TLUs. As shown in Figure 7.3-2, there are a total of 12 SLUs with four in each of the three SSLC divisions. The SLUs are contained in the portion of the SSLC designated as the logic channel in the TS. The actuated ESF equipment controlled from the SLUs may be in single, redundant or triplicated system trains as described in the system descriptions above. The ESF equipment is divided between the SLUs 1&2 and SLUs 3&4 logic functions to provide some diversity in case of a failure. For example, the HPCF and RCIC high-pressure reactor injection functions are provided by different SLUs than the RHR low-pressure reactor makeup function. The full distribution of systems on the various output channels is presented in the MPL design specifications.

The SLU architecture is arranged so that each ESF logic function has two SLUs performing the same function. Both SLUs (1&2 or 3&4) receive the same input from the DTM, manual controls and bypasses, and in some cases they receive the same direct sensor inputs for interlock protection. The logic in both SLUs must agree before the initiation signal is processed via the output channel to the actuated equipment. This two-out-of-two voting arrangement occurs at the remote multiplexing unit. A single failure of an SLU or EMS channel (one of the two links within an electrical division) will not initiate an ESF function. With the exception of the containment isolation signals which are fail-safe, the logic for the ESF systems is designed to a fail-as-is condition.

Instrumentation and Controls

MPL A32-4080 (Rev. 0, April 25, 1990), "EMS/SSLC Interface Requirements, Requirements Specification," Figure 1, "EMS/SSLC Interface Block Diagram," uses the term "Auxiliary Supporting Features Logic" (ALU) for part of the SSLC while the SSAR and the CDM use the term "Safety System Logic Unit" for the same device. This inconsistency was identified as DFSEI Confirmatory Item 7.3.2-1. GE committed to revise the MPLs which use the ALU terminology to be consistent with the SLU terminology as used in the CDM and SSAR. The ITAAC will verify that these documents are consistent when the design is implemented. Therefore, this item is resolved.

7.3.3 Findings and Conclusions

The general ESF design and arrangements are in accordance with the requirements of the SRP to the extent information was available for review. Many of the issues that are identified in Section 7.2 concerning the RPS also apply to the ESF systems. The level of detail for the ESF systems was an open issue in the DSER. The staff found the level of detail available for review inadequate in the DSER. However, because these systems will be included in the digital I&C design development and ITAAC process, the DSER open issue is resolved. The potential for common-mode software problems may also exist with the ESF systems. This was an open issue in the DSER. The issue of defense against common-mode failures was resolved as discussed in Section 7.2 of this report.

GE had not provided a detailed FMEA for I&C of the ESF systems as required to demonstrate conformance with the requirements of IEEE-279 and the guidelines of NUREG-0493 regarding defense-in-depth analysis. This was identified as an open issue in the DSER and was part of the common-mode failure discussion (DSER Open Item 7.2.6-1). This item is resolved as discussed in Section 7.2 of this report.

The design description of the ESF I&C systems in the SSAR was evaluated to confirm commitments to the SRP and the applicable regulatory guides and industry codes and standards. This review was concerned with the EMS, SSLC, and the ESF system initiation and actuation circuits. Based on staff review of the information provided for the initiation and actuation circuits, the staff concludes that the SSAR provides acceptable commitments to the appropriate SRP criteria.

The ESF actuation system includes systems and components that GE has committed to be designed to survive the effects of earthquakes, other natural phenomena, abnormal environments, and missiles. The buildings containing ESF systems and components will be designed to meet and withstand the probable maximum

flood at the site, and meteorological events. The structures containing the ESF components and the ESF instrumentation and electrical equipment will be seismically qualified. To protect the ESF systems in the event of a postulated fire, the redundant portions of the systems will be separated by fire barriers. The ESF system instrument taps and sensing lines located inside the drywell will be qualified to remain functional during and following a LOCA. The staff, therefore, concludes that the GE commitments meet the requirements of GDC 2, "Design Bases for Protection Against Natural Phenomena," and GDC 4, "Environmental and Missile Design Bases," for the ESF systems.

As discussed above, in Sections 7.3.1 and 7.3.2 of this report, GE has committed that all components of the ESF systems are qualified for the environments in which they are located. Separation and isolation will be preserved, both mechanically and electrically, in accordance with IEEE-279 and RG 1.75. Commitments to other requirements of IEEE-279, such as testing, bypasses, and manual initiation, and corresponding provisions in the design are also described in the SSAR and discussed above. The staff concludes that the GE commitments to the design basis requirements of IEEE-279 and the requirements of GDC 20, "Protection System Functions," for the functional requirements of the ESF actuation systems are acceptable.

In conjunction with SSLC discussed in Section 7.2 of this report, ESF system logic and component testing capabilities will be provided to fully test ESF systems during reactor operation. The staff concludes that periodic testing of the ESF I&C system as described in the SSAR conforms with the criteria of RG 1.22 and IEEE-338 as supplemented by RG 1.118 and is, therefore, acceptable. The staff further concludes that GE commitments to IEEE-279 with regard to system reliability and testability are consistent with the requirements of GDC 21, "Protection System Reliability and Testability," and are acceptable.

Divisional separation of sensor inputs and output channels that will be provided in the design of various ESF systems are discussed above. The evaluation of the SSLC regarding channel separation and electrical isolation is discussed in Section 7.2 of this report. The staff concludes that the ESF actuation systems as defined by GE meet the criteria of IEEE-384 as supplemented by RG 1.75 for protection system independence with the exceptions noted in SSAR Table 1.8-7, "Summary of Differences from SRP Section 7," and discussed in the SSAR. The staff finds the identified exceptions acceptable, and the staff concludes that the ESF actuation systems meet the requirements of GDC 22, "Protection System Independence."

Based on the staff's review of results of the FMEA of the ESF I&C systems in conjunction with the results of the studies of the digital I&C system design for defense against common-mode failures, the staff concludes that the design of the I&C of ESF systems adequately meets the requirements of GDC 23, "Protection System Failure Modes." The ESF systems and components fail in the as-is position in that they require power to operate (i.e., energize to operate). Electrical power is required to perform the emergency functions of the ESF systems and components. The redundancy provided in the design of the ESF I&C, as discussed above, assure that no single failure can cause ESF system failure when required, or inadvertent initiation. The evaluation of defense against common-mode failure in the I&C systems is discussed in Section 7.2 of this report.

The I&C system for ESF has no control function of non-safety systems. However, it does provide isolation signals and inputs to non-safety portions of cooling systems, and annunciators and computers through appropriate isolation devices. Such circuits will be treated as associated circuits or non-Class 1E circuits. Associated circuits will be in accordance with Class 1E circuit requirements up to and including the isolation devices. Non-Class 1E circuits will be separated and isolated from Class 1E circuits or be treated as associated circuits. The staff also concludes that the GE commitments to the design basis requirements of IEEE-279 regarding control and protection system interaction meets the requirements of GDC 24, "Separation of Protection and Control Systems."

The staff's review also included consideration of the ESF system quality and diversity as discussed previously in Section 7.2 of this report. The staff also concludes that the staged audit approach of the ITAAC implementation discussed in Section 7.1 of this report is an important aspect in the final acceptance of the ESF I&C systems.

Based on the above discussions and findings, the staff concludes that the design and the design process of the ESF I&C systems as described by GE in the SSAR and CDM meets the relevant requirements of GDC 2, 4, 20, 21, 22, 23, and 24, and 10 CFR 50.55a(h) (IEEE-279). The design of the ESF I&C systems is, therefore, acceptable.

7.4 Systems Required for Safe Shutdown

7.4.1 System Description

The following systems are identified in the SSAR as required for safe shutdown of the reactor:

- (1) alternate rod insertion function
- (2) standby liquid control system
- (3) reactor shutdown cooling mode of the RHR system
- (4) remote shutdown system

This section provides a discussion of the I&C aspects of these systems, with a review of the interface effects between these systems and the RPS, ESF, EMS, and the SSLC. The four systems addressed in this section have a corresponding CDM section. The review of the CDM for these systems had not been completed at the time the DFSER was issued, and this was identified as DFSER Open Item 7.4.1-1. GE provided the CDM for these systems. The adequacy and acceptability of the CDM is evaluated in Section 14.3 of this report. On the basis of this evaluation, this item is resolved.

7.4.1.1 Alternate Rod Insertion System

The ARI function is accomplished by the rod control and information system (RC&IS), the reactor flow control (RFC) system, and the FMCRD. The ARI system provides the capability for automatic insertion of all rods by an alternate and diverse method from the RPS as necessary for mitigation of an anticipated transient without scram on receipt of high reactor dome pressure and low reactor water level (Level 2) signals. The RC&IS, including the portion for ARI actuation, is not classified as a safety-related system, but is single-failure proof and incorporates features in its design for high reliability and availability. (RC&IS is discussed further in SSAR Section 7.7 and Section 7.7 of this report.) The Level 2 low reactor vessel water level signal is provided via the SSLC (ESF portion) and, therefore, the sensors for this input are Class 1E. In the ARI SSAR description, GE did not indicate whether the RPV water Level 2 inputs to the SSLC are hardwired and, therefore, would not share common equipment with the RPS input. The SSAR needed to clearly state the design for this feature. This was DFSER Confirmatory Item 7.4.1.1-1. GE subsequently revised the SSAR to specify which SSLC signals are multiplexed and which are hardwired. Therefore, this item is resolved.

The requirement for a reactor shutdown system for operational transients is identified in 10 CFR 50.62, "Requirements for Reduction of Risk from Anticipated Transients Without Scram Events for All LWR Designs." Topical Report NEDE-31096-A was submitted by GE to address ATWS events for currently operating BWRs and was approved by the staff. GE indicated, in response to staff questions, their intent to fully conform to NEDE-31096-A for the ABWR. This was identified as DFSER Confirmatory Item 7.4.1.1-2. GE revised the SSAR to include a commitment to the topical report.

Instrumentation and Controls

Therefore, this item is resolved. The staff concludes that the ARI design meets the requirements of 10 CFR 50.62.

The ARI sensor trip setpoints are set above the corresponding RPS settings to allow the RPS trip to occur first. The RPS reactor water level trip is at Level 3 compared to Level 2 for the ARI. The reactor dome pressure trip setpoint is higher than the RPS reactor pressure trip setpoint. Manual actuation of rod insertion requires two manual actions to be taken in order to avoid inadvertent actuation of the rod trips. The logic for the ARI function as described in the SSAR is designed such that no single failure results in the failure to insert more than one operable control rod when the ARI is actuated.

The RPV Level 2 signal from the SSLC is provided to each of the triple redundant recirculation flow control (RFC) system controllers where a two-out-of-four vote is taken in each controller. The reactor dome pressure signals from the steam bypass and pressure control (SB&PC) system are provided to each of the RFC controllers where a two-out-of-three vote is taken. Manual ARI initiation is provided to the same controllers where a two-out-of-two vote is taken. Any of these three RFC controller output signals will initiate the ARI function. The output (trip or no-trip) from each of the RFC controllers is provided as input to the redundant RC&IS controllers where a two-out-of-three vote is taken to initiate FMCRD run-in. (Scram-follow input signals are also provided to the RC&IS.) The output of the two channels of the RC&IS are provided, along with the output of the FMCRD emergency insertion logic channel, to the FMCRD inverter controllers where they are combined in a three-out-of-three vote logic to initiate rod insertion. The output from the RFC controllers also provides input to a two-out-of-three vote logic for the ARI function performed by the redundant scram air header exhaust valves. The ARI system also initiates a recirculation pump trip as described in Section 7.7 of this report.

7.4.1.2 Standby Liquid Control System

The SLCS I&C are designed to initiate the injection of liquid neutron absorber (borated water) into the reactor. The SLCS is a two-train system with one pump for each train. The I&C system is designed to withstand seismic Category I earthquake loads and I&C equipment is mounted in seismically qualified panels. Power for the I&C is provided from the Class 1E instrument bus. The system is designed to be highly reliable with many safety-related system features but it is not classified as a safety-related system. In response to Q420.125 (SSAR Chapter 20), GE stated that the SLCS is hardwired and

does not interface with the EMS. Therefore, the SLCS I&C system does not share any components with the RPS.

Reactor pressure and SLCS borated water storage tank level sensing equipment are used to determine that the liquid neutron absorber is being pumped into the reactor. The system is capable of being tested while the plant is operational by using the SLCS pumps to inject demineralized water into the reactor vessel. Indications, controls and annunciators for SLCS status and control are located in the control room. SLCS status indicators are also provided at the SLCS local control panel to indicate system operating conditions.

The SLCS was initially described in the ABWR design as a manually-initiated system with no capability for automatic initiation. By letter dated June 2, 1993, (regarding important features identified by the ABWR PRA) GE stated that the SLCS would be automatically initiated in order to avoid the potential for operator error and further reduce the probability of adverse consequences due to an ATWS. This change is consistent with the requirements of 50.62(c)(4) for the SLCS automatic initiation. The two SLCS pumps and associated valves will be initiated upon an ATWS initiation signal derived from high RPV pressure and SRNM ATWS permissive for 3 minutes, or low RPV level and SRNM ATWS permissive for 3 minutes. If the control rods have been inserted by the RPS or ARI (automatically or manually) the APRM should indicate downscale before 3 minutes and, therefore, the SLCS would not initiate. The staff finds the SLCS design modification to be in conformance with the requirements of 50.62(c)(4), therefore, acceptable.

7.4.1.3 Reactor Shutdown Cooling Mode of the RHR System

The reactor shutdown cooling mode of the RHR system is initiated by manual operator action with interlocks on the RHR valves to ensure correct cooling mode alignment. The RHR reactor shutdown mode is entered during both normal and emergency shutdown. Normal shutdown is accomplished with all three of the RHR trains in operation and brings the reactor to approximately 51.7 °C (125 °F) within 20 hours following a reactor scram. Emergency shutdown operation brings the reactor to cold shutdown (less than 100 °C (212 °F)) with two RHR pumps operating within 36 hours after control rod insertion. The RHR equipment is Class 1E and redundant, and is seismically and environmentally qualified for the installed location. RHR system controls are located in the control room. The staff finds the reactor shutdown cooling mode I&C design acceptable.

7.4.1.4 Remote Shutdown System

The remote shutdown system provides a means to accomplish reactor shutdown functions (controls and indications) from outside the main control room and bring the reactor to cold shutdown. By letter dated June 2, 1993, (regarding important features identified by the ABWR PRA) GE identified the RSS as an important feature for reducing ABWR core damage frequency due to a control room fire.

The RSS does not include reactor scram capability or complete control of ESF systems. The RSS design assumes that the operator scrams the reactor from the main control room prior to going to the RSS, and that there is no coincident design-basis accident. Two divisionalized RSS panels are provided at separate locations. Their operation is administratively and procedurally controlled. The remote shutdown controls include manual transfer switches to transfer control functions from the control room to the RSS. In addition, on transfer of controls to the RSS, an alarm actuates in the control room.

The staff requested (Q420.15) additional clarification of the intended use of the RSS and the degree of isolation and independence of the RSS from the SSLC and EMS. In the response, GE stated that the RSS is totally separate and independent from the SSLC and EMS because it is hardwired from the sensors to the RSS panels and from the RSS panels to the actuated devices, and does not use multiplexed signal interfaces. Inclusion of this clarification into the SSAR was DFSER Confirmatory Item 7.4.1.4-1. GE revised the SSAR to incorporate the above clarification and, therefore, this item is resolved. The RSS capability is a consideration in the staff's resolution of the common-mode failure issue as discussed in Section 7.2.6 of this report.

The two RSS panels are powered by separate Divisions I and II Class 1E power. Equipment controlled from these panels is powered from the same divisions as when normally controlled from the SSLC. The RSS includes controls for one train of HPCF, two trains of RHR, two trains of RCW, two trains of RSW system, two trains of electrical power distribution system, and the flammability control system. Transfer switches that transfer controls from the control room for these trains of equipment and the emergency diesel generators, are located in the RSS. Indication is also provided to monitor shutdown functions.

7.4.2 Specific Findings

The ARI function and the SLCS instrumentation are part of the resolution of the issue of potential common-mode failure of the EMS and SSLC components. The analysis

to resolve this issue also considered the function of the systems discussed above for mitigation of SSAR Chapter 15 events in combination with postulated common-mode failure in the safety-related digital I&C system, and how operation of these shutdown systems may need to be reconsidered. Specifically, the RSS operation may require reconsideration of the equipment required or the time available to the operator to achieve shutdown using the RSS. This was part of the I&C system common-mode failure analysis which was an open issue in the DSER (SECY-91-294) and DFSER Open Item 7.4.2-1. The use of the equipment described in this section to assist in mitigation of the consequences of a common-mode failure is addressed in Section 7.2 of this report. This open item is, therefore, resolved.

The SSAR did not initially describe how the transfer of sensor outputs from the control room to the RSS would occur without the loss of the calibration data updates stored in the SSLC system microprocessors. The information required to address this issue was part of Open Issue 1 in the DSER (SECY-91-294). GE subsequently revised the SSAR to state that when transfer is made to the RSS, the 4-20ma outputs are routed directly to the RSS, and the automatic calibration function in the RMUs is no longer part of the input signal to the RSS. During the use of the RSS, the automatic calibration function will not be available. Conventional, manual calibration of I&C equipment is available if the length of time of operation of the RSS requires recalibration. This is acceptable to the staff, and this DSER (SECY-91-294) open issue is, therefore, resolved.

7.4.3 Evaluation Findings

The review of the system interfaces for the systems required for safe shutdown included the sensors, circuitry, redundancy features and the actuated devices that provide the I&C functions to prevent the reactor from returning to criticality and provide a means for adequate residual heat removal. The review also addressed the interfaces between the safe shutdown systems and the RPS. The primary characteristics of these systems and requirements are included and verified in the CDM.

The staff concludes that the systems required for safe shutdown are acceptable and meet the relevant requirements of General Design Criteria (GDC) 2, 4, 13, 19, 34, 35, 38, and 44 and the applicable standards and regulatory guides. This conclusion is based on discussions above and summarized as follows:

The staff examined the information submitted for this design to determine its conformance to the GDC, standards and guidelines identified in the SSAR Section 7.1 and the

Instrumentation and Controls

SRP. The staff finds that there is reasonable assurance that systems conform fully to the guidelines applicable to these systems.

The staff's review has included the identification of those systems and components required for safe shutdown which are designed to survive the effects of earthquakes, other natural phenomena, abnormal environments and missiles. Based upon our review we conclude that the design of those systems and components is consistent with the design bases. Additional evaluation is provided in Section 7.1 of this report. Therefore, the staff finds that the design of these systems and components satisfies this aspect of the GDC 2 and 4.

Based on the review, the staff concludes that instrumentation and controls have been provided to maintain variables and systems which can affect this fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems within prescribed operating ranges during plant shutdown. Therefore, the staff finds that the systems required for safe shutdown satisfy the requirements of GDC 13.

Instrumentation and controls have been provided within the control room to allow actions to be taken to maintain the nuclear power unit in a safe condition during shutdown including a shutdown following an accident. Equipment at appropriate locations outside the control room have been provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown, and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures. Therefore, the staff concludes that the systems required for safe shutdown satisfy the requirements of GDC 19.

The staff review of I&C required for safe shutdown systems included determination of conformance to the requirements for testability, operability with onsite and offsite electrical power, and single failure. The staff concludes that these systems as described above incorporate provisions for testability and operability with onsite and offsite power, and to cope in the event of a single failure where applicable and, therefore, meet the relevant requirements of GDC 34, 35, 38, and 44.

7.5 Safety-Related Display Instrumentation and Information Systems Important to Safety

7.5.1 System Description

Safety-related display systems are those which provide information (1) for manually initiated and manually controlled safety functions, (2) to indicate that the plant safety functions are being accomplished, and (3) to provide information from which appropriate action can be taken to mitigate the consequences of anticipated operational occurrences and accidents.

The information systems important to safety provide the operator with the status of the plant to allow manual safety actions to be performed when necessary. The following systems are identified in the SSAR as information systems important to safety:

- a. Safety parameter display system (SPDS)
- b. Information systems associated with emergency response facilities
- c. Nuclear data link

The plant site emergency response center and communications links with the NRC emergency response center are conditions of the COL and are not addressed in this section.

This report evaluates the instrumentation aspects of the information systems with emphasis on interfaces between these systems and the SSLC and EMS as well as the application of advanced technology to processing and display of data important to safety. GE presented in the SSAR a comprehensive list of variables that were considered essential for providing safety-related information to the operators. Tables of conformance and specific exceptions to the guidelines of RG 1.97 were provided in the SSAR, and functional requirements for display of data were provided in the SSAR process system descriptions. One difference between the ABWR and currently operating BWRs is the incorporation of a Class 1E NMS that is in conformance with RG 1.97 Category I instrumentation criteria.

The manner in which the required data is processed and displayed, and dependencies on supporting hardware and software were not described in the SSAR. This was identified as part of the open item in the DSER (SECY-91-294) concerning level of detail (Open Item 1). The details of the displays will be determined during the implementation of the human factor CDM process described in Section 18 of this report. GE has committed to provide the displays for Category 1 RG 1.97 parameters on the fixed mimic panel. These parameters are also available at the operators display console. SSAR Chapter 18, Table 18F, provides a listing of the parameters and the general location of the associated display equipment. The supporting I&C display system equipment is described in Section 7.2 of this report with a similar emphasis on the use of the ITAAC process to implement the I&C design.

The accuracy of the RG 1.97 displays is not specified in the RG 1.97 guidelines or in the SSAR. However, the Emergency Procedure Guidelines imply an accuracy in the displays. For example, the Primary Containment Control Guideline (Ref. SSAR 18A.5) specifies an entry condition when the suppression pool water level is above 7.1 meters or below 7.0 meters. The TS also have similar accuracy requirements. The staff finds that the accuracy requirements implied will be met with the expected equipment. However, because the equipment has not been selected, accuracy will be confirmed in the ITAAC process. The final TS prepared by the COL applicant prior to fuel load will also include the specific setpoints and accuracy for the selected equipment.

NUREG-0737 Item I.D.2 requires that each applicant install a SPDS that will display to operating personnel a minimum set of parameters which define the safety status of the plant. This can be attained through continuous indication of direct and derived variables as necessary to assess plant safety status. Operating reactors have implemented the SPDS with a stand-alone display design. The ABWR is significantly different in that the SPDS parameters are integrated into the total main control room information display design as are the RG 1.97 parameter displays. The staff finds this acceptable. Additional discussion on the SPDS design is provided in Section 18 of this report.

7.5.2 Findings and Conclusions

The scope of the staff's review included an assessment of the proposed application and design of the EMS and SSLC to support operator displays important to safety. Other documentation normally reviewed for information system design such as component states, functional control diagrams, electrical and physical layout drawings, and

descriptive information were not available and will be part of the ITAAC. The staff has considered applicable criteria, guidelines, and design bases, including those for indication of bypassed or inoperable safety systems, in the review discussed in this section.

The staff evaluated the information submitted for the ABWR information system design to determine its conformance to the guidelines identified in the SSAR Section 7.1 and the SRP. The review was concerned with the interfaces between the safety-related information system and the safety-related I&C for systems such as the RPS, LDS, and ECCS.

Table 7.5-7 of the SSAR lists drywell pressure as one of the variables required for indication of manual actions necessary for reactor shutdown from outside the control room. However, the parameters listed in SSAR Section 7.4 for display on the remote shutdown panel do not include this parameter. GE stated that this parameter is not required for shutdown using the RSS in the absence of a postulated design-basis event and should, therefore, not be listed in Table 7.5-7. Revision of this table was identified as DFSE Confirmatory Item 7.5.2-1. Table 7.5-7 was revised to remove the extraneous reference to drywell pressure and, therefore, this item is resolved.

The safety-related display information provided to the operator is derived from the SSLC and EMS. Additional display information is provided from the non-essential systems. The display information was considered in the resolution of the I&C system common-mode failure issue addressed in Section 7.2 of this report because the analysis of a common-mode failure in conjunction with an event included the information available to the operator.

Based on the above discussions and findings, the staff concludes that the ABWR design includes the necessary operator display information and, therefore, meets the requirements of RG 1.97 for post-accident monitoring instrumentation and TMI Action Plan Item I.D.2 for the SPDS, and is acceptable. The detailed information on the safety-related display instrumentation will be reviewed during the implementation of the ITAAC.

The staff concludes that the safety-related display instrumentation and information systems important to safety are acceptable and meet the requirements of GDC 2, 4, 13 and 19. This conclusion is based on discussions above and summarized as follows:

The staff concluded in Section 7.1 of this report that GE has identified in the SSAR the I&C systems which are

Instrumentation and Controls

important to safety. Additional evaluation is provided in Section 7.1 of this report.

The staff has reviewed the design of those systems and components which are required to survive the effects of earthquakes, other natural phenomena, abnormal environments and missiles. Based on the staff's review, the staff concludes that the design of those systems and components is consistent with the design bases. Therefore, the staff finds that the design of these systems and components satisfies this aspect of the GDC 2 and 4.

The staff concludes that the display and information systems to safety include appropriate variables and that their range and accuracy are consistent with the plant safety analysis. Therefore, we find that the information systems satisfy the requirements of GDC 13 and the applicable guidelines satisfy the requirements of GDC 19 with respect to information to operate the unit safely under normal conditions and to maintain it in a safe condition under accident conditions.

7.6 All Other Instrumentation Systems Required for Safety

The instrumentation systems included in this section are those required for safety but not previously discussed in other sections of this report, although some aspects of these systems are included in previous sections of this report.

7.6.1 System Description

The following systems are described in this section:

- (1) neutron monitoring system
- (2) PRM system
- (3) high-pressure/low-pressure interlocks
- (4) drywell vacuum relief system
- (5) containment atmosphere monitoring (CAM) system
- (6) SPTM System

All of these systems are included in the CDM. The CDM had not been reviewed at the time of issuance of the DFSE, and this was identified as DFSE Open Item 7.6.1.1. Descriptions of the above systems have been included in the CDM and this open item is, therefore, resolved. Section 14.3 of this report discusses the acceptability of the CDM.

7.6.1.1 Neutron Monitoring System

The safety-related subsystems of the NMS consist of the SRNM, the LPRM, and the APRM subsystems. The LPRM and the APRM together are referred to as the

PRNM. The non-safety-related portions of the NMS (the automatic traversing in-core probe (ATIP) and multi-channel rod block monitor (MRBM)) are discussed in Section 7.7 of this report.

The SRNM monitors neutron flux from the source range ($1.E+3nv$) to 15 percent of rated power. The SRNM subsystem has 10 SRNM channels with each channel having one fixed in-core regenerative fission chamber sensor. The SRNM preamplifier signals are transmitted to the SRNM digital measurement and control (DMC) units in the main control room. The DMC units contain the software algorithms for signal processing, neutron flux, and power calculations. The SRNM was described by GE as being functionally the same as the wide range NMS in currently operating BWR plants. The SRNM provides trip signals to the RPS, and rod block signals to the rod controls. Each of the four trip channels receives input signals from a different set of SRNM channels. Unlike the other sensor inputs to the SSLC, the NMS provides a trip/non-trip decision directly to the TLU without use of the DTM to process the sensor data.

Three SRNM channels provide input to each of SSLC Divisions I and III, and two SRNM channels provide input to each of SSLC Divisions II and IV. The 10 SRNM channels are divided into three bypass groups. A total of three SRNM channels can be bypassed with no more than one SRNM bypassed per SSLC channel. No additional divisional bypass is allowed. If two (in Divisions II and IV) or three (in Divisions I and III) of the SRNMs are out of service, one channel of the RPS will be tripped.

The PRNM consists of the LPRM and APRM subsystems. The LPRM monitors power in the power range. The LPRM provides signals to the APRM and to the plant computer. The LPRM consists of 52 detector assemblies, each with four fission chamber detectors. The LPRM channels provide trip signals when an LPRM is upscale, downscale, or bypassed. The APRMs consist of four DMC APRM channels, each of which receives the 52 LPRM signals as inputs. The APRM DMC units average the inputs to provide a core average neutron flux which corresponds to the core average power. Each APRM channel is associated with a single RPS trip channel. The APRM also provides rod block functions.

GDC 12 requires that the reactor core and associated coolant, control, and protection systems be designed to assure that power oscillations which result in conditions exceeding specified acceptable fuel design limits are not possible, or can be reliably and readily detected and suppressed. BWR licensees were requested in NRC Bulletin No. 88-07 to take actions to prevent the occurrence of uncontrolled power oscillations during all

modes of operation. The NRC also requested that the BWROG perform generic evaluations of the BWR plant response to core thermal hydraulic instabilities, and provide long-term solutions to prevent or quickly mitigate oscillations or operation in potentially unstable power/flow regions. The BWROG committee on thermal hydraulic stability has developed a long-term stability solution. A licensing Topical Report, NEDO-31960, "BWR Owner's Group Long-Term Stability Solutions Licensing Methodology," has been submitted for NRC review and approval. This topical report describes several options for BWR licensees to choose to deal with stability concerns.

In response to NRC Q440.187, GE stated that in order to meet the stability design requirements specified in the ALWR Utility Requirements Document, Option III, the OPRM system, a microprocessor-based protection system, will be implemented in the ABWR design. The OPRM system uses microprocessors to monitor groups of APRM signals. Upon detection of neutron flux oscillations characteristic of a thermal-hydraulic instability, the system will initiate an automatic suppression function (ASF) to suppress oscillations prior to exceeding safety limits.

Licensing Topical Report NEDO-31960 states that the OPRM which also uses the LPRM input is a Class 1E protection system and conforms to all applicable requirements of IEEE-279-1971. There are four OPRM channels, each of which provides inputs to trip logics which initiate an ASF. The OPRM function is in parallel with, and independent of, the existing Class 1E and non-Class 1E functions of the power range NMS. The OPRM does not affect the design bases for the existing power range monitoring components, their calibration, or their separation.

The OPRM function provides inputs to the ASF for the purpose of suppressing oscillations prior to exceeding the plant minimum critical power ratio safety limit. The OPRM is installed and maintained as an RPS protection function or for a select rod insert (SRI) function. The SRI function is intended to reduce core power to less than the turbine bypass capacity, so that the unit avoids a scram during a load rejection event. For implementation as an RPS function, four OPRM channels provide four separate inputs, one for each RPS trip channel. Any two channels in trip will result in a reactor scram.

GE has indicated and the staff concurs that Option III, LPRM-based, OPRM system is the preferred method of addressing stability. The ABWR OPRM system meets the requirements specified in the EPRI URD for ALWR and is consistent with the guidelines in Topical Report NEDO-31960. The staff, therefore, finds the OPRM system in the ABWR design to be acceptable. This was

identified as DFSEER Confirmatory Item 7.2.1-2. The SSAR and the CDM have been revised to include the OPRM. This item is, therefore, resolved.

7.6.1.2 Process Radiation Monitoring System

Radiation monitoring is provided on a number of process lines, HVAC ducts, and vents. The following radioactive material discharge routes are monitored for radiation:

- (1) MSL tunnel area
- (2) reactor building ventilation exhaust (including fuel exchange area)
- (3) radwaste liquid discharge
- (4) off-gas discharge
- (5) gland steam condenser and mechanical vacuum pump off-gas discharge
- (6) stack discharge
- (7) turbine building vent exhaust
- (8) standby gas treatment ventilation exhaust
- (9) drywell sump liquid discharge
- (10) control building air intake supply
- (11) radwaste building ventilation exhaust

The four MSL tunnel monitors are Class 1E inputs to the RPS and are input to the DTMs for their respective channels.

The reactor building ventilation system radiation monitoring is Class 1E and includes the four exhaust air radiation monitors, the four fuel handling area exhaust radiation monitors, and the eight control building air intake radiation monitors. The exhaust air and the fuel area provide inputs to the leak detection system, and trip the ventilation systems on indication of high radiation. The control building air intake monitors isolate the HVAC for the control room upon indication of high radiation.

7.6.1.3 High-Pressure/Low-Pressure Systems Interlock Protection Functions

The only high/low-pressure interfaces for the reactor vessel involve the low-pressure modes of the RHR system. The logic for the pressure and level sensor inputs which provide the RHR system isolation valve closure signals is a two-out-of-four high reactor pressure or low RPV water level signal.

The inboard and outboard containment/pressure isolation valves for each of the three trains of the RHR system are powered from separate electrical divisions. The valves have permissive logics which prevent them from being opened when reactor pressure is greater than RHR system design pressure or when reactor water level is less than Level 3. These valve closure signals are provided by four

Instrumentation and Controls

divisionalized sensors in a two-out-of-four logic. An additional interlock is provided for the RHR system isolation valves from the RHR system area ambient temperature. The SSAR description of the valve interlocks originally contained an inconsistency concerning the RPV water level at which isolation occurs. This was DFSE Confirmatory Item 7.6.1.3-1. GE subsequently revised the SSAR to resolve the inconsistency. Therefore, this item is resolved.

7.6.1.4 Fuel Pool Cooling and Cleanup System

The fuel pool cooling and cleanup system is classified as a non-safety-related system and is discussed in Section 7.7 of this report.

7.6.1.5 Wetwell-to-Drywell Vacuum Breaker System

Direct control of the wetwell-to-drywell vacuum breakers is not provided. However, the open/close position of these vacuum breaker valves is monitored. The related instrumentation which indicates proper function of the vacuum breakers is described in Section 6.2.1.7 of the SSAR. This instrumentation includes the open/close position indicators for the wetwell-to-drywell vacuum breaker valves and indications of wetwell-to-drywell differential pressure.

7.6.1.6 Containment Atmosphere Monitoring System

The CAM system is a two-train Class 1E system. Each CAM system division monitors the total gamma-ray dose rate and concentration of hydrogen and oxygen in the drywell and/or the suppression chamber. The CAM system is a microprocessor-based system which provides measurement, recording, and alarms in the control room for operator information. Each divisional gamma radiation monitoring channel can be energized manually by the operator or automatically by the LOCA signal. Each divisional hydrogen/oxygen monitoring subsystem is powered continuously during plant operation.

7.6.1.7 Suppression Pool Temperature Monitoring System

The two-train SPTM system consists of eight sensor locations around the circumference of the pool, each of which has a group of four sensors for the two trains. The signal processing for the SPTM system is performed by microprocessors. The I&C of the SPTM system are powered by four divisionally separated electrical buses. The SPTM system initiates RHR suppression pool cooling, RCW load shedding, and RPS trip signaling. It also provides information for the operator in the control room and the remote shutdown panel.

7.6.2 Specific Findings and Evaluation

The SSAR includes an analysis of the safety-related portions of all instrumentation systems required for safety discussed above. The analysis is to show conformance to general functional requirements and specific regulatory requirements. The staff reviewed the information submitted for these systems to determine their conformance to the general design criteria, standards, and guidelines identified in the SSAR Section 7.1 and the SRP. Based on the review, the staff concludes that the design of the instrumentation for the systems in Section 7.6 of the SSAR and discussed in this section meet the requirements of GDC 2, 4, 10, 12, 13, 19, 23, 24, 28, 33 and 44, and the guidelines of applicable standards, regulatory guides, and branch technical positions, as applicable and are, therefore, acceptable. The additional requirements listed in Sections 7.1 and 7.2 of this report for SSLC qualification also apply to the safety-related portions of these systems. These systems are included in the CDM.

7.7 Control Systems

7.7.1 System Description

This section discusses control systems which are considered by GE to be not essential for the safety of the plant. These systems primarily use microprocessor-based equipment and transmit information via non-essential multiplexors. These systems include:

- (1) NBS reactor vessel instrumentation
- (2) RC&IS
- (3) RFC system
- (4) FDWC system
- (5) process computer system and power generation control system (PGCS)
- (6) NMS, ATIP and MRBM subsystems
- (7) automatic power regulator (APR) system
- (8) SB&PC system
- (9) non-essential multiplexing system
- (10) fire protection system
- (11) drywell cooling system
- (12) instrument air systems
- (13) makeup water system
- (14) atmospheric control system
- (15) fuel pool cooling and cleanup system
- (16) communications system

Although these systems are not directly needed for the performance of safety functions, their operation is important to the reliability of the plant. The non-safety systems are designed such that their failure will not prevent the proper operation of the safety systems. These systems are also designed to be of high quality to minimize the

challenges to safety system functions. Because the CDM for these systems had not been reviewed when the DFSEER was written, the staff identified the incomplete review of the CDM and ITAAC as DFSEER Open Item 7.7.1-1. The key features of these systems are now included in the CDM and this item is, therefore, resolved. The codes and standards required for the safety systems as discussed in Section 7.1 of this report do not apply to these systems unless otherwise specified. The digital system design process for the safety systems are similar for these non-safety systems, except that documentation requirements are not as stringent.

7.7.1.1 Nuclear Boiler System Reactor Vessel Instrumentation

Only the non-safety portion of the NBS is included in this section. This part of the NBS provides monitoring and control input of variables during normal plant operations. The variables monitored include:

- (1) reactor vessel temperature,
- (2) reactor vessel water level (shutdown, narrow, wide, and fuel zone ranges),
- (3) reactor core differential pressure,
- (4) reactor vessel pressure,
- (5) SRV seal leak detection,
- (6) feedwater temperature

Sensors for the above variables share sensing lines with the safety system sensors. Separation and isolation is maintained between the safety and non-safety portions of the system.

7.7.1.2 Rod Control and Information System

The RCIS is a non-safety-related system which provides the operator with the information necessary to make changes in nuclear reactivity so that reactor power level and power distribution can be controlled by manipulating the control rods. This system includes those interlocks which inhibit rod movement (rod block) under certain conditions. The RCIS is also used to implement the ARI control rod insertion function, and a backup scram follow function. Upon an RPS scram, this system is initiated and starts the motor-driven FMCRD, to follow the RPS hydraulic scram of the rods.

The RCIS consists of two independent channels for monitoring and control rod positioning during normal

operations. Disagreement between the two channels results in a rod block. The RCIS is designed to be a single-failure proof system. The CRD components which are required for shutdown of the plant and/or whose failure can result in gross fuel damage, are designed to meet the requirements of a safety-related system.

The two RCIS cabinets contain the rod worth minimizer, automated thermal limit monitor, and rod block functions. The RCIS receives the scram follow command from the RPS. It also receives the selected control rod run-in signal from the RFC system and the APR system. The RCIS also provides the signals to the RFC system to reduce flow when the RCIS fully inserts the rods as the result of an ARI initiation or scram follow.

The RCIS allows the operator to completely bypass up to eight control rods by declaring them inoperable and placing them in bypass. The operator can substitute a position for the bypassed rod into the RCIS. The RCIS has a dedicated control interface in the main control room.

7.7.1.3 Recirculation Flow Control System

The RFCS provides each of the two channels of the RCIS with separate isolated trip signals indicating the need for automatic selected control rod run-in upon trip or run-back of the recirculation pumps. This system receives reference power level signals from the NMS and compares the reference power level signals with the power level setpoint. The signals provided represent the validated total core flow. The primary purpose of this system is to control the speed of the 10 reactor internal pumps.

In the DSER (SECY-91-294), the staff expressed concern regarding GE's request to change the normal classification of a loss of all forced circulation from a moderate frequency event to an event not expected to occur in the lifetime of the plant. The staff presented a discussion of the reasons that it did not agree with GE's position. Included in that discussion was the concern that a potential common-mode software error could result in a loss of all forced circulation. In response, GE provided information concerning potential common-mode software failures and the reliability of the RFCS.

The failures addressed by GE included those of sensors, control modules, power supplies, multiplexor links, and speed controllers. Other failure mechanisms such as incorrect operator action, or maintenance and common-mode software errors, were also addressed with means indicated to reduce their likelihood, but such mechanisms are still possible causes of simultaneous reactor internal pump trip. The staff concluded that improvements were made in the reactor recirculation system of the ABWR

Instrumentation and Controls

when compared to currently operating plants; however, the staff considers the simultaneous trip of all of the recirculation pumps due to a common-mode failure to be a credible event that is expected to occur during the lifetime of the plant. Therefore, measures are required to mitigate such an event, or analysis provided to verify that the consequences are acceptable. This event is evaluated in Chapter 15 of this report.

7.7.1.4 Feedwater Control System

The FDWC system is a three-train system that controls the flow of feedwater into the RPV. The I&C system for feedwater operation is a triplicated system which is single-failure proof. The feedwater system can be controlled manually or automatically. Normal automatic control is based on reactor water level (when steam flow is very low) or on reactor water level, main feedwater line flow, and feedpump suction flow measurements during power operation.

7.7.1.5 Process Computer System and Power Generator Control System

The process computer is intended to provide a determination of core thermal performance, and to improve data reduction, accounting, and logging functions. The PGCS, a separate function of the process computer system, monitors overall plant conditions, issues control commands, and adjusts setpoints of lower level controllers to support automation of the normal plant startup, shutdown, and power range operations. The PGCS issues command signals to the turbine master controller.

The staff has reviewed MPL C71-4010 (Rev. 1, July 2, 1990), "Reactor Protection System Design Specification," which requires that both the tripped and reset conditions of the RPS-related sensor instrument channels and the RPS automatic or manual trip systems be logged by the process computer. For all conditions that cause reactor trip, the computer shall identify the specific trip variable, the divisional channel identity and the specific automatic or manual trip system. The staff identified the inputs to the plant computer as DFSER Confirmatory Item 7.7.1.5-1. The plant process computer and its inputs to the trip scram data logger are included in the CDM and the SSAR. This item is resolved.

GE indicated that the plant computer will have the capability to trend performance of all safety-related sensors. This will provide the capability for detection of problems which have occurred previously regarding loss of oil in oil-filled transmitters at operating plants. This resolved the operating experience concern discussed in NRC Bulletin 90-01, and Supplement 1 to Bulletin 90-01.

7.7.1.6 Neutron Monitoring System, ATIP and MRBM Subsystems

The ATIP subsystem of the NMS is comprised of three TIP machines, each with a neutron sensor attached to a flexible cable. The system includes the associated drive mechanisms and guide tubes, and is used to obtain flux readings along the axial length of the core. The MRBM subsystem logic issues a rod block signal to the RCIS. This microprocessor-based system receives input neutron flux signals from the LPRMs and APRMs, core flow data from the NMS, and control rod status to determine when rod block signals are required.

7.7.1.7 Automatic Power Regulator System

The APR system controls reactor power by providing commands to rod position or reactor RFC instrumentation. The APR receives input from the plant process computer, the PGCS, the SB&PC system, and the operator's control console. The output demand signals from the APR are to the RCIS and the RFC and SB&PC systems. The APR logic is performed by redundant microprocessors.

7.7.1.8 Steam Bypass and Pressure Control System

The SB&PC system controls the reactor system pressure during normal operation. The system regulates the position of the turbine control and/or steam bypass valves.

7.7.1.9 Non-Essential Multiplexing System

The NEMS is separate from the EMS but is similar in function. The NEMS supports communication between the non-safety I&C systems. The NEMS will be diverse from the EMS (hardware and software).

7.7.1.10 Fire Protection System

The I&C aspects of the fire protection system consist of the detection and suppression portions. This system is automatically actuated by smoke, infrared, or temperature detectors when fire is indicated.

7.7.1.11 Drywell Cooling System

The drywell cooling system is used to limit the temperature of the various drywell zones within ranges dictated by the equipment requirements.

7.7.1.12 Instrument Air System

The instrument air system is discussed in Section 9.3.6 of the SSAR and is not discussed in this section.

7.7.1.13 Makeup Water System

The makeup water system is discussed in Section 9.2.3 of the SSAR and is not discussed in this section.

7.7.1.14 Atmospheric Control System

The atmospheric control system is discussed in Section 6.2.5 of the SSAR and is not discussed in this section.

7.7.1.15 Fuel Pool Cooling and Cleanup System

The fuel pool cooling and cleanup system was reclassified as a completely non- Class 1E system since the DFSER was written. It is an independent system which monitors and controls the fuel pool temperature and maintains the water quality of the pool. This system consists of redundant trains with power supplies that are backed by the combustion turbine generator.

7.7.1.16 Communications System

The paging system is designed to provide facilities for mutual communication and simultaneous broadcasting within the plant. The sound-powered telephone system provides communications primarily for fuel transfer, testing, calibration, and maintenance.

The communication systems consist of a power-actuated paging facility and a separate network of cables and jacks to facilitate the use of sound-powered telephones for maintenance and repair. The paging system is primarily used for intraplant communication during plant operation. Handsets and speakers are installed in the rooms indicated below:

- (1) main control room
- (2) electrical equipment room
- (3) fuel replacement area
- (4) turbine operation area
- (5) periphery of control rod hydraulic units area
- (6) feedwater pump room
- (7) elevators
- (8) exteriors of plant buildings

In addition to the basic paging function, the paging equipment can be used for automatic surveillance of the main amplifier and manual switching to a spare amplifier as necessary. The paging equipment produces an emergency signal (siren) upon actuation of an emergency pushbutton. The circuits from the main paging equipment to each junction box are wired in separate routes.

A separate telephone communication system using portable sound-powered telephones is provided, but is outside the scope of the ABWR design. The system provides communication between boards in the main control room, between the main control room and field stations, and between field stations during testing or inspections.

The communication systems do not have a safety-related function. However, they are used during emergencies and because the communication system is required when plant control is at the remote shutdown station, its availability must be demonstrated assuming a main control room fire. This was DFSER Open Item 7.7.1.15-1. GE subsequently revised SSAR Chapter 9.5.2 to describe the survivability of the sound-powered phones in the event of a control room fire. This item is, therefore, resolved.

No identification of the EMI radiation levels or frequency range is identified for the communication transmitter/receivers to be installed in the plant. In addition, the sensitivity of the safety computer systems to the electromagnetic fields is undefined. Therefore, a test program with field measurements and operational descriptions is required if spurious effects upon safety-related I&C equipment due to communications is to be avoided. This was identified as DFSER Open Item 7.7.15-2. This will be verified as part of the EMI ITAAC that is included in the instrumentation and controls CDM. Therefore, this item is resolved.

Based on the above, the staff concludes that the design of the paging communication system ensures its availability as required and is, therefore, acceptable.

7.7.2 Specific Findings and Evaluations

In the SSAR, GE provided an analysis to demonstrate that the above non-safety-related I&C systems are not required for any plant safety function, and that the plant protection systems are capable of coping with all failure modes of these I&C systems. The analysis shows how the design of the above I&C systems conforms to general functional requirements and specific regulatory requirements. The staff reviewed the information submitted for these systems to determine their conformance to the GDC, standards, and guidelines identified in the SSAR Section 7.1 and the SRP. Based on the review, the staff concludes that the design of the I&C systems in Section 7.7 of the SSAR and discussed in this section meet the requirements of GDC 13 and 19, and the guidelines of applicable standards and regulatory guides and is, therefore, acceptable.

7.8 COL License Information

Section 7.8 of the SSAR provides a discussion of topics that are not specifically addressed in the SRP but are important to the design and operation of safety-related equipment. These topics will be addressed by the COL applicant in its application. These topics include: (1) effects of station blackout on HVAC systems and the subsequent effects on plant electronics; (2) effects of ESD on exposed electrical and electronic equipment components; (3) effects of localized high heat spots in semiconductor materials for computing devices; (4) criteria for interfaces between the ABWR I&C systems and systems outside of the scope of the ABWR design certification; and (5) comprehensive functional tests necessary to meet the plant TS.

The staff requested (Q420.014) that GE address the effects of station blackout on that portion of the HVAC system which is required to maintain the function of plant electronics. GE responded that this issue will be addressed as a COL applicant requirement by performance of a temperature/heat rise analysis for the station blackout (no HVAC) scenario using the resulting environmental temperatures for the specific plant location as a basis for confirming appropriate electronic equipment performance. The heat rise analysis will verify that the required electronic equipment has been qualified to the highest expected temperature assuming station blackout. This is acceptable and will be verified as part of the ITAAC process by the COL applicant.

The staff requested (Q420.90) that GE address the possible effects of ESD on the proper performance of keyboards, keyed switches, and other exposed electrical equipment components. GE's response described the damage to components and system upsets that ESD can cause. The response described the steps used in modern equipment design to protect the equipment from ESD and precautions that should be used in their installation. The equipment design standards, installation and maintenance procedures are included as part of the EMC ITAAC. The COL applicant will verify by the EMC ITAAC that the GE recommendations for grounding and shielding are followed or provide an acceptable alternative.

The staff requested (Q420.92, Q420.94, Q420.95) that GE address possible localized hot spots in semiconductor materials due to high current densities and their effects on equipment reliability. GE's response described the requirement for a thermal analysis which will follow the methods of MIL-HDBK-217 and MIL-HDBK-251. Because the worst case for potential hot spots occurs during a postulated station blackout scenario (no HVAC) and, therefore, will require plant-specific information to be

incorporated in the thermal analysis, the staff agrees that this issue will be addressed as part of the ITAAC process. The COL applicant will verify that adequate compensation is provided for internal heat rise to ensure proper electronic equipment performance.

GE performed an interface study of each of the I&C systems included in Chapter 7 of the SSAR, and determined that there are no safety-related electrical signal interfaces between safety systems and systems that are site specific that have not already been included in the requirements of the previous sections. Therefore, GE indicated that there are no interface requirements necessary to ensure safety-related system performance. However, the SSAR did not specifically address non-safety I&C system interfaces with safety-related systems outside the ABWR scope. If there are any such interfaces with equipment outside of the scope of the ABWR SSAR, the existing requirements for safety/non-safety I&C system isolation will apply and will be verified during the ITAAC phase. The COL applicant will verify that any safety/non-safety I&C interfaces are adequately separated and isolated.

The TS contain several different surveillance test requirements. One of those tests is a comprehensive functional test of the safety-related I&C systems that will be performed during each refueling outage. The TS require the tests, and the bases for the TS provide a short description of the testing that is to be done. GE has also included a detailed description of the comprehensive functional test in the SSAR. This description will be included by the COL applicant in its maintenance program.

7.9 Appendix 7A - Design Response to Appendix B of ABWR Licensing Review Bases

Appendix B to the GE ABWR LRB, dated August 1987, noted that the SRP did not provide standards and criteria for the review of state-of-the-art fiber optics, multiplexing, and computer controls. LRB Appendix B contains questions from the staff regarding digital I&C systems. The questions were intended to solicit additional information from the applicant beyond that normally provided in SAR because of the use of digital technology in the design. GE provided their responses to these questions in Appendix 7A of the ABWR SSAR. Since the DFSE was issued, GE also added SSAR Chapter 7 Appendices B and C to further address digital I&C system design issues, including defense-in-depth.

The responses in Appendices 7A, 7B, and 7C provide a level of design information similar to that provided in the I&C systems sections of the SSAR and in responses to

other staff requests for additional information. The information provided was not sufficiently detailed in and of itself to permit the staff to reach its safety conclusions. An example is the GE statement in the response to NRC Request 11 (SSAR Appendix 7A) that "software development will, in general, follow RG 1.152." The staff concludes that the Appendix 7A response provides a commitment to address the issues involving digital system design as identified in Appendix B to the LRB, but does not present sufficient description of the design to demonstrate how the commitments are to be met. Appendix 7A was revised following the DSER (SECY-91-294) to include a commitment that the ABWR digital systems will meet the criteria of the additional standards listed in Sections 7.1 and 7.2 of this report. Verification that the I&C system design conforms with this commitment will be accomplished during the ITAAC phase.

In addition to the hardware and software aspects of the safety-related I&C system design that have been discussed previously in this report, SSAR Appendix 7A provides a commitment to the guidelines of RG 1.153 which endorses IEEE-603. For the I&C systems of the ABWR design, IEEE-603 provides guidance which is similar to that of IEEE-279. GE stated that the ABWR safety-related I&C system design be in conformance with IEEE-603, and the implementation of the I&C design will be verified during the ITAAC. This is acceptable to the staff.

Based on the review of the information provided in the SSAR (Appendices 7A, 7B, and 7C) and in related CDM, the staff considers that issues regarding standards and criteria for digital equipment, raised by the LRB Appendix B questions, have been adequately addressed.

7.10 Unresolved Safety Issues, Generic Safety Issues, and Operating Experience

7.10.1 Unresolved Safety Issues and Generic Safety Issues

USIs and GSIs are discussed in Chapter 20 of this report.

7.10.2 Operating Experience

The EPRI ALWR Utility Requirements Document, Volume II, for evolutionary reactor design provides a general description of the operating experience that is necessary before equipment should be considered for use in future nuclear power plants. The general guidelines specify that approximately 3 years of successful experience in applications similar (but not necessarily nuclear plant-related) to the intended nuclear power plant installation is appropriate. Failure to meet this EPRI requirement results in increased prototyping as specified in the EPRI

requirements. GE has committed to the EPRI requirements and the staff finds this acceptable. This was DFSER Confirmatory Item 7.10.1-1. The SSAR has been revised to address operating experience as indicated above, and this is acceptable. DFSER Confirmatory Item 7.10.2-1 is, therefore, resolved.

In addition, as part of the operating experience review, the staff reviewed the following NRC bulletins and generic letters (GLs) which have been issued since 1980. The primary intent of this review effort was to assure that the operating experience gained as described in these documents is incorporated into the design and operating features for the ABWR.

The generic communications reviewed by the staff for operating experience in the I&C systems area and a discussion of the review follows:

NRC Bulletins

- a. Bulletin 80-01 (January 11, 1980), "Operability of ADS Valve Pneumatic Supply."

This item is not applicable to the SSAR Chapter 7 I&C system review.

- b. Bulletin 80-06 (March 13, 1980), "Engineered Safety Feature Reset Controls."

Bulletin 80-06 listed three actions to be taken by the licensee. The first was to review the I&C system schematics and verify that upon reset of an ESF actuation signal, the safety-related equipment remains in its emergency mode. For the ABWR, the ESF systems are reset individually and manually, and will remain in their emergency mode. This is acceptable.

The second item requires verification that the as-built I&C system configuration is in conformance with the schematics. This will be verified by the COL applicant during the ITAAC phase, and is acceptable.

The third item pertains to plant-specific corrective actions by operating plants and does not affect the ABWR. The concerns of Bulletin 80-06 have been adequately addressed in the ABWR design.

- c. Bulletin 80-20 (July 31, 1980), "Failures of Westinghouse Type W-2 Spring Return to Neutral Control Switches."

Based on the information provided by GE, this type of switch will not be used in the ABWR design and, therefore, Bulletin 80-06 is not applicable.

Instrumentation and Controls

- d. Bulletin 81-02 (April 9, 1981), "Failure of Gate Type Valves to Close Against Differential Pressure."

This item is not applicable to the SSAR Chapter 7 I&C system review.

- e. Bulletin 82-04 (December 3, 1982), "Deficiencies in Primary Containment Electrical Penetration Assemblies."

This item is not applicable to the SSAR Chapter 7 I&C system review.

- f. Bulletin 88-07 (June 15, 1988), "Power Oscillations in Boiling Water Reactors."

The measures provided in the ABWR I&C system design to deal with power oscillations are addressed in Sections 7.2 and 7.6 of this report.

- g. Bulletin 90-01 (March 9, 1990), "Loss of Fill Oil in Transmitters Manufactured by Rosemount."

GE committed for the ABWR I&C system design not to use the transmitters of concern identified in the bulletin which were built before July 1989. In addition, the plant computer for the ABWR will have the capability to trend the operational data of all safety-related transmitters. This trending is capable of detecting the type of problems described in Bulletin 90-01. This is, therefore, acceptable.

NRC Generic Letters

- a. GL 80-03 (April 10, 1980), "Clarification of the Term "Operable" as it Applies to Single Failure Criterion for Safety Systems Required by TS."

This item is not applicable to the SSAR Chapter 7 I&C system review.

- b. GL 83-27 (July 6, 1983), "Surveillance Intervals in Standard Technical Specifications."

This item is not applicable to the SSAR Chapter 7 I&C system review.

- c. GL 88-02 (January 20, 1988), "Integrated Safety Assessment Program II (ISAP II)."

This item is not applicable to the SSAR Chapter 7 I&C system review.

- d. GL 88-20 (November 23, 1988), "Individual Plant Examination of Severe Accident Vulnerabilities."

This item is not applicable to the SSAR Chapter 7 I&C system review.

- e. GL 89-14 (August 21, 1989), "Line-Item Improvements in Technical Specifications - Removal of 3.25 Limit on Extending Surveillance Intervals."

Issues in GL 89-14 related to I&C system TS are discussed in Section 7.11 of this report.

- f. GL 91-04 (April 2, 1991), "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle."

Issues in GL 91-04 related to I&C system TS are discussed in Section 7.11 of this report.

- g. GL 91-09 (June 27, 1991), "Modification of Surveillance Interval for Electrical Protection Assemblies in Power Supplies for the Reactor Protection Systems."

Issues in GL 91-09 related to I&C system TS are discussed in Section 7.11 of this report.

One issue associated with operating experience concerns an aspect of the adequacy of the defense-in-depth and the diversity of the instrumentation design required to provide defense against common-mode failures remains open. Specifically, the issue concerns the need for diverse reactor pressure vessel water level measurement instrumentation. This issue was Open Item 20.3.8 in the DFSER and is also addressed in Section 7.2 of this report. The primary concern is in identical measurement instrument techniques used (condensing chambers and differential pressure transmitters) to measure the RPV water level and provide signals to the I&C logic.

Recent anomalies have been observed in RPV level instrumentation (and discussed in GL 92-04, "Resolution of the Issues Related to Reactor Vessel Water Level Instrumentation in BWRs Pursuant to 10 CFR 50.54(f)," Information Notice 92-54, "Level Instrumentation Inaccuracies Caused by Rapid Depressurization," and Bulletin 93-03, "Resolution of Issues Related to Reactor Vessel Water Level Instrumentation in BWRs") that were caused by the effects on non-condensable gas in the condensing chamber and reference leg of the water level instrumentation. GE committed to a modification of the design for the ABWR water level instrumentation by revising the sensing line connections and adding a water backfill capability to minimize the possibility of entrapping non-condensable gas in the reference legs and condensing chambers. The proposed modifications will be tested to

validate the appropriateness of the modification. This issue is discussed in more detail in Chapter 20 of this report.

Operating experience at Grand Gulf Station (found during testing) has demonstrated that the solenoid coils on scram pilot valves can be damaged by undervoltage and thereby prevent the scram function. During the previous licensing review of Hatch 2, this issue was also raised and included questions of possible damage due to undervoltage to the RPS equipment in general, not just to the scram pilot valve solenoid coils. Resolution of this concern is addressed in Section 7.2.5 of this report.

A significant feature of the I&C system testing capability for the ABWR described by GE in the SSAR is the elimination of the need to lift leads and install jumpers when performing tests. Lifting leads and installing jumpers has been a source of problems at operating plants. This issue was part of DFSER Confirmatory Item 7.2.2.5-2. GE provided more detailed descriptions in the SSAR of the I&C system self-diagnostics and testing equipment to be used for surveillance and maintenance. The ABWR design has eliminated the necessity to lift leads or add jumpers for normally scheduled surveillance and maintenance. This item is, therefore, resolved.

Based on the above, the staff concludes that GE had adequately addressed and incorporated into the ABWR design for the I&C systems, features which deal with concerns identified from past operating experience. Therefore, operating experience issues are resolved.

7.11 Technical Specifications

The staff reviewed the draft ABWR TS provided prior to the issuance of the DFSER. Substantial questions were identified as discussed below, which required resolution prior to the staff finding the ABWR TS acceptable and, therefore, the TS were an open issue in the DFSER.

The design of the ABWR I&C systems is substantially different from that used in currently operating BWR plants and, therefore, the current BWR standard TS sections for the I&C systems were not readily applicable to the ABWR I&C systems. GE has substantially revised the ABWR TS to reflect the specific design of the ABWR.

Because the proposed ABWR I&C systems have not yet been in operation, there is no specific equipment history that can be used to assess the equipment surveillance intervals. The specific I&C equipment and vendors have not been selected, and are not required to be selected until after the COL is issued. The specifics of the I&C system equipment self-diagnostics are to be considered in assessing the TS surveillance intervals.

Prior to the issuance of the DFSER, GE had not claimed credit for the self-diagnostics to meet any particular TS requirements. The self-diagnostics are an integral part of the design, and form a significant basis for the establishment of surveillance intervals for equipment beyond the self-diagnostics. The inclusion of the self-diagnostics in the design is necessary to the acceptance of the TS as discussed in Section 16 of this report.

MPL C71-5030 (Rev. 0, April 23, 1990), "Reactor Protection System Verification and Validation Criteria Design Specification," states that field installation and validation tests for the RPS will be performed while the multiplexing loops for each RPS division are operational. The DFSER stated that the limiting conditions for plant operation must address the possibility of a multiplexor loop being out of service. The ABWR TS incorporate limiting conditions for this situation, and therefore, this item is resolved.

MPL C71-4010 (Rev. 1, July 2, 1990), "Reactor Protection System Design Specification," states that one sensor channel may be bypassed and the system actuation will be based on a two-out-of-three logic. This bypass is implemented at the input to the TLU. Bypass of one division of output logic will also result in a two-out-of-three coincidence logic. This bypass is implemented at the input to the OLU. There is no indication in the MPL document if both bypasses are allowed at the same time. The initial draft ABWR RPS TS referred only to the traditional channel check calibration and functional tests. In the DFSER, the staff stated that a description of how these I&C channel and division bypasses are to be implemented on the ABWR was required. A functional channel test should be defined. Subsequent versions of the ABWR TS added the definitions of sensor, logic, and output channels as well as a discussion of the various surveillance requirements including bypass conditions. This item is, therefore, resolved.

Plant operation in the event of power supply failures needed to be considered in the I&C system TS. A loss of one channel of SSLC power not only disables one channel of sensors and one channel of coincidence logic, but also disables 1/4 of the inputs to the other three coincidence logics. The DFSER stated that the TS needed to consider each of the credible equipment failures and the specific TS action that it invokes, if any. The TS were revised to consider credible failures and the appropriate statements of action have been incorporated. This item is, therefore, resolved.

The draft ABWR TS initially indicated that an indefinite bypass (no repair required until the next refueling outage) was appropriate for a channel out of service. No bases

Instrumentation and Controls

were provided to justify this condition. The staff noted that there is a potential single failure of the Division II, 6.9 KV/480 Vac transformer or 480 Vac switchgear which could disable both the Divisions II and IV SSLC, and at the same time not cause a reactor scram. If a Division I or III SSLC channel is in bypass at the same time, the result could be a loss of RPS function.

By letter dated June 2, 1993, on important features identified by the ABWR PRA, GE identified that the four divisions of the SSLC are designed to be highly reliable with features that reduce the possibility of inadvertent actuations. The PRA assumed a self-diagnostic fault detection rate for the SSLC of 0.95 with the remainder of the faults expected to be found during the quarterly SSLC surveillance required by TS. As discussed earlier, there is a lack of operational data for the SSLC equipment to support the PRA number.

Because of the above reasons, GE has revised the ABWR TS to remove the indefinite bypass of an SSLC channel. This is acceptable to the staff.

In the DFSER, the staff noted that specific operability aspects of the ABWR I&C system digital technology needed to be further evaluated when developing the TS. For example, if a failure of a safety-related I&C system (channel or division) during surveillance can be attributed to a software error, the appropriate TS operability requirements for other systems which may also be subject to failure as a result of the same software error are needed to be established. The TS were revised to include limits on continued operation and programmatic consideration (including reporting to the NRC) of software errors when they are discovered. This is acceptable to the staff.

The TS for design certification include I&C system setpoints that will be established by the COL applicant during the I&C system detailed design development. The setpoint methodology is included in the CDM and its implementation is included in the ITAAC.

Based on the above, the staff concludes that the ABWR TS for design certification incorporate the necessary operability and surveillance requirements for digital I&C systems and are, therefore, acceptable.

8 ELECTRIC POWER SYSTEMS

8.1 Introduction

The staff reviewed the advanced boiling water reactor (ABWR) design descriptions and commitments documented in Chapter 8 of GE Nuclear Energy's (GE's) standard safety analysis report (SSAR) Amendment 32. The bases for evaluating the adequacy of ABWR electric power systems presented in SSAR Chapter 8 were the acceptance criteria and guidelines for electric power systems contained in standard review plan (SRP) Chapter 8 and Regulatory Guides (RGs) 1.153 (Rev. 0), "Criteria for Power, Instrumentation, and Control Portions of Safety Systems," and 1.155 (Rev. 0), "Station Blackout." The Nuclear Regulatory Commission (NRC) approved these regulatory guides following the issuance of SRP Revision 3 (July 1983), and they apply to the ABWR electric power system design.

The staff's initial findings were included in the draft safety evaluation report (DSER), SECY-91-355. In addition, the staff's interim findings were included in the draft final safety evaluation report (DFSER), SECY-92-349. Open, confirmatory, and COL action items from these documents have been referenced in the discussions below.

8.2 Offsite Electric Power System

The offsite electric power system is commonly called the "preferred" power system. The staff's evaluation of this system focused on the system's importance as the preferred supplier of electric power for the onsite power system (that is, the Class 1E ac-distribution system), which supplies power to safety systems.

For the ABWR, the preferred power system comprises the following circuits:

- Normal preferred power circuit - a back-feed circuit from the transmission network to the input terminals of each of the three redundant, onsite Class 1E ac-distribution systems through the main transformer and three unit auxiliary transformers.
- Alternate preferred power circuit - from the transmission network through one reserve auxiliary transformer to the input terminals of each of the three redundant, onsite Class 1E ac-distribution systems.

Because GE shares the ABWR design responsibility for this system with the combined license (COL) applicants, those parts that are outside the scope of design of the ABWR standard plant, and those parts that are within the scope of design of the ABWR standard plant are described and evaluated as follows.

8.2.1 Preferred Offsite Circuits Outside the ABWR Scope of Design

The following portions of the preferred power circuits are outside the scope of design of the ABWR standard plant:

- Normal preferred power circuit from the transmission network through the main power transformer to the low-voltage terminals of the main transformer.
- Alternate preferred power circuit from the transmission network through the reserve auxiliary transformer to the low-voltage terminals of the reserve auxiliary transformer.

8.2.1.1 Scope of GE Design of Offsite Preferred Circuits

Section 3.1.2.2.8.2.2 and Sections 8.2.1, 8.2.2, and 8.2.3 of SSAR Amendments 7 and 10 were inconsistent with regard to which parts of the offsite system are within (or outside) the ABWR standard design scope (DSER (SECY-91-355) Open Item 22). Subsequently, GE committed to revise Section 3.1.2.2.8.2.2 of SSAR Amendment 7 so that it would be consistent with the above defined scope of design.

Verification that GE included appropriate changes in a future SSAR amendment to reflect the above information was DFSER (SECY-92-349) Confirmatory Item 8.2.1.1-1. GE included this information, in Sections 3.1.2.2.8.2.2 and 8.2.1 of SSAR Amendment 32, which is acceptable.

8.2.1.2 Definition of Offsite System

GE's draft SSAR submittal of April 3, 1992, indicated that the offsite power system begins at the terminals on the transmission network side of the circuit breakers connecting the switching stations to the offsite transmission network. That draft also indicated that the offsite power system ends at the terminals of the plant's main generator and at the circuit breaker input terminals of the medium-voltage (6.9 kV) switchgear. This description is not consistent with the NRC SRP definition of an offsite system DSER (SECY-91-355) Open Item 27. Specifically, GE's definition appeared to exclude the transmission network, as well as the plant's main generator and gas turbine generator. GE committed to revise the SSAR in a future amendment to be consistent with the NRC SRP definition of an offsite system.

Verification that GE included appropriate changes in a future SSAR amendment to reflect the above information was DFSER (SECY-92-349) Confirmatory Item 8.2.1.2-1. GE included this information, in Sections 8.1.2.1 and

Electric Power Systems

8.2.1.1 of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

8.2.1.3 Offsite Power System Interfaces

This section addresses the staff's evaluation of GE's response to DSER SECY-91-355 Open Item 24.

In the draft SSAR submittal dated April 3, 1992, GE responded to Open Item 24 from the DSER (SECY-91-355) by defining interface requirements for the offsite circuits outside the scope of design of the ABWR standard plant. GE committed to document similar interfaces in the SSAR in a future amendment. Verification that GE included appropriate changes in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.2.1.3-1. GE included the appropriate changes in Section 8.2.3 of SSAR Amendment 33, which is acceptable. Therefore, this item is resolved.

Section 8.2.3.1 of GE's draft SSAR submittal of April 3, 1992, indicated that a COL applicant "should" meet the interface requirements defined in Section 8.2.3 of the ABWR SSAR. Applicants who reference the ABWR design will be required to address all interface requirements in its design scope. GE committed to revise the SSAR in a future amendment to indicate that interface requirements "shall" be met by the applicant. Verification that GE included this change in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.2.1.3-2. GE included this change in Section 8.2.3 of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

GE also committed to revise in a future SSAR amendment the section on interface requirements to include a listing of the regulatory requirements and associated regulatory and industry guidance, which a COL applicant must address in its design scope as part of the COL application. For the offsite system, these should include, as a minimum, the requirements of General Design Criterion (GDC) 17 and 18 of 10 CFR Part 50, Appendix A, as well as the guidelines of IEEE 765-1983. Verification that GE provided the above design information in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.2.1.3-3.

Subsequently, during a November 18, 1992 meeting, GE indicated that SSAR interface requirements were not meant to define the totality of the requirements (including regulatory requirements) which a COL applicant's interfacing design must ultimately meet. The design interface requirements need only include those requirements that are needed by the ABWR design to assure that the design as completed by the COL applicant

meets the in scope design basis requirements in the SSAR. The staff agrees with this assessment. COL applicant design will be required to meet all applicable regulatory requirements and associated regulatory and industry guidance as part of its COL application. The applicable regulatory requirements and associated regulatory and industry guidance are delineated in the code of federal regulations, the NRC SRP, and industry standards and codes. These requirements and guidance do not need to be repeated as interface requirements. DFSER (SECY-92-349) Confirmatory Item 8.2.1.3-3 is therefore resolved. The staff's evaluation of design interface requirements is also addressed in Section 8.2.1.3.1 of this report.

In interface requirement (2), Section 8.2.3.1 of the draft SSAR submittal dated April 3, 1992, indicated that a COL applicant who references the ABWR design is expected to establish the size of the unit auxiliary transformers to ensure a voltage dip of no more than 20 percent during motor starting. It was the staff's understanding, based on discussions with GE, that the sizing of the unit auxiliary transformers is within the ABWR scope of design responsibility. GE committed to revise the SSAR in a future amendment to reflect the above as part of GE's scope. Verification that GE included this change in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.2.1.3-4. GE included this change in Section 8.2.1.2 of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

In interface requirement (4), Section 8.2.3.1 of the draft SSAR submittal dated April 3, 1992, GE indicated that it is acceptable and recommended to normally power all three divisions of the Class 1E ac-distribution system from the normal preferred power source. This interface requirement is not consistent with design commitments that are currently documented in other sections of the SSAR (see Section 8.2.3.5 of this report). GE committed to delete this requirement (specifically, the second sentence of interface requirement (4) in Section 8.2.3.1 of the draft SSAR revision dated April 3, 1992) when the SSAR was revised in the future. Verification that GE included this change in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.2.1.3-5. GE included this change in Section 8.2.3 of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

In interface requirement (5), Section 8.2.3.1 of the draft SSAR submittal dated April 3, 1992, GE indicated that the two offsite circuits will be connected to different transmission systems. It was the staff's understanding, based on discussions with GE, that there is only one transmission system. GE committed to clarify the interface

requirement in a future amendment to indicate that the main and reserve offsite power circuits will be connected to different transmission circuits or lines (rather than systems) and that the transmission circuits or lines will be independent and separate. Verification that GE provided the above design information in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.2.1.3-6. GE included this information in Section 8.2.3(5) of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

Section 8.2.3.1 of GE's draft SSAR submittal dated April 3, 1992, did not include additional interface requirements defining what is required of the COL applicant in order to have independent and separate transmission circuits and switching stations. GE committed to include in a future SSAR amendment explicit interface requirements defining independence and separation of transmission lines and switching stations or switchyards. Verification that GE revised the SSAR to reflect as a minimum the following design commitments in a future amendment was DFSER (SECY-92-349) Confirmatory Item 8.2.1.3-7:

- The two designated preferred power circuits will not have a common take-off structure or use common structures for support.
- The two lines from the transmission network that are designated as the preferred power circuits will be designed to minimize their simultaneous loss as a result of failure of any transmission tower or failure from crossing lines.
- The preferred power circuits originating from the transmission network will be designed to minimize their simultaneous failure as a result of failure of a single breaker, switchyard bus, switchgear bus, or cable.
- System studies will be performed to demonstrate that the preferred power supply will not degrade below a level consistent with the availability goals of the plant as a result of contingencies such as loss of any of the following elements
 - nuclear power generating unit
 - largest generating unit
 - most critical transmission circuit or intertie
 - largest load
- The interconnection between the transmission network and the switchyard will consist of a minimum of two transmission lines that are designated as the preferred power supply circuits. Where more than two lines are

available from the transmission network, any combination of two lines may be designated and used as the preferred power supply circuits, provided that each combination of two circuits meets the offsite system design requirements.

- Switchyard equipment will be designed to adequately withstand stresses from the worst-case faults.
- The physical design of the switchyards will minimize the probability that a single equipment failure will cause the simultaneous or subsequent loss of both preferred power supply circuits.

GE indicated in response to this item that SSAR interface requirements were not meant to define the totality of the requirements (including the requirements needed to assure independence and separation between transmission circuits) which a COL applicant's interfacing design may ultimately have to meet. The staff agrees with this assessment. Under 10 CFR Part 52, the site-specific portion of a COL applicant's design will be required to meet all applicable regulatory requirements and associated regulatory and industry guidance as part of their COL application. The applicable regulatory requirements and associated regulatory and industry guidelines are delineated in the code of federal regulations, the NRC SRP, and the industry standards and codes. These requirements and guidelines, therefore, do not need to be repeated as interface requirements in ABWR SSAR. In addition, GE indicated that the above listed items that were identified in the DFSER (SECY-92-349) to assure independence between offsite circuits would be considered as part of their proposed conceptual design description to be included in the site-specific part of the SAR for a facility referencing the certified design. The staff concludes that interface requirements defining what is required of the COL applicant in order to have independent and separate transmission circuits and switching stations are not required to be included as interface requirements in the ABWR SSAR. DFSER (SECY-92-349) Confirmatory Item 8.2.1.3-7 is therefore resolved. The staff's evaluation of interface requirements relating to capacity, capability, and independence of offsite circuits is in Section 8.2.2 of this report. The staff's evaluation of GE's conceptual design for the offsite system is in Section 8.2.1.3.2 of this report.

In interface requirement (4), Section 8.2.3.1 of the draft SSAR submittal of April 3, 1992, GE indicated that the COL applicant will analyze incoming transmission lines to ensure that their expected availability is as good as assumed in performing the plant's probability risk analysis (PRA). Based on discussions with GE, the staff contended that the assumptions made in performing the plant's PRA

Electric Power Systems

are within the scope of the ABWR design. GE committed to explicitly state as part of this interface requirement, the expected availability of incoming transmission lines, in a future SSAR amendment. Verification that GE provided the above design information in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.2.1.3-8. GE included this information in Section 8.2.3(4) of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

Design commitments documented in the SSAR indicated that the conceptual design of the ABWR offsite preferred power system will include two separate and independent switching stations (or switchyards). Given that there will be two separate and independent switchyards, the interface requirements within individual switchyards presented in GE's draft submittal of April 3, 1992, go beyond industry-recommended practice for offsite preferred circuits. The staff understood that GE will revise the SSAR to describe the parts of the offsite system switchyard not subject to interface requirements specified in Section 8.2.3 of the draft submittal of April 3, 1992. Verification that GE provided the above design information in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.2.1.3-9. GE indicated that they would re-evaluate their proposed interface requirements and conceptual design for the portion of the offsite system that is outside of their scope of supply. Interface requirements and the conceptual design are evaluated in Sections 8.2.1.3.1 and 8.2.1.3.2 of this report respectively. On the basis of these evaluations, this item is resolved.

8.2.1.3.1 Interface Requirements

This section addresses, in part, the staff's evaluation of GE's response to DSER (SECY-91-355) Open Item 24 and DFSER (SECY-92-349) Confirmatory Items 8.2.1.3-3 and 8.2.1.3-9 and complements the resolution of these items.

GE revised their interface requirements included in Section 8.2.3.1 of GE's draft SSAR submittal of April 3, 1992, and moved interface requirements that were included in Section 8.2.4 of SSAR Amendment 21 to Section 8.2.3 of SSAR Amendment 33. In addition, GE provided a representative conceptual design for the portion of the offsite system outside of their scope of supply (see Section 8.2.1.3.2 of this report).

Section 52.47(a)(1)(vii) of 10 CFR dictates that interface requirements be provided (for those portions of the offsite power system design for which GE's application does not seek certification) that are sufficiently detailed to allow completion of the final safety analysis. The staff concludes that GE has provided interface requirements in Section 8.2.3 of SSAR Amendment 33. The design,

therefore, meets the above defined requirements of Section 52.47(a)(1)(vii) of 10 CFR and is acceptable.

The staff's assessment of the adequacy of these interface requirements is addressed in Section 8.2.2 of this report.

8.2.1.3.2 Conceptual Design

This section addresses, in part, the staff's evaluation of GE's response to DSER (SECY-91-355) Open Item 24 and DFSER (SECY-92-349) Confirmatory Items 8.2.1.3-7 and 8.2.1.3-9 and complements the resolution of these items.

GE added a Section 8.2.5 to SSAR Amendment 33 to address their conceptual design for the offsite system outside of its design scope. The conceptual design will consist of two independent offsite circuits from the transmission network to the electrical systems for which GE is seeking certification. These offsite circuits, including their associated I&C circuits, will be independent. Each circuit will have sufficient capacity and capability. Each circuit will have redundant I&C circuits including their ac and dc power supplies. In addition, the total offsite power system will have an expected availability that is as good as the assumptions made in performing the plant PRA.

To aid the staff in its review of the SSAR and to permit assessment of the adequacy of the interface requirements, Section 52.47(a)(1)(ix) of 10 CFR dictates that a representative conceptual design be provided in the ABWR SSAR for those portions of the plant's offsite power system design for which the GE's application does not seek certification. The staff concludes that GE has provided a representative conceptual design described above. The design, therefore, meets the above defined requirements of Section 52.47(a)(1)(xi) of 10 CFR and is acceptable. GE has included the above described conceptual design in Section 8.3.5 of SSAR Amendment 33 which is acceptable.

The staff's assessment of the adequacy of the interface requirements with the aid of the above described conceptual design is addressed in Section 8.2.2 of this report.

Certified Design Material

The certified design material (CDM) for the GE ABWR Design Stage 2 Submittal, transmitted by letter dated April 6, 1992, did not specify the appropriate interface-related ITAAC material for the offsite systems outside the ABWR scope of design. In the ITAAC, (GE) must address this interface area. This was DFSER Open Item 8.2.1.4-1. GE provided a revised set of design

descriptions and ITAAC. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. On the basis of this evaluation, this item is resolved.

8.2.2 Preferred Offsite Circuits Within the ABWR Scope of Design

This section addresses, in part, the staff's evaluation of GE's response to DSER (SECY-91-355) Open Item 24 and DSER (SECY-92-349) Confirmatory Item 8.2.1.3-7 and complements the resolution of these items.

The following portions of the preferred power circuits are within the scope of design of the ABWR standard plant:

- Preferred power circuit from the low-voltage terminals of the main transformer through the three unit auxiliary transformers to the input terminals of each of the three redundant, onsite Class 1E ac-distribution systems.
- Alternate preferred power circuit from the low-voltage terminals of the reserve auxiliary transformer to the input terminals of each of the three redundant, onsite Class 1E ac-distribution systems.

8.2.2.1 Physical Separation (Transformers and Circuits)

1. Physical Separation of Transformers

By its draft submittal dated April 3, 1992, GE revised the SSAR to indicate that the reserve auxiliary transformer will be separated from the unit auxiliary transformers by a minimum distance of 15 m (50 ft) and that each transformer will be provided with an oil collection pit and drain to a safe disposal area. In addition, GE indicated by letter dated April 6, 1992, that the reserve auxiliary transformer and its input feeders will be separated from the main power transformer and its input feeders and from the unit auxiliary transformers by a minimum of 15 m (50 ft). Section 9A.4.6 of SSAR Amendment 14 also indicated that the main, unit auxiliary, and reserve transformers will have oil collection pits and that each of these transformers will have automatic deluge water spray systems. This information was provided in response to DSER SECY-91-355 Open Item 23.

In order to assure that the COL applicant's design for offsite systems does not negate their proposed design for physical separation of transformers, GE indicated that the COL applicant's design will be required (i.e., by interface requirement) to be independent and compatible with the above described design commitments.

GDC 17 of 10 CFR Part 50, Appendix A, requires that provisions be included in the design of the offsite electric power system which will minimize to the extent practicable the likelihood of simultaneous failure of both normal and alternate offsite preferred power circuits under operating and postulated accident and environmental conditions. The staff concludes that a design which meets the above described commitments will provide reasonable assurance that the energy available from a credible failure of one of the offsite circuit transformers (such as, failure by fire or by explosion) will not propagate to the other offsite circuit and cause its failure. The oil collection pit will contain a transformer oil fire at the transformer. The automatic deluge water spray system will minimize the intensity of the heat that may be generated by a transformer fire. In addition, the 15 m (50 ft) of distance to other offsite circuit transformers will dissipate the energy from heat that may be generated by a transformer fire or by missiles that may be generated by transformer explosion.

Consequently, the design meets the above defined requirement of GDC 17 and is acceptable. Verification that GE incorporated the above commitments into a future Chapter 8.0 SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.2.2.1-1. GE has included these commitments in Sections 8.2.2.1(2) and 8.2.3(6) of SSAR Amendment 33, which is acceptable. Therefore, this item is resolved.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.2.2.1-1 related to the design description and the inspections, tests, and analyses, and acceptance criteria (ITAAC) for the physical separation of transformers. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

2. Physical Separation of Circuits

By its draft submittal dated April 3, 1992, in response to DSER Open Item 23, GE revised the SSAR to indicate that separation of the normal and alternate preferred power feeds within the turbine, control, and reactor buildings will be accomplished by floors and walls, except within the switchgear rooms where they must be routed to the same switchgear lineups. Based on discussions, GE further indicated that normal and alternate circuits in the switchgear rooms will be separated to the maximum extent feasible; that is, the circuits will be routed on opposite sides of the room and will be connected to the switchgear lineup on opposite ends. Also, based on discussions, GE indicated that the isolated phase bus duct and/or cables located outside the turbine, control, and reactor buildings

Electric Power Systems

that are affiliated with the normal preferred power circuits will be separated by a minimum of 15 m (50 ft) from the reserve auxiliary transformer. Likewise, the isolated phase bus duct and/or cables located outside the turbines, control, and reactor buildings that are affiliated with the alternate preferred offsite circuit will be separated by a minimum of 15 m (50 ft) from the unit auxiliary and main transformers.

In order to assure that the COL applicant's design for offsite system does not negate their proposed design for physical separation of circuits, GE indicated that the COL applicant's design will be required (i.e., by interface requirement) to be independent and compatible with the above described design commitments.

GDC 17 of 10 CFR Part 50, Appendix A, requires that provisions be included in the design of the offsite electric power system which will minimize to the extent practicable the likelihood of simultaneous failure of both normal and alternate offsite preferred power circuits under operating and postulated accident and environmental conditions.

The staff concludes that a design which meets the above described commitments will provide reasonable assurance that failure of one circuit will not propagate and cause failure of the other offsite circuit. Consequently, the design meets the above defined requirements of GDC 17 of 10 CFR Part 50, Appendix A, and is acceptable. Verification that GE has included appropriate changes in a future SSAR amendment to reflect the above commitments was DFSER (SECY-92-349) Confirmatory Item 8.2.2.1-2. GE has included these commitments in Section 8.2.1.3 and 8.2.3 of SSAR Amendment 33, which is acceptable. Therefore, this item is resolved.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.2.2.1-2 related to the design description and the ITAAC for the physical separation for offsite preferred circuit. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

8.2.2.2 Physical Separation of Instrumentation and Control (I&C) Cables for the Offsite Power System

Based on GE's draft submittal dated April 3, 1992, in response to DSER (SECY-91-355) Open Item 23 and subsequent discussions, GE indicated that instrumentation and control cables that are affiliated with the normal and alternate preferred offsite circuits will be separated as follows:

- The instrumentation and control cables that are affiliated with the normal preferred offsite circuit will be routed in raceways corresponding to the load group of their power source.
- The instrumentation and control cables that are affiliated with the alternate preferred offsite circuit will be routed in dedicated raceways. The alternate preferred offsite instrumentation and control circuit cables will not share raceways with any other cables.
- The separation between the normal and alternate preferred offsite instrumentation and control cables will be the same as the separation between the normal and alternate preferred offsite power circuits (that is: floors, walls, or 15 m (50 ft) of physical separation).

In order to assure that the COL applicant's design for offsite systems does not negate their proposed design for physical separation of I&C circuits, GE indicated that the COL applicant's design will be required (i.e., by interface requirement) to be independent and compatible with the above described design commitments.

GDC 17 of 10 CFR Part 50, Appendix A, requires that electric power from the transmission network to the onsite electric distribution system shall be supplied by two physically independent circuits designed and located so as to minimize to the extent practicable the likelihood of their simultaneous failure under operating and postulated accident and environmental conditions.

The staff concludes that a design which meets the above commitments will provide reasonable assurance that failure of one circuit will not cause the failure of the other circuit. The design, therefore, meets the above defined requirements of GDC 17 of 10 CFR Part 50, Appendix A, and is acceptable. Verification that GE has provided the above commitments in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.2.2.2-1.

GE included the above design commitments in Section 8.2.1.3 of an SSAR markup dated March 31, 1993, and Section 8.2.3 of SSAR Amendment 33 with the exception of I&C circuits located in the control room area and interlock circuitry required to prevent paralleling of the two offsite sources. GE indicated that these circuits are not separated by floors, walls, or 15 m (50 ft) but instead indicated that these circuits will be electrically isolated and will not be routed together in the same raceway. The staff concludes that a design which meets the above described commitments as revised will also provide reasonable assurance that failure of one circuit will not cause failure of the other circuits. Consequently, the revised offsite system design also meets the above defined

requirements of GDC 17 of 10 CFR Part 50, Appendix A, and is acceptable. GE included the additional design provisions in Section 8.2.1.3 of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

During ITAAC development, GE recognized that the offsite system outside of their scope of design was unnecessarily restricted by design requirements stipulated within their scope of design. The requirements did not allow sharing the non-Class 1E dc systems within GE's scope of design between the independent offsite circuits. During ITAAC review, the staff agreed that an offsite system design which utilized the non-Class 1E systems within GE's scope of design and shared these systems between the independent offsite circuits could meet regulatory requirements and could thus be found acceptable. In order to remove this limitation, GE further revised Section 8.2.1.3 of its SSAR markup dated March 31, 1993, to indicate the following:

- a. Instrumentation and control circuits at their dc power sources may be electrically isolated, and not routed together in the same raceway instead of being separated by floors, walls, or 15 m (50 ft).
- b. Instrumentation and control circuits for the independent offsite circuits will not rely on a single common dc power source.

The staff agrees that these revisions will allow utilization of the non-Class 1E dc systems within GE's scope of design and sharing of these dc systems between independent offsite circuits within a COL applicant's offsite system design.

If non-Class 1E dc power systems within GE's scope of design are utilized as the dc source for instrumentation and control in a COL applicant's proposed offsite system design and these dc power sources are shared between the independent offsite circuits, the staff concludes above a design which meets the above described commitments as further revised will also provide reasonable assurance that failure of one circuit will not cause failure of the other circuit.

Consequently, the design meets the above defined requirements of GDC 17 of 10 CFR Part 50, Appendix A, and is acceptable.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.2.2.2-1 related to the design description and the ITAAC for the physical separation of power, instrumentation, and control cables for the offsite preferred power system. The adequacy and

acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

8.2.2.3 Electrical Independence

Based on GE's draft SSAR submittal dated April 3, 1992, in response to DSER (SECY-91-355) Open Item 23 and discussions, GE indicated that there will be no electrical interconnections between the normal and alternate preferred power, instrumentation, and control circuits except where the power circuits connect to common Class 1E and non-Class 1E switchgear lineups. At the common switchgear, one open and one closed circuit breaker will maintain the electrical independence. These circuit breakers will be interlocked so that the closed breaker must be opened before the open breaker can be closed. Transfer from normal to alternate (or alternate to normal) preferred power circuits will be manual. Instrumentation and control circuits (including their power supply) that are affiliated with the normal preferred offsite circuit will be electrically independent from (that is, they will be electrically isolated from or will have no electrical interconnection with), the instrumentation and control circuits (including their power supply) affiliated with the alternate preferred power supply.

In order to assure that the COL applicant's design for offsite systems does not negate their proposed design for electrical independence, GE indicated that the COL applicant's design will be required (i.e., by interface requirement) to be independent and compatible with the above described design commitments.

GDC 17 of 10 CFR Part 50, Appendix A, requires that electric power from the transmission network to the onsite electric distribution system shall be supplied by independent circuits designed and located so as to minimize to the extent practicable the likelihood of their simultaneous failure under operating and postulated accident and environmental conditions.

The staff concludes that a design which meets the above described design commitments will provide reasonable assurance that failure of one circuit will not cause failure of the other circuit. Consequently, the design meets the above defined requirements of GDC 17 of 10 CFR Part 50, Appendix A, and is acceptable.

Verification that GE provided the above design commitments in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.2.2.3-1. GE included these commitments in Sections 8.1.3.1.1.1, 8.2.3, and 8.2.1.3 of SSAR Amendment 33 which is acceptable. Therefore, this item is resolved.

Electric Power Systems

As indicated above in Section 8.2.2.2 of this report, GE further revised Section 8.2.1.3 of SSAR markup dated March 31, 1993, to remove design restrictions within its scope of design which did not allow sharing of the non-Class 1E dc systems within GE's scope of design between the independent offsite circuits.

If non-Class 1E dc power systems within GE's scope of design are utilized as the dc power source for instrumentation and control in a COL applicant's proposed offsite system design and these dc power sources are shared between the independent offsite circuits, the staff concludes that a design which meets the above described commitments as revised will also provide reasonable assurance that failure of one circuit will not cause failure of the other circuit.

Consequently, the design meets the above defined requirements of GDC 17 of 10 CFR Part 50, Appendix A, and is acceptable.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.2.2.3-1 related to the design description and the ITAAC for the electrical independence. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

8.2.2.4 Testing for the Offsite Power System

Based on GE's draft submittal dated April 3, 1992, in response to DSER (SECY-91-355) Open Item 25 and discussions, GE indicated that all systems, equipment, and components that are affiliated with the normal and alternate offsite preferred power circuits within the GE scope of design — except generator breakers — will have the capability to be tested periodically during normal plant operation. Based on this commitment, GE indicated that the design will permit verification of the following offsite power system capabilities:

- The generator breaker can open on demand.
- The instrumentation, control, and protection systems, equipment, and components that are affiliated with the normal and alternate offsite preferred circuits are properly calibrated and perform their required functions.
- All required Class 1E and non-Class 1E loads can be powered from their designated preferred power supply within the capacity and capability margins specified in the SSAR for the offsite system circuits.
- The loss of the offsite preferred power supply can be detected.
- Transfer between preferred power supplies can be accomplished.
- The batteries and chargers that are affiliated with the preferred power system can meet the requirements of their design loads.

In addition, GE indicated that the design of high- and medium-voltage bus ducts and cables provide ready access for regularly inspecting, cleaning, and tightening terminals, and for inspecting and cleaning insulators. The bus duct design also includes provisions for excluding debris and fluids, and for draining condensate.

GDC 18 of 10 CFR Part 50, Appendix A, requires that the offsite electric power systems be designed (1) to permit appropriate periodic inspection and testing of important areas and features (such as wiring, insulation, connections, and switchboards) to assess the continuity of the systems and the condition of their components, (2) the capability to test periodically the operability and functional performance of the components of the systems, and (3) the capability to test periodically the operability of the systems as a whole and (under conditions as close to design as practical) the full operation sequence that brings the systems into operation.

The staff concludes that a design which meets these testability requirements will ensure that the offsite electric power systems, equipment, and components will be designed with the capability to be periodically tested. Consequently, the design meets the above defined requirements of GDC 18 of 10 CFR Part 50, Appendix A, and is acceptable.

Verification that GE provided the above testability requirements in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.2.2.4-1. GE included these testability requirements in Section 8.2.2.1(3) of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

With regard to periodic testing of the systems, equipment, and components, GE indicated the following:

- Periodic verification will ensure that the normal and alternate offsite power circuits are energized and connected to the appropriate Class 1E distribution system division at least once every 12 hours.

- Tests and inspections will be performed at appropriately scheduled intervals for each of the items highlighted above.
- The test and inspection intervals will be established and maintained according to industry recommended practice defined in Section 6.5, "Test Intervals," of IEEE 338-1977.

Verification that GE specified in a future SSAR amendment a statement that the COL applicant must include the above specified periodic tests and inspections in appropriate plant procedures was DFSER (SECY-92-349) COL Action Item 8.2.2.4-1. GE included this action item in Section 8.2.4.1 of SSAR Amendment 32, which is acceptable.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.2.2.4-1 was related to the design description and the ITAAC for testing for the offsite power system. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

8.2.2.5 Generator Breaker

The low-voltage generator breaker must open on a turbine trip to maintain the normal preferred power supply to the safety buses. The generator breaker cannot be tested during normal plant operation without tripping the main turbine generator.

Based on discussions, GE indicated that the opening of the generator breaker to establish the normal offsite preferred power circuit to safety buses will be verified each time the reactor is shutdown (at intervals in accordance with the plant's technical specifications (TS)). In addition, based on information presented in the draft SSAR submittal of April 3, 1992, GE indicated that there are industry published test results showing a reliability number of 0.9967 for the generator breaker.

The draft SSAR submittal of April 3, 1992, also indicated that during all modes of plant operation (including shutdown, refueling, startup, and run), the normal preferred power supply will be connected to two of the three safety buses, and the alternate preferred power supply will be connected to one of the three safety buses. If the normal preferred supply is lost because the generator breaker fails to open, offsite power will still be available immediately through the alternate preferred power supply to one of the three safety buses. It will also be available on a delayed basis (within minutes by manual action from

the control room) to the two other safety buses through the alternate preferred power supply.

GDC 17 of 10 CFR Part 50, Appendix A, requires provisions be included in the design to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated by the nuclear power unit, the loss of power from the transmission network, or the loss of power from the onsite electric power supplies. GDC 18 of 10 CFR Part 50, Appendix A, requires that the offsite electric power systems be designed (1) to permit appropriate periodic inspection and testing of important areas and features (such as wiring, insulation, connections, and switchboards) to assess the continuity of the systems and the condition of their components, (2) the capability to test periodically the operability and functional performance of the components of the systems, and (3) the capability to test periodically the operability of the systems as a whole and (under conditions as close to design as practical) the full operation sequence that brings the systems into operation.

The staff concludes that a design which meets the above described provisions, will minimize to the extent feasible the likelihood of simultaneous failure of both normal and alternate offsite preferred power circuits and will be testable. Consequently, the design meets the above defined requirements of GDC 17 and 18 of 10 CFR Part 50, Appendix A, and is acceptable.

Verification that GE specified in a future SSAR amendment a statement that the COL applicant must include surveillance requirements for generator breakers and the above design provisions in appropriate plant procedures was DFSER (SECY-92-349) COL Action Item 8.2.2.5-1. GE included this action item in Sections 8.2.4.1(7) and 8.3.4.9 of SSAR Amendment 32, which is acceptable.

8.2.2.6 Capacity and Capability of the Offsite Power System

Based on discussions with GE and information presented in the April 3, 1992, draft submittal of SSAR Chapter 8.0, in response to DSER (SECY-91-355) Open Item 26, GE indicated that the offsite power system will be designed to provide the following capacity and capabilities:

- Each circuit of the preferred power supply will be designed to provide sufficient capacity and capability to power equipment required to ensure that
 - fuel design limits and design conditions of the reactor coolant pressure boundary will not be

Electric Power Systems

exceeded as a result of anticipated operational occurrences

- in the event of plant design-basis accidents, the core will be cooled, and containment integrity and other vital functions will be maintained
- When used for normal operation, each preferred power supply will be sized to supply the maximum expected coincident Class 1E and non-Class 1E loads.
- The secondary winding of the reserve auxiliary transformer, which supplies the Class 1E load groups, will have an oil/air rating greater than or equal to the combined load of the three Class 1E load groups.
- The normal and alternate offsite preferred power circuits will be designed with sufficient capacity and capability to limit variations of the operating voltage of the onsite power distribution system to a range appropriate to ensure
 - normal and safe steady-state operation of all plant loads
 - starting and acceleration of the limiting drive system with the remainder of the loads in service
 - reliable operation of the control and protection systems under conditions of degraded voltage

Specifically, when measured at the load terminals, the voltage variation at any voltage level will not exceed the following limits

- plus or minus 10 percent of the load-rated voltage during all modes of steady-state operation
- minus 20 percent of the motor-rated voltage during motor starting
- Voltage levels at the low-voltage terminals of the auxiliary and reserve transformers will be analyzed to determine the maximum and minimum load conditions that are expected throughout the anticipated range of voltage variations of the offsite transmission system and the main generator. Separate analyses will be performed for each possible circuit configuration of the offsite power supply system.
- During their operation, normal and alternate preferred power circuits are subject to environmental conditions (such as, wind, ice, snow, lightning, temperature variations, or flood). These circuits will be designed in accordance with industry-recommended practice in

order to minimize the likelihood that they will fail while operating under the environmental conditions to which they are subject.

- During their operation, normal and alternate preferred power circuits can be subjected to the transmission system's steady-state and transient conditions (such as switching and lightning surges, maximum and minimum voltage ranges for heavy and light load conditions, frequency variation, or stability limits). The preferred power circuits will be designed such that these conditions will not subject the onsite Class 1E systems, equipment, and components to conditions that are beyond the limits for which they are designed and qualified. These design considerations apply to all onsite Class 1E loads and systems that use the services of the preferred power supply during startup, normal operation, safe shutdown, accident, and post-accident operation. The staff's evaluation of this item is addressed in Section 8.2.3.8 of this report.
- Performance and operating characteristics of the normal and alternate preferred power circuits will be required to meet operability and design-basis requirements. These requirements include but are not limited to; the ability to withstand short-circuits, equipment capacity, voltage and frequency transient response, voltage regulation limits, step load capability, coordination of protective relaying, and grounding.
- The generator circuit breaker will be designed to withstand the maximum root mean squared and crest momentary currents. Further, the breaker will be designed to interrupt the maximum asymmetrical and symmetrical currents determined to be produced by a three-phase fault at the location that results in the maximum fault currents.
- The main step-up transformers and the unit auxiliary and reserve transformers will be designed and constructed to withstand the mechanical and thermal stresses produced by external short circuits. In addition, these transformers will meet the corresponding requirements of the latest revisions of ANSI C57.12.00, "General Requirements for Liquid-Immersed Distribution, Power, and Regulating Transformers."
- Circuit breakers and disconnecting switches will be sized and designed according to the latest revision of ANSI C37.06, "Preferred Ratings and Related Capabilities for ac High-Voltage Circuit Breakers Rated on a Symmetrical Current Basis."

In order to assure that the COL applicant's design for offsite systems does not negate their proposed design for capacity and capability of the offsite system, GE indicated that the COL applicant's design will be required (by interface requirements) to meet the following:

- Voltage variations of the offsite transmission network during steady state operation will not cause voltage variations at the loads of more than plus or minus 10 percent of the loads nominal ratings.
- The normal steady state frequency of the offsite transmission network will be within plus or minus 2 hertz of 60 hertz during recoverable periods of system instability.
- The offsite transmission circuits from the transmission network through and including the main step-up power and reserve auxiliary transformers will be sized to supply their load requirements, during all design operating modes, of their respective Class 1E divisions and non-Class 1E load groups.
- The impedance of the main step-up power and reserve transformers will be compatible with the interrupting capability of the plant's circuit interrupting devices.
- Instrumentation and control system loads will be compatible with the capacity and capability design requirements of dc systems within the ABWR standard plant scope.

GDC 17 of 10 CFR Part 50, Appendix A, requires that the offsite electric power system have sufficient capacity and capability to permit safety systems to perform their required safety function.

The staff concludes that a design which meets the above described commitments and characteristics will have sufficient capacity and capability to ensure that:

- Specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary will not be exceeded as a result of anticipated operational occurrences.
- The core will be cooled, and containment integrity and other vital functions will be maintained in the event of postulated accidents.

The design, therefore, meets the above defined requirements of GDC 17 of 10 CFR Part 50, Appendix A, and is acceptable.

Verification that GE provided the above noted design commitments and characteristics in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.2.2.6-1. GE included this information in Sections 8.1.2.1, 8.2.1.2, 8.2.2.1(2), and 8.2.2.1(8) of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved. (Note: A portion of this confirmatory item is discussed in Section 8.2.3.8 of this report.)

In response to DFSER (SECY-92-349) COL Action Item 8.2.2.6-1, GE clarified operational restrictions as follows for the offsite system:

- Operational restrictions will require the reserve auxiliary transformer to supply one of three Class 1E load groups during normal plant operation.
- Operational restrictions will assure that the forced oil/air ratings of the reserve auxiliary transformer, or any unit auxiliary transformer, will not be exceeded under any operating mode.
- Continued plant operation will be appropriately limited when one of the three unit auxiliary transformers is inoperable (that is, when one of the three safety buses will not have access to both normal and preferred offsite circuits).
- Continued plant operation will be appropriately limited when the reserve auxiliary transformer is inoperable.

GE included the above listed aspects of this action item in Sections 8.2.4.2, 8.2.4.5, and 8.3.4.9 of SSAR Amendment 32, which is acceptable.

As indicated above in Section 8.2.2.2 of this report, GE further revised Section 8.2.1.3 of SSAR markup dated March 31, 1993, to remove design restrictions within its scope of design which did not allow sharing of the non-Class 1E dc systems within GE's scope of design between the independent offsite circuits.

If non-Class 1E dc power systems within GE's scope of design are utilized as the dc power source for instrumentation and control in a COL applicant's proposed offsite system design and if these dc power sources are (or are not) shared between the independent offsite circuits, the staff concludes that a design which meets the above design commitments and characteristics as revised will also have sufficient capacity and capability to ensure that:

- Specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary will not be exceeded as a result of anticipated operational occurrences.

Electric Power Systems

- The core will be cooled, and containment integrity and other vital functions will be maintained in the event of postulated accidents.

The design, therefore, meets the above defined requirements of GDC 17 of 10 CFR Part 50, Appendix A, and is acceptable.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.2.2.6-1 was related to the design description and the ITAAC for the capacity and capability of the offsite power system. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved. (Note: This open item is also addressed in Section 8.2.3.8 of this report).

8.2.2.7 Grounding and Lightning Protection

This section was moved to section 8.2.3.6 of this report.

8.2.2.8 Compliance With IEEE 765

Compliance with the guidelines of IEEE 765-1983 is addressed in Section 8.3.1.1 of this report.

8.2.2.9 Alternate Source of Power for Non-Safety Loads (NRC Policy Issue SECY-91-078, Section II.B of Enclosure 1 to SECY-93-087)

In SECY-91-078, the staff recommended that the Commission approve its position that an evolutionary plant design should include an alternate power source to the non-safety loads unless it can be demonstrated that the design margins will result in transients for a loss of non-safety power event that are no more severe than those associated with the turbine-trip-only event in current existing plant designs. In its August 15, 1991, SRM, the Commission approved the staff's position. The staff in NUREG-1242, "NRC Review of Electric Power Research Institute's Advanced Light Water Reactor Utility Requirements Document," Volume 2 (August 1992), clarified the intent of this position. Further, the staff has evaluated the electrical design for the ABWR relative to the provision of an alternate source of power for non-safety loads against the following proposed applicable regulation, which is based on the position included in NUREG-1242: "The electric power system of the standard design must include an alternate power source that is provided to a sufficient string of non-safety loads so that forced circulation could be maintained, and the operator has available the complement of non-safety equipment that would most facilitate the ability to bring the plant to a stable shutdown condition, following a loss of the normal power supply and plant trip."

GE revised the SSAR to indicate that the ABWR reserve auxiliary transformer provides the required alternate power source. The staff concluded that the ABWR design meets the requirements of this applicable regulation, and is acceptable. GE included this information in Section 8.3.1.2(4)(b) of SSAR Amendment 32.

8.2.3 Independence Between Offsite and Onsite Power Systems, Equipment, and Components

The preferred power supply furnishes electric power from the offsite system's transmission network (a common source of electric power) to redundant, onsite Class 1E systems, equipment, components, and loads. This common source of electric power may be used during all modes of plant operation to supply power to redundant Class 1E load groups. Because redundant load groups are powered from a common power source, they can be subjected to conditions which may cause their common failure due to single events or failures of this single source of electric power. This section of the staff's evaluation addresses ABWR design provisions for minimizing the probability of: (1) common mode failure of redundant onsite Class 1E systems due to single events or failures of this common source of electric power; and, (2) failure of the onsite system from causing loss of the offsite system in accordance with the requirements of GDC 17 of 10 CFR Part 50, Appendix A. The staff's DSER (SECY-91-355) addressed this part of the design in Open Item 28.

The following sections discuss the areas addressed in the staff's evaluation of ABWR design provisions intended to ensure an adequate level of independence between offsite and onsite systems.

8.2.3.1 Independence Between Offsite Circuits and Onsite Class 1E dc Systems

DC control, protection, and instrumentation power for offsite circuits, (originally proposed for the ABWR design), were derived from the Class 1E dc system through dc to dc converters that GE considered isolation devices. By the draft SSAR revision of April 3, 1992, GE eliminated all electrical interconnections between the offsite control, protection, and instrumentation circuits and the onsite Class 1E dc systems, equipment, and components.

GE indicated that the offsite system circuits will derive their control, protection, and instrumentation power from a non-Class 1E dc system that is independent of the onsite Class 1E dc system. GDC 17 of 10 CFR Part 50, Appendix A, requires provisions be included in the design to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident

with, the loss of power generated by the nuclear power unit, the loss of power from the transmission network, or the loss power from the onsite electric power supplies. The staff concludes that a design which meets the above described design commitment (1) will provide reasonable assurance that failure in the offsite system will neither challenge nor possibly cause the loss of onsite Class 1E dc systems, (2) will provide reasonable assurance that any single failure of a Class 1E dc system or its component parts will not cause loss of offsite power, and (3) will provide reasonable assurance that common-cause failure will not occur between offsite and onsite power sources that are affiliated with a single load group. Therefore, the design meets the above defined requirements of GDC 17 of 10 CFR Part 50, Appendix A, and is acceptable.

Verification that GE provided the above design commitments in a future SSAR amendment was DFSE (SECY-92-349) Confirmatory Item 8.2.3.1-1. GE has included these commitments in Section 8.2.1.2 of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

Certified Design Material

DFSE (SECY-92-349) Open Item 8.2.3.1-1 was related to the design description and the ITAAC for the electrical independence between offsite circuits and onsite Class 1E DC system. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

8.2.3.2 Independence During Loss of or Degraded Offsite Voltage

The ABWR design incorporates two distinct levels of protection to ensure the independence of offsite and onsite systems during loss of offsite voltage or degraded offsite voltage conditions:

(1) Loss of Offsite Voltage (First Level of Protection)

In the draft SSAR revision of April 3, 1992, Section 8.3.1.1.7(1), GE indicated that the onsite Class 1E systems will be normally energized from the offsite normal and alternate preferred power system. Should the voltage on the Class 1E bus decay to less than 70 percent of its nominal rated value for a predetermined time, a bus transfer will be initiated. As a result of this transfer, the onsite Class 1E buses will be transferred so that they are powered by the onsite standby diesel generators (rather than the offsite normal and alternate preferred power systems). Upon initiation of a bus

transfer, a signal will be generated to open the offsite supply breaker to the Class 1E bus.

In the draft SSAR revision of April 3, 1992, the first paragraph of Section 8.3.1.1.7 indicated that the time delay for bus transfer initiation will change (be reduced) from 3 to 0.4 seconds if a loss-of-coolant accident (LOCA) signal is present when a loss of preferred power (LOPP) occurs. (The 3 and 0.4 second time delay setpoints for initiation of bus transfer are considered to be approximations of the actual setpoint that will be in effect during plant operation. The actual setpoints will be established as part of an overall system voltage and load analysis.) Discussions with GE confirmed that the proposed ABWR design will include a time delay relay to establish the predetermined time before bus transfer initiation. When only a LOPP signal is present, the time delay relay will be set for 3 seconds. When both LOPP and LOCA signals are present, the time delay relay will be set for 0.4 seconds. The purpose of the time delay is to reduce unnecessary transfer from offsite to onsite sources during offsite power system transients. If voltage on the offsite system drops below 70 percent for less than 3 seconds (or 0.4 seconds when both LOCA and a LOPP signals are present), both the LOPP and time delay relay will reset after voltage recovery and transfer will not be initiated.

(2) Degraded Offsite Voltage (Second Level of Protection)

In draft SSAR revision of April 3, 1992, Section 8.3.1.1.7(8), GE indicated that when the bus voltage degrades to 90 percent or less of its rated value and remains degraded throughout a time delay, an undervoltage condition will be annunciated in the control room. Simultaneously, a 5-minute timer will be started, allowing the operator to take necessary corrective action. (The 5 minute time delay setpoints for initiation of bus transfer are considered to be approximations of the actual setpoint that will be in effect during plant operation. The actual setpoints will be established as part of an overall system voltage and load analysis.) After 5 minutes, the feeder breaker affected by the degraded voltage will be tripped. Should a LOCA occur during the 5 minute time delay, the affected feeder breaker will be tripped immediately.

In draft SSAR submittal of April 3, 1992, Section 8.3.1.2.1(3)(d), GE indicated that the design for degraded offsite voltage will meet the

Electric Power Systems

guidelines of Branch Technical Position, Power Systems Branch 1.

GE indicated that the ABWR electric system design will comply with the guidelines of IEEE 308-1980 and IEEE 603-1980, as specified by Section 8.3.1.2.1(2)(c) of the April 3, 1992, draft SSAR revision and Section 1.8.2 (Tables 1.8-20 and 1.8-21) of SSAR Amendment 17. Because both levels of protection are required to support safety-related systems, the staff concluded that GE's commitment to IEEE standards indicates that the electric system design will meet the requirements of Section 5.2 of IEEE 308-1980. Section 5.2 of IEEE 308-1980 requires that components, equipment, or systems required to provide some protective action, such as containment integrity protection, or utilized to provide isolation protection are covered by all safety system requirements which are defined in IEEE 603-1980. The systems described above protect safety-related electrical systems from degraded or loss of voltage conditions. Thus, they provide some protective action and are covered by all safety system requirements defined in IEEE 603-1980. The staff, therefore, concluded that GE's commitment to IEEE standards indicates that systems, equipment, and components included in the design for both the first and second levels of voltage protection will meet all requirements of IEEE 603-1980.

GE's expressed commitment to IEEE Standards also applies to safety-related equipment (including the reactor trip system, engineered safety features, auxiliary supporting features, and other auxiliary features equipment) that requires ac power from the offsite system to perform safety functions. Specifically, GE indicated that such safety-related equipment will be designed and qualified (by type test, previous operating experience, analysis, or any combination of these three methods) to be capable of performing their required safety functions before, during, and after the following design basis operating conditions:

- voltages at the load at either +10 percent or -10 percent of the nominal voltage rating
- for 5 minutes with voltages at the load at 70 percent of the nominal voltage rating
- for 3 seconds with voltages at the load below 70 percent (e.g., for 3 seconds at 35 percent) of the nominal voltage rating

GDC 17 of 10 CFR Part 50, Appendix A, requires provisions be included in the design to minimize the

probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated by the nuclear power unit, the loss of power from the transmission network, or the loss of power from the onsite electric power supplies. The staff concludes that a design which meets the above commitments will provide reasonable assurance that the common failure of both offsite and onsite systems will not occur as a result of loss of or degraded voltage conditions. The design, therefore, meets the above defined requirements of GDC 17 of 10 CFR Part 50, Appendix A and is acceptable.

Verification that GE provided the above design commitments in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.2.3.2-1. GE included this information in Section 8.3.1.1.7 of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

GE also indicated that both first- and second-level voltage protection will be tested periodically. Verification that GE included a COL action item to include these periodic tests in appropriate plant procedures was DFSER (SECY-92-349) COL Action Item 8.2.3.2-1. GE included this action item in Section 8.3.4.20 of SSAR Amendment 32, which is acceptable.

During the process of verifying inclusion in the SSAR of the above design information and periodic tests, the staff noted and expressed the concern that the diesel generator may be overloaded if there is a LOCA with a delayed LOPP. Section 8.3.1.1.7(3) of SSAR Amendment 21, LOPP following LOCA, indicates that all loads that have been connected to the Class 1E bus and are operating from the offsite power supply in response to the LOCA will be connected in one block to the standby diesel generator (operating at no load due to the LOCA) if there is a LOPP. The diesel generator may not have sufficient capacity to supply, in one block, the loads which may be operating on the Class 1E bus at the time of the LOPP. Because this item involved the capacity of the diesel generator, it has been moved to Section 8.3.8.4 of this report which addresses the capacity of the diesel generator.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.2.3.2-1 related to the design description and the ITAAC for the electrical independence during loss of, or degraded, offsite voltage. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

8.2.3.3 Independence During Parallel Operation of the Offsite and Onsite Systems During Periodic Load Tests of the Diesel Generator

(1) LOCA During Parallel Operation

Section 8.3.1.1.7(5) of SSAR Amendment 4 states that if a LOCA occurs when the diesel generator is being operated in parallel with the preferred power source during testing, and the test is being conducted from the local control panel, control must be returned to the main control room or the test operator must trip the diesel generator breaker. GE subsequently revised Section 8.3.1.1.7(4) of SSAR Amendment 4 in Section 8.3.1.1.7(5) of the draft SSAR revision dated April 3, 1992 (see the response to Question 435.19). Through that revision, GE changed the design commitment to indicate that if a LOCA occurs when the diesel generator is being operated in parallel with the offsite system, the diesel generator will automatically be disconnected from the 6.9 kV emergency bus, regardless of whether the test is being conducted from the local control panel or the main control room.

In addition, Section 8.3.1.1.8.8 of the draft SSAR revision dated April 3, 1992, indicated that the ABWR design will include interlocks to the LOCA and LOPP sensing circuits, in order to terminate parallel operation and cause the diesel generator to automatically revert to its standby mode if either a LOCA or a LOPP signal appears during a test. GE further indicated that the interlock design will have the capability to be tested periodically.

The ABWR design will include provisions for automatic switchover from system test mode to operating mode in case of either an accident signal or a loss of preferred offsite power signal, regardless of whether the test is being conducted from the local control panel or the main control room. GDC 17 of 10 CFR Part 50, Appendix A, requires provisions be included in the design to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated by the nuclear power unit, the loss of power from the transmission network, or the loss of power from the onsite electric power supplies. The staff concludes that a design which meets the above described commitments will provide reasonable assurance that loss of electric power from the offsite and onsite systems will not occur due to loss of power from the nuclear power unit due to a LOCA. The design, therefore, meets the above defined requirements of GDC 17 of 10 CFR Part 50, Appendix A, and is acceptable.

Verification that GE provided the above design commitments in a future SSAR amendment was DFSE (SECY-92-349) Confirmatory Item 8.2.3.3-1. GE has

included these commitments in Sections 8.3.1.1.7(5), 8.3.1.1.7(6), and 8.3.1.1.8.8 of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

GE also indicated that the interlock design (which terminates parallel operation and causes the diesel generator to automatically revert to its standby mode) will periodically be tested. Verification that GE would include a COL action item to include this periodic test in appropriate plant procedures was DFSE (SECY-92-349) COL Action Item 8.2.3.3-1. GE included this action item in Section 8.3.4.21 of SSAR Amendment 32, which is acceptable.

Certified Design Material

DFSE (SECY-92-349) Open Item 8.2.3.3-1 was related to the design description and the ITAAC for LOCA during parallel operation. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

(2) LOPP During Parallel Operation

In Section 8.3.1.1.7(6) of the draft SSAR revision dated April 3, 1992, GE indicated that the diesel generator circuit breaker will automatically trip on overcurrent if the offsite power supply is lost during the diesel generator paralleling test. In addition, Section 8.3.1.1.8.8 of that revision indicated that interlocks to the LOPP sensing circuits will terminate a parallel operation test, causing the diesel generator to automatically revert to its standby mode if a LOPP signal appears during a test.

When a standby power supply is being operated in parallel with the preferred power supply, Section 5.1.4.3 of IEEE 741-1986 requires that protection be provided to separate the two supplies if either degrades to an unacceptable level. However, this protection shall neither lock out nor prevent the availability of the power supply that is not degraded. In addition, Section 6.2.4.6.3 of IEEE 308-1992 requires provisions to detect a LOPP during testing, when the standby generator is connected to the offsite power source.

The staff concluded that a design complying with these industry-recommended IEEE practices will minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power from the transmission network. This design meets the above defined requirements of GDC 17 of 10 CFR Part 50, Appendix A, and is acceptable.

Electric Power Systems

Based on GE's design commitment that interlocks to the LOPP sensing circuits will be included in the ABWR design to terminate a parallel operation test and cause the diesel generator to automatically revert to its standby mode if LOPP signal appears during a test, the staff concluded that the ABWR design meets the above described industry-recommended interlock practice.

The staff concludes that a design that meets the above described design commitments will provide reasonable assurance that electric power from the onsite diesel generator supplies will not be lost due to loss of the offsite transmission network. The design, therefore, meets the above defined requirements of GDC 17 of 10 CFR Part 50, Appendix A, and is acceptable.

Verification that GE provided the above design commitments in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.2.3.3-2. GE has included these commitments in Section 8.3.1.1.7(6) of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.2.3.3-2 was related to the design description and the ITAAC for LOPP during parallel operation. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

(3) Duration of Parallel Operation

Table 1.8-21 of SSAR Amendment 17 and Section 1.8.2 of SSAR Amendment 12 indicated that the ABWR design complies with IEEE 308-1980. Based on this statement of compliance, GE indicated that the ABWR design will satisfy Section 6.1.3 of IEEE 308-1980, which requires that the design minimize the duration of the connection between the preferred and standby power supplies. In addition, Section 8.3.1.1.8.1 of the draft SSAR revision dated April 3, 1992, indicated that the ABWR design requires that each diesel generator set be operated independently of the other sets and be connected to the utility power system only by manual control during testing or for bus transfer.

The staff concludes that a design which meets these commitments will minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated by the nuclear power unit, the loss of power from the transmission network, or the loss of power from the onsite electric power supplies. Consequently, this design meets

the above defined requirements of GDC 17 of 10 CFR Part 50, Appendix A, and is acceptable.

Verification that GE included in a future SSAR amendment an action item that the COL applicant include the above commitments in appropriate plant procedures was DFSER (SECY-92-349) COL Action Item 8.2.3.3-2. GE included this action item in Section 8.3.4.21 of SSAR Amendment 32, which is acceptable.

(4) Diesel Generator Protective Relaying with the Diesel Generator Operating in Parallel with the Offsite System

Section 8.3.1.1.6.4 of the draft SSAR revision dated April 3, 1992, indicated that protective relaying of the diesel generator (generator differential, engine overspeed, low jacket water pressure, loss of excitation, anti-motoring (reverse power) overcurrent voltage restraint, high jacket water temperature, and low lube oil pressure) will be used to protect the machine when it is operated in parallel with the normal power system during periodic tests. In addition, Section 8.3.1.1.6.2 of the draft SSAR revision dated April 3, 1992, indicated that each diesel generator will be high-resistance grounded to maximize availability.

Section 8.3.1.2.1(2)(c) of the draft SSAR revision dated April 3, 1992, and Section 1.8.2 (Tables 1.8-20 and 1.8-21) of SSAR Amendment 17 indicated that the ABWR electric system design will comply with the requirements of IEEE 308-1980 and IEEE 603-1980. Through this commitment to these IEEE standards, GE indicated that the electric system design will satisfy the requirements of Section 5.2 of IEEE 308-1980, and that systems, equipment, and components included in the design for protective relaying will satisfy all requirements of IEEE 603-1980. Because protective relaying is required to minimize the likelihood of simultaneous loss of both offsite and onsite sources during testing, these components are required to satisfy all requirements of IEEE 603-1980 during parallel operation of offsite and onsite power supplies.

The staff concludes that a design which meets the above described commitments will provide additional assurance that there will not be a common failure between onsite and offsite power supplies during testing. The design, therefore, meets the above defined requirements of GDC 17 of 10 CFR Part 50, Appendix A, and is acceptable.

Verification that GE provided the above design commitments in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.2.3.3-3. GE included this information in Sections 8.3.1.1.6.2 and

8.3.1.1.6.4 of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

GE also indicated that protective relaying of the diesel generator will be tested periodically. Verification that GE included in a future SSAR amendment a statement that the COL applicant will include these periodic tests in appropriate plant procedures was DFSER (SECY-92-349) COL Action Item 8.2.3.3-3. GE included this action item in Section 8.3.4.22 of SSAR Amendment 32, which is acceptable.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.2.3.3-3 was related to the design description and the ITAAC for the diesel generator protective relaying when the diesel generator is operating in parallel with the offsite system. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

(5) Synchronizing Interlocks

Based on discussions, GE indicated that the ABWR design will meet the guidelines of Section 5.1.4.2 of IEEE 741-1986 which requires synchronizing interlocks to prevent incorrect synchronization whenever a standby power source is required to operate in parallel with the preferred power supply. GE also indicated that the synchronizing interlocks will have the capability to be tested periodically.

The staff concludes that a design complying with the above described commitments will reduce the likelihood of simultaneous loss of both offsite and onsite power supplies as a result of synchronization. The design, therefore, meets the above defined requirements of GDC 17 of 10 CFR Part 50, Appendix A, and is acceptable.

Verification that GE provided the above design commitments in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.2.3.3-3. GE included this information in Section 8.3.1.1.6.4 of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

GE also indicated that synchronizing interlocks will be tested periodically. Verification that GE included in a future SSAR amendment a statement that the COL applicant will include these periodic tests in appropriate plant procedures was DFSER (SECY-92-349) COL Action Item 8.2.3.3-4. GE included this action item in Section 8.3.4.23 of SSAR Amendment 32, which is acceptable.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.2.3.3-4 was related to the design description and the ITAAC for the synchronizing interlocks. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

8.2.3.4 Independence of Safety Systems During Operation or Failure of Non-Class 1E Loads (NRC Policy Issue SECY-91-078, Section II.B of Enclosure 1 to SECY-93-087)

The ABWR design must minimize the effects that operation or failure of non-Class 1E loads may have on Class 1E systems. To achieve this design objective, the staff took the position that the Class 1E system should be connected directly to a winding of the offsite power system's transformers that is separate from the winding that feeds the non-Class 1E loads. In other words, Class 1E and non-Class 1E loads should not be powered from the same transformer winding.

In SECY-91-078, the staff recommended that the Commission approve this position for evolutionary plant designs. The evolutionary plant design should include at least one offsite circuit to each redundant safety division supplied directly from one of the offsite power sources with no intervening non-safety buses in such a manner that the offsite source can power the safety buses upon a failure of any non-safety bus. In its August 15, 1991, SRM, the Commission approved the staff's position.

Therefore, the staff's proposed applicable regulation for offsite power sources is as follows:

The electric power system of standard plant design must include at least one offsite circuit to each redundant safety division supplied directly from one of the offsite power sources with no intervening non-safety buses in such a manner that the offsite source can power the safety buses upon a failure of any non-safety bus.

The initial design proposed by GE for the ABWR offsite preferred power system satisfied this staff position. However, by the draft SSAR revision dated April 3, 1992, GE changed the design to use a single transformer winding to supply power to both Class 1E and non-Class 1E loads. As a result, the staff became concerned that operation or failure of non-Class 1E loads could adversely affect operation and thus the independence of Class 1E systems.

Electric Power Systems

Based on discussions, GE indicated that any single failure of a non-Class 1E load or load group will affect only one of the three Class 1E redundant load groups. The ABWR design will consist of three, non-Class 1E load groups, as well as three Class 1E divisional load groups and three transformers. Each of the three non-Class 1E load groups will be affiliated with only one of the three Class 1E divisional load groups by being powered from the same offsite power system transformer winding. Thus, failure of any one of three transformers due to failure of non-Class 1E loads or load groups can affect only one of the three Class 1E divisional load groups. In addition, the alternate offsite preferred circuit design (which does not include provision for powering of non-Class 1E loads from the same transformer winding as Class 1E loads) and operating procedures (which require that one of the three divisional buses be fed by the alternate offsite power source during normal operation) will also provide assurance that the three Class 1E divisional load groups will not be subjected to common abnormal conditions due to failure of non-Class 1E loads or load groups.

In addition, GE indicated during discussions with the staff that the ABWR design will include provisions to limit the harmonic effect on the Class 1E divisional load group power supply to less than 5 percent for operation or failure of reactor internal pumps (RIPs) or other non-Class 1E loads.

GDC 17 of 10 CFR Part 50, Appendix A, requires provisions be included in the design to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated by the nuclear power unit, the loss of power from the transmission network, or the loss of power from the onsite electric power supplies. The staff concludes that a design which meets the above described commitments will minimize the effects of non-Class 1E system operation/failure causing loss of electric power to safety systems or failure of Class 1E safety systems and loads. The design, therefore, meets the above defined requirements of GDC 17 of 10 CFR Part 50, Appendix A, and the staff's proposed applicable regulation for offsite power sources is acceptable.

Verification that GE provided the above design commitments in a future SSAR amendment was DFSE (SECY-92-349) Confirmatory Item 8.2.3.4-1. GE included this information on Figure 8.3-1 and in Sections 8.3.1.0.1 and 8.3.4.9 of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

Certified Design Material

DFSE (SECY-92-349) Open Item 8.2.3.4-1 was related to the design description and the ITAAC for the independence of Class 1E systems from the influences of non-Class 1E loads. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

8.2.3.5 Physical Separation Between Offsite and Onsite Class 1E Circuits

Based on discussions with GE and information presented in Section 8.2.1.3 of the April 3, 1992, draft revision to the SSAR, the ABWR design is such that the offsite circuits will be physically separated from any Class 1E systems, equipment, components, cables, or loads by floors or walls up to the point where the offsite circuits enter the reactor building. From the point where the alternate preferred circuit enters the Division II side of the reactor building to the Class 1E switchgear rooms, and from the point where the normal preferred circuit enters the Divisions I and III side of the reactor building to the Class 1E switchgear rooms, GE has indicated (by their commitment to IEEE 384), that the offsite circuits would be physically separated from circuits of the Class 1E systems by a minimum physical separation distance of 0.9 m (3 ft) horizontal and by 1.5 m (5 ft) vertical. In addition, GE indicated that safety systems (for example, rotating equipment with potential for being a missile hazard) whose failure could potentially affect the operation of an offsite circuit will not be located in the same rooms with the normal or alternate offsite circuits, or barriers will be installed to preclude possible interaction between offsite and onsite systems.

GDC 17 of 10 CFR Part 50, Appendix A, requires provisions be included in the design to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated by the nuclear power unit, the loss of power from the transmission network, or the loss of power from the onsite electric power supplies. The staff concludes that a design which meet the above described commitments will provide reasonable assurance that failure of offsite circuits will not cause loss of onsite circuits and failure of onsite safety-related equipment will not cause loss of offsite circuits. The design, therefore, meets the above defined requirements of GDC 17 of 10 CFR Part 50, Appendix A, and the staff's proposed applicable regulation for offsite power sources, and is acceptable.

Verification that GE provided the above design commitments in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.2.3.4-2. GE has included these commitments in Sections 8.2.1.3 and 8.3.3.1 of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

In addition, Figure 8.3.1 of the draft SSAR revision dated April 3, 1992, indicated that the offsite power connection from the reserve auxiliary transformer is normally supplied through the Division II Class 1E equipment areas to the Division III load group. Similarly, GE indicated that the offsite connection from the unit auxiliary transformers is normally supplied through the Divisions I and III Class 1E equipment areas to the Divisions I and II load groups. To further minimize the likelihood of interaction between the offsite and onsite systems during operation, the staff contended that the normal configuration for the connection of the offsite circuit to the onsite Class 1E distribution system should be configured with the reserve auxiliary transformer normally connected to the Division II load group and the unit auxiliary transformers normally connected to the Divisions I and III load groups. This was DFSER (SECY-92-349) Open Item 8.2.3.4-2.

This is an operation rather than design issue. Therefore, in response to this item, GE specified that the COL applicant would establish the preferred configuration of offsite circuits for normal operation based on the reliability/stability of offsite circuits, the Class 1E bus loads, and the separation of the offsite feeds as they pass through the divisional areas. Based on this requirement, the staff concludes that the operational configuration of offsite circuits will minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated by the nuclear power unit, the loss of power from the transmission network, or the loss of power from the onsite electric power supplies. GE has included this COL action item in Section 8.3.4.9 of SSAR Amendment 32, which is acceptable. Therefore, DFSER Open Item 8.2.3.4-2 is resolved.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.2.3.5-2 was related to the design description and the ITAAC for the physical separation between offsite and onsite class 1E circuits. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

8.2.3.6 Grounding and Lightning Protection

This section addresses the staff's evaluation of GE's response to DFSER (SECY-92-349) Confirmatory Item 8.2.2.7-1 and Open Item 8.2.2.7-1.

The ACRS, in an April 13, 1992, letter to the staff discussed a concern that SSAR Chapter 8 did not discuss any requirements or design considerations for station grounding. In response GE has committed to meet the following Electric Power Research Institute (EPRI) plant grounding guidelines:

- A station grounding grid, consisting of bare copper cables, will be provided that will limit step-and-touch potentials to safe values under all fault conditions.
- Bare copper risers will be furnished for all underground electrical ducts and equipment, and for connections to the grounding systems within buildings.
- The design and analysis of the grounding system will follow the procedures and recommendations specified by the latest revision of IEEE 665, "Guide for Generation Station Grounding."
- Each building will be equipped with grounding systems connected to the station grounding grid. As a minimum, every other steel column of each building perimeter will connect directly to the grounding grid.
- The plant's main generator will be grounded with a neutral grounding device. The impedance of that device will limit the maximum phase current under short-circuit conditions to a value not greater than that for a three-phase fault at its terminals.
- Provisions will be included to ensure proper grounding of the isophase buses when the generator is disconnected.
- The onsite, medium-voltage ac-distribution system will be resistance grounded at the neutral point of the low-voltage windings of the unit auxiliary and reserve transformers.
- Grounding of the neutral point of the generator windings of the onsite standby power supply units (Class 1E diesel generators and non-Class 1E combustion turbine generator (CTG)) will be through distribution-type transformers and loading resistors, sized for continuous operation in the event of a ground fault.

Electric Power Systems

- The neutral point of the low-voltage ac-distribution systems will be either solidly or impedance grounded, as necessary, to ensure proper coordination of ground fault protection.
- The dc systems will be left ungrounded.
- Each major piece of equipment, metal structure, or metallic tank will be equipped with two ground connections diagonally opposite each other.
- The ground bus of all switchgear assemblies, MCCs, and control cabinets will be connected to the station ground grid through at least two parallel paths.
- One bare copper cable will be installed with each underground electrical duct run, and all metallic hardware in each manhole will be connected to this cable.
- Plant instrumentation will be grounded through separate radial grounding systems consisting of isolated instrumentation ground buses and insulated cables. The instrumentation grounding systems will be connected to the station grounding grid at only one point and will be insulated from all other grounding circuits.
- Separate instrumentation grounding systems shall be provided for plant analog and digital instrumentation systems.
- A lightning protection system will be provided for each major plant structure, including the containment enclosure building. The design and installation of these systems will comply with the National Fire Protection Association's Lightning Protection Code, NFPA-78, and the Nuclear Energy Property Insurance Association's "Basic Fire Protection for Nuclear Power Plants" document.
- Lightning arresters will be provided in each phase of all tie lines connecting the plant electrical systems to the switching station(s) and offsite transmission system. These arresters will be connected to the high-voltage terminals of the main step-up and reserve transformers.
- Plant instrumentation and monitoring equipment located outdoors or connected to cabling that runs outdoors will be equipped with built-in surge suppression devices to protect the equipment from lightning-induced surges.

GDC 17 of 10 CFR Part 50, Appendix A, requires provisions be included in the design to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the

loss of power generated by the nuclear power unit, the loss of power from the transmission network, or the loss of power from the onsite electric power supplies. The staff concludes that a design for plant structures, systems, and equipment which meets the above described commitments will be appropriately grounded and protected from lightning. The design, therefore, meets the above defined requirements of GDC 17 and is acceptable.

Verification that GE included the above design commitments in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.2.2.7-1. GE has included these commitments in Section 8.3.1.1.6.2 and Appendix 8A.1 of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.2.3.6-2 related to the design description and the ITAAC for the grounding and lightning protection. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

8.2.3.7 Operating Restrictions on the Offsite Preferred Power Circuits

Section 8.3.1.1.1 of the draft SSAR revision dated April 3, 1992, in response to DSER (SECY-91-355) Open Item 28, indicated that during normal operation (which includes: shutdown, refueling, startup, and run modes of plant operation), the normal preferred power supply feeds two of the three Class 1E load groups. The remaining load group is fed from the alternate power source. SSAR Section 8.3.4.9 specifies that COL applicants that implement the ABWR design must include in their operating procedures that one of the three divisional buses must be fed by the alternate power source during normal operation. The intent of this provision is to arrange the offsite power supply circuits to the Class 1E buses so that all three Class 1E divisional buses are not simultaneously de-energized on the loss of only one of the offsite power supplies.

GDC 17 of 10 CFR Part 50, Appendix A, requires provisions be included in the design to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated by the nuclear power unit, the loss of power from the transmission network, or the loss of power from the onsite electric power supplies. The staff concludes that a design which includes this operating procedure will minimize the probability of losing all Class 1E buses as a result of, or coincident with, the loss

of power generated by the nuclear power unit or the loss of power from the transmission network. The design, therefore, meets the above defined requirements of GDC 17 of 10 CFR Part 50, Appendix A, and is acceptable.

Verification that GE included in a future SSAR amendment the statement that the COL applicant include the above operating procedure was DFSER (SECY-92-349) COL Action Item 8.2.4-1. GE included this action item in Sections 8.2.4.2 and 8.3.4.9 of SSAR Amendment 32, which is acceptable.

8.2.3.8 Protection of Onsite Systems

The normal and alternate preferred offsite power circuits can be subjected to transmission system's steady-state and transient conditions (such as switching and lightning surges, maximum and minimum voltage ranges for heavy and light load conditions, frequency variation, or stability limits). GE indicated that provisions will be included in the design of the offsite and onsite systems to minimize the probability of losing electric power from any of the remaining sources as a result of these conditions in accordance with the requirements of GDC 17 of 10 CFR Part 50, Appendix A. In response to DSER (SECY-91-355) Open Item 26, GE revised the SSAR to indicate that these provisions include:

- Switching and lightning surge protection is provided by the station grounding and surge protection systems described in Appendix 8A of the SSAR, by independent feeds (i.e., normal and alternate preferred power circuits described in SSAR Section 8.2.1.2), and by grounding and lighting protection specified in SSAR Section 8.2.3.
- Maximum and minimum voltage ranges are specified in SSAR Section 8.2.3 and transformers are designed per SSAR Sections 8.2.1.2 and 8.2.3. Allowable frequency variation or stability limitations are addressed in SSAR Section 8.2.3.
- Protection for degraded voltage conditions is discussed in SSAR Section 8.3.1.1.7(8).

GDC 17 of 10 CFR Part 50, Appendix A, requires provisions be included in the design to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated by the nuclear power unit, the loss of power from the transmission network, or the loss of power from the onsite electric power supplies. The staff concludes that a design which meets these design provisions will minimize the probability of losing electric power

from any of the remaining sources as a result of transmission system's steady-state and transient conditions. The design, therefore, meets the above defined requirements of GDC 17 of 10 CFR Part 50, Appendix A, and is acceptable.

Verification that these design provisions were included in a future SSAR Amendment was part of DFSER (SECY-92-349) Confirmatory Item 8.2.2.6-1. GE included these provisions in Section 8.2.2.1(2) of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

Certified Design Material

This part of DFSER (SECY-92-349) Open Item 8.2.2.6-1 related to the design description and the ITAAC for the protection of offsite and onsite circuits from transmission system's steady-state and transient conditions. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

8.2.4 Power Supply for the Reactor Internal Pumps

Section 15.3.1.1.1 of SSAR Amendment 10 stated that, since four buses are used to supply power to the ten reactor internal pumps (RIPs), the worst single failure can only cause three RIPs to trip. In addition, the response to Question 435.4 of SSAR Amendment 10 stated that the probability of any additional RIP trips is low (less than $10E-6$ per year). Therefore, the simultaneous trip of more than three RIPs was classified by GE as a limiting fault. This classification was identified as DSER (SECY-91-355) Open Item 29.

The staff subsequently classified this postulated event in the special category of anticipated transients involving a common-mode software failure, and developed special acceptance criteria for the radiological dose calculation. The staff's evaluation of this postulated event in a special category is addressed in Section 15.2 of this report.

8.3 Onsite Class 1E Power System

GE provided in the SSAR markup dated March 31, 1993 as amended by SSAR markup dated May 11, 1993, a description of the ABWR design of the onsite Class 1E power system as being comprised of the following systems:

- Class 1E Alternating Current (ac) Power System

The Class 1E ac power system will consist of three redundant Class 1E ac safety system divisions (Divisions I, II, and III). Each of these divisions will

Electric Power Systems

include a Class 1E diesel generator standby power supply, ac-distribution system, and load group. Each of the three Class 1E diesel generator standby power supplies will consist of all components from the stored energy (fuel) to the connection to the distribution system's supply circuit breaker. Each Class 1E ac-distribution system will consist of the following Class 1E ac-distribution systems:

- 6.9 kV medium voltage ac-distribution system

Each of the three Class 1E 6.9 kV medium voltage ac-distribution systems (one per division) will consist of all equipment in the distribution circuit, from the power side of the offsite and onsite power supply breakers to and including the 6.9 kV Class 1E safety system loads. Equipment in each of the three Class 1E 6.9 kV ac-distribution circuits will include; one Class 1E circuit connection from the Class 1E diesel generator standby power supply, three circuit connections from non-Class 1E power supplies (the normal preferred offsite power circuit, the alternate preferred offsite power circuit, and the CTG), one 6.9 kV medium voltage Class 1E ac switchgear, the Class 1E circuit connections to and including one or more 6.9 kV medium voltage Class 1E safety system loads, the Class 1E circuit connections to two Class 1E 480-volt low voltage ac distribution systems, one Class 1E circuit connection to ground, and the Class 1E circuit connections through the Class 1E 480-volt low voltage ac-distribution systems to each of the following systems:

- one Class 1E 120-volt ac I&C distribution system,
- one or more Class 1E 120/240-volt ac-distribution systems,
- one or two Class 1E dc power systems, and
- one or two Class 1E vital 120-volt ac I&C power systems.

Equipment in Division I of the Class 1E 6.9 kV ac-distribution circuits will also include three circuit connections to non-Class 1E fine motion control rod drive (FMCRD) motor loads (one circuit connection to each of the three FMCRD motor load groups). Each of the three circuit connections will consist of all equipment in the distribution circuit from the 6.9 kV medium voltage switchgear to and including the 480-volt FMCRD motor loads. Equipment contained in the circuit will include a Class 1E zone selective interlock circuit between the Class 1E medium voltage supply and load breakers, a pair of Class 1E 6.9 kV interlocked breakers, one non-

Class 1E 6.9 kV/480-volt transformer, one non-Class 1E 480-volt MCC, and circuit connections from medium voltage switchgear through the interlocked breakers, 6.9 kV/480-volt transformer, and 480-volt MCC to FMCRD system motor loads. The circuit connection between the Class 1E medium voltage switchgear load breaker and the pair of interlocked breakers will be classified as associated. The circuit connections from interlocked breakers to the FMCRD loads and from the non-Class 1E medium voltage switchgear to the interlocked breakers will be classified as non-Class 1E. Control power for the interlocked breakers will be from the Division I Class 1E 125-volt dc system.

- 480-volt low-voltage ac-distribution system

Each of the six Class 1E 480-volt low voltage ac-distribution systems (two per division) will consist of all equipment in the distribution circuit from the 6.9 kV medium voltage side of the 6.9 kV/480-volt transformer to and including the 480-volt Class 1E safety system loads. Equipment in each of the six Class 1E 480-volt ac-distribution circuits will include one Class 1E circuit connection from the Class 1E medium voltage distribution system, one Class 1E 6.9 kV/480-volt transformer, one Class 1E 480-volt switchgear, Class 1E circuit connection to and including one or more 480-volt Class 1E safety system loads, one Class 1E circuit connection to ground, and Class 1E circuit connection to and including one or more Class 1E 480-volt ac MCCs and their affiliated 480-volt Class 1E safety system loads.

- 120-volt I&C ac-distribution system

Each of the three Class 1E 120-volt I&C ac-distribution systems (one per division) will consist of all equipment in the distribution circuit from the 480-volt side of the 480/120-volt transformer to and including the 120-volt Class 1E safety system I&C loads. Equipment in each of the three Class 1E I&C ac-distribution circuits will include one Class 1E 480/120-volt transformer, two Class 1E 120-volt ac-distribution panels, and Class 1E circuit connection to and including 120-volt Class 1E safety system I&C loads.

- 120/240-volt low-voltage ac-distribution system

Each of the Class 1E 120/240-volt low voltage ac-distribution systems (one or more per division) will consist of all equipment in the distribution circuit

from the 480-volt side of the Class 1E 480/120/240-volt transformer to and including Class 1E safety system loads. Equipment in each of the Class 1E 120/240-volt distribution circuits will include one Class 1E 480/120/240-volt transformer, one or more Class 1E 120/240-volt ac distribution panels, and Class 1E circuit connection to and including Class 1E 120- and 240-volt safety system loads.

• Class 1E Direct Current (dc) Power System

The Class 1E dc power system will consist of four redundant 125-volt Class 1E dc safety system divisions (Divisions I, II, III, and IV). Each of these divisions will include a Class 1E battery and battery charger power supply, 125-volt dc-distribution system, and load group. Each of the four Class 1E battery power supplies will consist of storage cells, connectors, and connections to the distribution system supply circuit interrupting device. Each of the four Class 1E battery charger power supplies will consist of all equipment from the connection to the 480-volt Class 1E ac-distribution system to its distribution system's supply breaker. Each of the four Class 1E 125-volt dc-distribution systems will consist of all equipment in the distribution circuit, from the power side of the battery interrupting device and the battery charger supply breaker to and including the 125-volt Class 1E dc safety system loads. Equipment in each of the four distribution circuits will include one or more Class 1E distribution panels and connections to and including 125-volt Class 1E dc safety system loads.

Divisions I and III of the 480-volt Class 1E ac-distribution system will feed the Divisions I and III battery charger power supplies, respectively. Division II of the 480-volt Class 1E ac-distribution system will feed the Divisions II and IV battery charger power supplies.

• Class 1E Vital ac I&C Power System

The Class 1E vital ac I&C power system will consist of four redundant Class 1E vital 120-volt ac I&C safety system divisions (Divisions I, II, III, and IV). Each of these divisions will include a Class 1E Constant Voltage Constant Frequency (CVCF) power supply, a 120-volt Class 1E ac-distribution system, and a load group. Each of the four CVCF power supplies will consist of the power source (the static inverter, ac and dc static transfer switches, and a regulating step down transformer as an alternate ac power supply) and its connection to the distribution supply circuit interrupting device. Each of the four Class 1E vital 120-volt

ac I&C distribution systems will consist of all equipment in the distribution circuit from the power side of the constant CVCF power supply breaker to and including the Class 1E safety system I&C loads. Equipment in each of the four Class 1E vital 120-volt ac distribution circuits will include one or more 120-volt ac distribution panels and connections to and including vital 120-volt Class 1E safety system I&C loads.

Each divisional CVCF power supply will be supplied power from its affiliated divisional dc power system. (For example, the Division I CVCF power supply will be supplied from Division I 125-volt dc-distribution system.) In addition, Divisions I and III of the 480-volt ac-distribution system will supply power to the Divisions I and III CVCF power supplies, respectively. Similarly, Division II of the 480-volt ac-distribution system will supply power to the Divisions II and IV CVCF power supplies.

The staff concludes that the above design is consistent with the typical electrical system design defined by IEEE 308-1980. The staff's review of the ABWR electrical system design was based on this typical design. GE has included the above typical design in SSAR Amendment 32, which is acceptable.

In addition, GE indicated that operational restrictions would apply to the use of Class 1E receptacles which are powered from each of the Class 1E 120/240-volt distribution systems. Verification that these operational restrictions will be included in appropriate COL procedures to ensure compliance with the capacity, independence, and protection provisions required by GDC 2, 4, 17, and 18 of 10 CFR Part 50, Appendix A for Class 1E power systems was DFSE (SECY-92-349) COL Action Item 8.3-1. In response to this item, GE revised their electrical system design to eliminate Class 1E electrical receptacles. GE included their revised design in Section 8.3.1.1.3 of SSAR Amendment 32 negating the need for a specific COL action item which is acceptable.

GDC 17 of 10 CFR Part 50, Appendix A, requires that the Class 1E systems have sufficient (1) capacity and capability to permit safety systems to perform their required safety function and (2) sufficient independence and redundancy to perform their safety functions assuming a single failure. With the elimination of Class 1E receptacles, the potential for loss of independence between redundant Class 1E divisions and loss of sufficient capacity of the Class 1E power supplies due to incorrect use of the receptacles was eliminated from the design. Consequently, the design meets the above defined requirements of GDC 17 and is acceptable.

Electric Power Systems

Certified Design Material

DFSER (SECY-92-349) Open Item 8.3-1 related to the design description and the ITAAC for the onsite Class 1E power system design. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. On the basis of this evaluation, this item is resolved.

To ensure that the ABWR design incorporates sufficient capacity, capability, independence, redundancy, and testability of onsite Class 1E power systems, the staff's evaluation also addressed the areas discussed in the following sections.

8.3.1 Compliance with General Design Criteria

This section addresses the staff's evaluation of GE's response to DSER (SECY-91-355) Open Items 27 and 30.

Item (1)(b) of Section 8.3.1.2.1 of SSAR Amendment 10 indicated that the Class 1E ac power system complies with GDC 2, 4, 17, and 18 in part or as a whole, as applicable. GE's response to Question 435.26 (also of Amendment 10) provided clarification that there are no non-compliances, but also indicated that some portions of the GDC do not apply to the CVCF power supplies (for example, the statement in GDC 17 about two physically independent circuits from the transmission network). Based on the information presented, the staff could not ascertain which parts of the GDC GE considered not applicable to the CVCF power supplies.

In its draft submittal dated September 4, 1991, GE proposed modifying the response to Question 435.26 and Section 8.3.1.2.1 to indicate full compliance with the GDC. The proposed modification deleted (1) certain conflicting statements in the SSAR, (2) the example of non applicability to GDC 17, and (3) the phrase "the substance and intent of" from Section 8.3.1.4.2.1. In addition, GE agreed to revise Item 11 of Section 1.2.1.1.2 of SSAR Amendment 1 to clarify the systems or components to which IEEE 279, "Criteria for Protection Systems for Nuclear Power Generating Stations" (1971) applies and to correct inconsistencies concerning applicable SRP criteria within Table 8.1-1 and between Table 8.1-1 and Section 8.1.3.1.2.

The staff concludes that the ABWR electrical system design with the above mentioned changes will comply with GDC 17 requirements to GDC commitments and is acceptable. Verification that GE provided the above clarifications in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.3.1-1. GE included this information in Table 8.1-1 of SSAR Amendment 21

and in Sections 1.2.1.1.2(11), 8.3.1.2(1)(b), 8.3.3.6.2.1, and 8.1.3.1.2 of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

8.3.1.1 Compliance with IEEE Standards (NRC Policy Issue SECY-91-273, Section II.A of Enclosure 1 to SECY-93-087)

In the draft SSAR submittal dated September 4, 1991, GE indicated that there will be no limitation on the use of IEEE 384-1981 "IEEE Standard Criteria for Independence of Class 1E Equipment and Circuits," for separation in the ABWR design. The NRC staff has not formally reviewed and accepted the changes between the 1974 and 1981 versions of this standard. Also, IEEE 384 is not the only standard in this classification for which the NRC has not completed its formal evaluation and regulatory guide endorsement of the newer version of the standard. Thus, to allow the use of an updated IEEE standard which the NRC staff has not formally reviewed or endorsed, the staff felt that each difference between the old and new standard needed to be identified, justified, and approved for use. Such review is required in order to ensure that the design criteria of the new standards are equally conservative as those included in the standards currently approved by the staff. This was DSER (SECY-91-355) Open Item 41.

In order to identify, justify, and approve the differences between the old and new standards for use on the ABWR and also complete the evaluation of the electrical system design for the ABWR, the staff evaluated the guidelines in two of the newer standards with respect to the intent of criteria and guidelines contained in the SRP and existing regulatory guides. The staff believes that these two standards provide the majority of the basic criteria for the electrical power systems. The electrical design proposed for the ABWR was then evaluated using the newer standards.

The newer standards were revised to be consistent with IEEE 603-1980. The newer standards involved primarily clarification and amplification of guidelines contained in prior standards and were thus considered the more relevant base from which to evaluate the ABWR design. The newer standards included

- IEEE 308-1980, "IEEE Standard Criteria for Class 1E Power Systems for Nuclear Power Generating Stations" (This standard was specifically revised to be consistent with IEEE 603-1980, which was endorsed by the NRC staff in RG 1.153 in 1985.)
- IEEE-384-1981 "IEEE Standard Criteria for Independence of Class 1E Equipment Circuits"

In addition, IEEE has developed and issued other companion standards to provide additional guidance for certain areas. These standards include

- IEEE 741-1986, "Standard Criteria for the Protection of Class 1E Power Systems and Equipment in Nuclear Power Generating Stations"
- IEEE 765-1983, "Standard for Preferred Power Supply for Nuclear Power Generating Stations"
- IEEE 485-1983, "Recommended Practice for Sizing Large-Lead Storage Batteries for Generating Stations and Substations"
- IEEE 946-1985, "Recommended Practice for the Design of Safety-Related dc Auxiliary Power Systems for Nuclear Power Generating Stations"

Like the two standards cited above, these other companion standards have not been endorsed by an NRC regulatory guide. However, these standards were developed to be used with IEEE 308-1980 and they also clarify and amplify current SRP criteria and guidelines. The staff therefore considers these standards the more relevant base from which to evaluate the ABWR design.

In some cases, GE had not referenced these other companion standards in the ABWR SSAR. The staff proceeded with its review with the understanding that GE intended to use these other standards. Verification of this understanding in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory item 8.3.1.1-1. GE included a statement of conformance to these standards in Sections 8.3.1.2(5), 8.2.2.1(9), and 8.3.2.2.2(5) of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

8.3.1.2 Compliance with GDC 2 and 4

ABWR SSAR Chapter 8 was modified, in response to DSER (SECY-91-355) Open Item 31 and 63, by the draft revision dated April 3, 1992, as amended by an SSAR markup dated March 31, 1993, to include the following statements related to the compliance of the electrical system design to the requirements of GDC 2, "Design Bases for Protection Against Natural Phenomena," and GDC 4, "Environmental and Missiles Design Bases," of 10 CFR Part 50, Appendix A:

- "Electrical equipment and wiring for the Class 1E systems which are segregated into separate divisions are separated so that no design-basis event is capable of disabling more than one division of any ESF total function."

- "Redundant parts of the system are physically separated and electrically independent to the extent that in any design-basis event with any resulting loss of equipment, the plant can still be shut down with the remaining two divisions."
- "Class 1E electric equipment and wiring is segregated into separate divisions so that no single credible event is capable of disabling enough equipment to hinder reactor shut down and removal of decay heat by either of two unaffected divisional load groups or prevent isolation of the containment in the event of an accident."
- "Equipment arrangement and/or protective barriers are provided such that no locally generated force or missile can destroy any reactor protection system (RPS), nuclear steam supply system (NSSS), emergency core cooling system (ECCS), or engineered safety feature (ESF) functions. In addition, arrangement and/or separation barriers are provided to ensure that such disturbances do not affect both high pressure core flooders (HPCF) and reactor core isolation cooling (RCIC) systems."
- "Containment penetrations are so arranged that no design-basis event can disable cabling in more than one division."
- "The protection system and ESF control, logic, and instrument panels/racks shall be located in a safety class structure in which there are no potential sources of missiles or pipe breaks that could jeopardize redundant cabinets and raceways."
- "The standby ac power system is capable of providing the required power to safely shut down the reactor after LOPP and/or LOCA and to maintain the safe-shutdown condition and operate the Class 1E auxiliaries necessary for plant safety after shutdown."

Based on the above stated capability to safely shut down, other SSAR commitments regarding physical protection of electrical divisions, and discussions, GE indicated that there will be a limited number of design-basis events for which Class 1E systems, equipment, and components will be protected by the capability to maintain safe plant shut down with any one of the three load groups.

GDC 2 and 4 of 10 CFR Part 50, Appendix A, requires that Class 1E systems, equipment, and components be designed (1) to withstand the effects of natural phenomena without loss of capability to perform their safety functions, (2) to accommodate the effects of and to be compatible with the environmental conditions associated with normal

Electric Power Systems

operation, maintenance, testing, and postulated accidents, and (3) be appropriately protected against dynamic effects. GDC 17 of 10 CFR Part 50, Appendix A, requires that the Class 1E systems, equipment, and components have sufficient independence to perform their safety functions assuming a single failure. The staff concludes that a design that meets the above described commitments will reasonably assure that no design basis event (that is, failure of any one safety-related system division or failure of non-Class 1E equipment) will cause failure of more than one safety-related system division. The design, therefore, meets the above defined requirements of GDC 2, 4, and 17 of 10 CFR Part 50, Appendix A, and is acceptable. Verification that GE has provided the above design commitments in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.3.1.2-1. GE has included information in Sections 8.3.3.1, 8.1.3.1.1.1, 8.3.3.6.1.1, 8.3.3.6.2.3.2, 8.3.3.6.2.2.3, and 8.1.2.2 of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.3.1.2-1 related to the design description and the ITAAC for the capability of performing a safe shutdown (performing the required minimum safety function). The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

8.3.2 Physical Independence

8.3.2.1 Conduits to Open Tray Separation (Scram Cables)

Section 8.3.3.6.2.3.1 of the draft SSAR revision dated April 3, 1992, documented the following design commitments in response to DSER (SECY-91-355) Open Item 32:

- The reactor protective system scram solenoid circuits, from the actuation devices to the solenoids of the scram pilot valves of the control rod drive hydraulic control units, will be run in grounded steel conduits, containing no other wiring.
- Separate grounded steel conduits will be provided for the scram solenoid wiring for each of four scram groups.
- Separate grounded steel conduits will also be provided for both the A and B solenoid wiring circuits of the same scram group.

- Scram group conduits will have unique identification and will be separately routed as Divisions II and III conduits for the A and B solenoids of the scram pilot valves, respectively. This corresponds to the divisional assignment of their power sources.
- Conduits containing the circuits for the A solenoids of the scram pilot valves (Division II) will be separated from their B solenoid counterpart (Division III) by a minimum separation distance of 2.54 cm (1 in.), in accordance with divisional separation requirements.
- The scram group conduits will not be routed within the confines of any other tray or raceway system.
- The conduits containing the scram solenoid group wiring of any one scram group will also be physically separated by a minimum distance of 2.54 cm (1 in.) from the conduit of any other scram group and from conduits or metal-enclosed raceways affiliated with any of the four Class 1E divisions or any non-Class 1E (non-divisional) circuits.
- The conduits containing the scram solenoid group wiring of any one scram group will also be physically separated from non-enclosed raceways associated with any of the four safety-related electrical divisions or any non-safety-related (non-divisional) circuits in accordance with IEEE 384 and RG 1.75, Revision 2.
- Separation "in accordance with IEEE 384" means that conduits containing scram solenoid group circuit wiring will be separated from any non-enclosed raceway containing either safety or non-safety-related circuits. Specifically, the vertical separation distance will be 1.5 or more m (5 or more ft) and the horizontal separation distance will be .9 or more m (3 or more ft).

GDC 4 of 10 CFR Part 50, Appendix A, requires that Class 1E systems, equipment, and components be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents. GDC 17 of 10 CFR Part 50, Appendix A, requires that the onsite electric distribution system have sufficient independence to perform their safety function assuming a single failure. The staff concludes that a design that meets the above described commitments will reasonably preclude the common failure of reactor protection system scram solenoid circuits and other Class 1E or non-Class 1E circuits. The design, therefore, meets the above defined requirements of GDC 4 and 17 of 10 CFR Part 50, Appendix A, and is acceptable.

Verification that GE provided the above design commitments in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.3.2.1-1. In response to this item, GE further clarified their design commitment for separation in accordance with IEEE 384. Specifically GE indicated that the RPS conduits containing the scram group wiring for the A and B solenoids of the scram pilot valves (associated with Divisions II and III, respectively), will be separated from non-enclosed raceways associated with any of the four electrical divisions or non-divisional cables by 0.9 m (3 ft) horizontally, or 1.5 m (5 ft) vertically, or with an additional barrier that is separated from any raceway by 2.5 cm (1 in.). The staff concludes that a design that meets the above described commitments as revised will also reasonably preclude the common failure of reactor protection system scram solenoid circuits and other Class 1E or non-Class 1E circuits. The design, therefore, meets the above defined requirements of GDC 4 and 17 of 10 CFR Part 50, Appendix A, and is acceptable. GE has included the revised design commitments in Section 8.3.3.6.2.3.1 of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.3.2.1-1 related to the design description and the ITAAC for separation between scram cables and between scram and other cables. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

8.3.2.2 Dedicated Neutron Monitoring Raceways

Section 8.3.1.3.1.3 of the draft SSAR revision dated April 3, 1992, (in response to DSER (SECY-91-355) Open Item 32) indicated that neutron monitoring cables will be routed in their own divisional conduits and cable trays, separate from all other power, instrumentation, and control cables.

GE also committed that neutron monitoring cables will be routed in their own dedicated raceways from termination to termination. These dedicated raceways will be separated from raceways containing all other Class 1E or non Class 1E power, instrumentation, and control cables by the same separation provided between scram and other cables described in Section 8.3.2.1 of this report.

GDC 4 of 10 CFR Part 50, Appendix A, requires that Class 1E systems, equipment, and components be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents. GDC 17 of 10 CFR Part 50, Appendix A, requires that the

onsite electric distribution system have sufficient independence to perform their safety function assuming a single failure. The staff concludes that a design that meets the above described commitments will reasonably preclude the common failure of neutron monitoring circuits and other Class 1E or non-Class 1E circuits. The design, therefore, meets the above defined requirements of GDC 4 and 17 of 10 CFR Part 50, Appendix A, and is acceptable.

Verification that GE provided the above design commitments in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.3.2.2-1. In response to this item, GE further clarified in its design that neutron monitoring cables will be routed in their own dedicated raceways for the purpose of assuring their protection from the effects of electromagnetic interference (EMI). GE indicated that additional physical separation requirements are not necessary to assure protection from the effects of EMI.

The staff concludes that a design that meets the above described commitments as modified will also reasonably preclude the common failure of neutron monitoring circuits and other Class 1E or non-Class 1E circuits. The design, therefore, meets the above defined requirements of GDC 17 of 10 CFR Part 50, Appendix A, and is acceptable. GE has included this commitment in Section 8.3.3.5.1.3 of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

The issue related to protection of circuits from EMI is resolved in Chapter 7.0 of this report.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.3.2.2-1 related to the design description and the ITAAC for routing of I&C neutron monitoring circuits in dedicated raceways. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

8.3.2.3 Separation of dc Emergency Lighting Raceways

In Section 9.5.3 of the draft SSAR revision dated April 3, 1992, in response to DSER (SECY-91-355) Open Item 32, GE indicated that dc emergency lighting cables will be routed in their own divisional conduits and cable trays, separate from all other power, instrumentation, and control cables.

GE indicated that the dc emergency lighting cables will be routed in their own dedicated raceways from termination to termination. These dedicated raceways will be separated from raceways containing all other Class 1E or

Electric Power Systems

non-Class 1E power and I&C cables by the same separation provided between scram and other cables, as described in Section 8.3.2.1 of this report.

GDC 4 of 10 CFR Part 50, Appendix A, requires that Class 1E systems, equipment, and components be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents. GDC 17 of 10 CFR Part 50, Appendix A, requires that the onsite electric distribution system have sufficient independence to perform their safety function assuming a single failure. The staff concludes that a design that meets the above described commitments will minimize to the extent practicable the common failure of the standby and dc emergency lighting circuits. The design, therefore, meets the above defined requirements of GDC 4 and 17 of 10 CFR Part 50, Appendix A, and is acceptable.

Verification that GE provided the above design commitments in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.3.2.3-1. In response to this item, GE clarified their design to indicate that emergency dc lighting circuits will not share raceways with any other circuits in order to enhance lighting reliability. GE indicated that physical separation of these raceways within the same division is not required.

The staff concludes that a design that meets the above described commitment as modified will also minimize to the extent practicable the common failure of the standby and dc emergency lighting circuits. The design, therefore, meets the above defined requirements of GDC 4 and 17 of 10 CFR Part 50, Appendix A, and is acceptable. GE has included this design commitment in Sections 9.5.3 and 9.5.3.1.1(7)(d) of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.3.2.3-1 related to the design description and the ITAAC for routing of emergency dc lighting in dedicated raceways. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

8.3.2.4 Separation of Containment Electrical Penetrations

Separation Between Class 1E Penetrations of Redundant Divisions

Item (7) of Section 8.3.1.4.1.2 of SSAR Amendment 10 indicated that electric penetration assemblies of different

Class 1E divisions will be separated by distance, barriers, and/or location in separate rooms or on separate floors. The use of barriers and/or location in separate rooms or on separate floors exceeds separation guidelines for penetrations and are acceptable approaches. Separation by distance may also meet separation guidelines; however, SSAR Amendment 10 did not clearly define what constitutes the minimum allowable distance between penetrations. This was DSER (SECY-91-355) Open Item 33.

In the draft SSAR revision dated April 3, 1992, item (7) of Section 8.3.1.4.1.2 similarly indicated that electrical penetration assemblies of different Class 1E divisions will be separated by 3-hour fire-rated-barriers (that is locations in separate rooms or on separate floors). Separation by distance (without barriers) will be allowed only within the inerted containment. Section 8.3.1.1.5.1 of the April draft revision further indicated that penetration assemblies will be located around the periphery of the containment and at different elevations to facilitate reasonably direct routing of cables to and from the equipment.

GDC 4 of 10 CFR Part 50, Appendix A, requires that Class 1E systems, equipment, and components be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents. GDC 17 of 10 CFR Part 50, Appendix A, requires that the onsite electric distribution system have sufficient independence to perform their safety function assuming a single failure. The staff concludes that a design that meets the above described commitments will reasonably ensure that failure of Class 1E (or associated) penetration circuits in any one division will not cause failure of Class 1E (or associated) penetration circuits in a different Class 1E division. The design, therefore, meets the above defined requirement of GDC 17 of 10 CFR Part 50, Appendix A, and is acceptable.

Verification that GE provided the above design commitments in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.3.2.4-1. GE has included these commitments in Sections 8.3.3.1 and 8.3.3.6.1.2 of SSAR Amendment 32, which is acceptable.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.3.2.4-1 related to the design description and the ITAAC for the separation between Class 1E penetrations of redundant divisions. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

Separation of Class 1E Penetrations From Non-Class 1E Penetrations

Section 8.3.1.4.1.2 of the draft SSAR revision dated April 3, 1992, in response to DSER (SECY-91-355) Open Item 34 indicated that separation between penetrations containing non-Class 1E circuits and those containing Class 1E or associated circuits will be in accordance with IEEE 384. GE indicated that "separation in accordance with IEEE 384" means a vertical separation of 1.5 or more m (5 or more ft) and a horizontal separation distance of 0.9 or more m (3 or more ft).

The staff concludes that a design which meets the above described commitments will reasonably ensure that failure of non-Class 1E system penetration circuits will not cause failure of Class 1E (or associated) penetration circuits. The design, therefore, meets the above defined requirements of GDC 4 and 17 of 10 CFR Part 50, Appendix A, and is acceptable.

Verification that GE provided the above design commitments in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.3.2.4-2. In response to this item, GE further clarified their design to indicate that the separation between Electrical penetration assemblies containing non-Class 1E circuits and penetration assemblies containing Class 1E or associated Class 1E circuits is by walls, barriers, or floors that have a three-hour fire-rating. The staff concludes that a design which meets the above described commitments will also reasonably ensure that failure of non-Class 1E penetration circuits will not cause failure of Class 1E (or associated) penetration circuits. The design, therefore, meets the above defined requirements of GDC 4 and 17 of 10 CFR Part 50, Appendix A, and is acceptable. GE has included this design commitment in Section 8.3.3.6.1.2(7) of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.3.2.4-2 related to the design description and the ITAAC for the separation between Class 1E and non-Class 1E penetrations. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

Separation of Class 1E Penetrations from Non-Class 1E Cables or Other Divisional Cables

This section addresses the staff's evaluation of GE's response to DSER (SECY-91-355) Open Item 35.

GE indicated that penetrations containing Class 1E circuits will be separated from other divisional cables by routing through separate rooms and/or different floors outside containment and by maintaining a minimum separation of 0.9 m (3 ft) horizontal and 1.5 m (5 ft) vertical distance inside the inerted containment. In addition, separation between penetrations containing Class 1E circuits and non-divisional cables will be maintained at a minimum horizontal distance of 0.9 m (3 ft) and a vertical distance of 1.5 m (5 ft) both inside and outside of containment.

The staff concluded that a design which meets the above described commitments will also reasonably ensure (1) that failure of Class 1E (and associated) circuits of one division will not cause failure of Class 1E (or associated) penetration circuits in a different division and (2) that failure of non-Class 1E circuits will not cause failure of Class 1E (or associated) penetration circuits. The design, therefore, meets the above requirements of GDC 4 and 17 of 10 CFR Part 50, Appendix A, and is acceptable.

Verification that GE provided the above design commitments in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.3.2.4-3. In response to this item, GE further clarified their design to indicate that the separation between Electrical penetration assemblies containing Class 1E or associated circuits from other divisional or non-divisional cables is by walls, barriers, or floors that have a three-hour fire-rating. The staff concluded that this separation exceeds the guidelines of IEEE 384, meets the protection and independence requirements of GDC 4 and 17 of 10 CFR Part 50, Appendix A, and is acceptable. GE included this design information in Section 8.3.3.6.1.2(7) of SSAR Amendment 32, which is acceptable. Therefore, DFSER (SECY-92-349) Confirmatory Item 8.3.2.4-3 is resolved.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.3.2.4-3 related to the design description and the ITAAC for the separation between Class 1E penetrations to non-Class 1E cables or to other divisional cables. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

8.3.2.5 Separation and Protection of Cables Located Outside Cabinets and Panels

Section 8.3.1.1.5.1 of the draft SSAR revision dated April 3, 1992, documents the following design commitments related to separation of Class 1E cables in response to DSER (SECY-91-355) Open Item 36:

Electric Power Systems

- (1) Enclosed solid metal raceways are required for separation between Class 1E or associated cables of different safety divisions and between Class 1E or associated cables and non-Class 1E cables, if: the vertical separation is less than 1.5 m (5 ft), the horizontal separation distance is less than .9 m (3 ft), and the cables are located in the same fire area.
- (2) Both groupings of cables requiring separation (as specified in item 1) must be enclosed in solid metal raceways.

GE indicated that all power, control, and instrumentation cables (including fiber optic cables) located outside cabinets and panels throughout the plant will be supported in raceways in accordance with IEEE-recommended practice for support of cable systems. When Class 1E (or associated) cables of different Class 1E divisions are separated from each other or from non-Class 1E cables by less than 1.5 m (5 ft) vertically or .9 m (3 ft) horizontally, the cables will be supported in enclosed solid metal raceways (such as rigid or flexible metal conduits or totally enclosed cable trays).

In addition, Section 8.3.1.2.1(2)(f) of the draft SSAR revision dated April 3, 1992, and Section 1.8.2 (Table 1.8-21) of SSAR Amendment 17 indicated that the ABWR electric system design will comply with the requirements of IEEE 384-1981. Based on this commitment, GE indicated that the separation distance will be at least 2.54 cm (1 in.) between solid metal raceways containing Class 1E (or associated) cables of different Class 1E divisions or between solid metal raceways containing Class 1E (or associated) cables and non-Class 1E cables.

GDC 4 of 10 CFR Part 50, Appendix A, requires that Class 1E systems, equipment, and components be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents. GDC 17 of 10 CFR Part 50, Appendix A, requires that the Class 1E systems, equipment, and components have sufficient independence to perform their safety functions assuming a single failure. The staff concludes that a design which meets the above described commitments will reasonably ensure (1) that failure of Class 1E (or associated) cables in any one division (located outside of cabinets and panels and in any single raceway) will not cause failure of Class 1E (or associated) cables in a different Class 1E division and (2) that failure of non-Class 1E cables (located outside of cabinets and panels and in any single raceway) will not adversely affect Class 1E (or associated) cables. Consequently, the staff concludes

that the design meets the above defined requirements of GDC 4 and 17 of 10 CFR Part 50, Appendix A, and is acceptable.

Verification that GE provided the above design commitments in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.3.2.5-1. GE has included these commitments in Section 8.3.3.1 of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.3.2.5-1 related to the design description and the ITAAC for the separation and protection of cables located outside cabinets and panels. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

8.3.2.6 Separation of Cables Inside Cabinets and Panels

In response to DSER (SECY-91-355) Open Item 37, Sections 8.3.1.4.1, 8.3.1.4.1.2, 8.3.1.4.2, 8.3.1.3.1.3, and 8.3.1.4.2.2.3 of Amendment 10 to the SSAR and draft SSAR revision dated April 3, 1992, document the following design commitments related to separation of power, control, and instrumentation cables inside panels, racks, cabinets, and other enclosures located in the main control room and other areas of the plant.

- Single panels or instrument racks will not contain circuits or devices of different Class 1E safety system divisions, except under the following conditions:
 - Certain operator interface control panels may have operational considerations which dictate that Class 1E safety system circuits or devices of different divisions must be located in a single panel. These circuits and devices will be separated horizontally and vertically by a minimum distance of 15.2 cm (6 in.) or by steel barriers or enclosures.
 - The input and output circuits of isolation devices will be separated horizontally and vertically by a minimum distance of 15.2 cm (6 in.) or by steel barriers or enclosures.
- Class 1E circuits and devices will also be separated from the non-Class 1E circuits and devices which are present inside a panel. These circuits and devices will be separated from each other horizontally and vertically

by a minimum distance of 15.2 cm (6 in.) or by steel barriers or enclosures.

- If two panels containing circuits of different divisions are less than .9 m (3 ft) apart, there will be a steel barrier between the two panels. Panel ends closed by steel end plates will be considered to be acceptable barriers provided that terminal boards and wireways are spaced a minimum of 2.5 cm (1 in.) from the end plate.
- Penetration of separation barriers within a subdivided panel will be permitted, provided that such penetrations are sealed or otherwise treated so that fire generated by an electrical fault could not reasonably propagate from one section to the other and disable a protective function.

Based on the commitment to meet the guidelines of IEEE 384-1981 and RG 1.75, Revision 2, GE indicated that Class 1E or non-Class 1E power circuits located inside panels and cabinets will be limited to those required to operate systems, equipment, or components located inside the panels and cabinets. Power cables will not be permitted to traverse from one side of a panel or cabinet to the other without being terminated inside the panel. In addition, these circuits will be routed inside rigid or flexible conduits that will be physically separated from instrumentation and control cables by minimum horizontal and vertical distances of 15.2 cm (6 in.) or by steel barriers or additional enclosures.

GDC 4 of 10 CFR Part 50, Appendix A, requires that Class 1E systems, equipment, and components be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents. GDC 17 of 10 CFR Part 50, Appendix A, requires that the Class 1E systems, equipment, and components have sufficient independence to perform their safety functions assuming a single failure. The staff concludes that a design which meets the above described commitments will reasonably ensure (1) that failure of Class 1E (or associated) cables in any one division (located inside of cabinets or panels) will not cause failure of Class 1E (or associated) cables in a different safety division, (2) failure of non-Class 1E cables (located inside cabinets or panels) will not adversely affect Class 1E (or associated) cables, and (3) normal operation and/or failure of power circuits (Class 1E, associated Class 1E, or non-Class 1E) will not adversely affect I&C circuits. The design, therefore, meets the above defined requirements of GDC 4 and 17 of 10 CFR Part 50, Appendix A, and is acceptable.

Verification that GE provided the above design commitments in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.3.2.6-1. In response to this item, GE indicated that it was not their intent to commit to the routing of power circuits inside rigid or flexible conduits that will be physically separated from instrumentation and control cables of the same division by minimum horizontal and vertical distances of 15.2 cm (6 in.) or by steel barriers or additional enclosures. GE further indicated that the IEEE guideline "installed in enclosed raceways that qualify as barriers" has been interpreted to mean that for EMI considerations, power cables will be routed in metallic conduit wherever they come in close proximity with low level (V1) cables. For independence between power and I&C cables within the same division, the staff agreed with GE's interpretation. The staff concludes that a design that meets the above described commitments as clarified will reasonably ensure that normal operation and/or failure of power circuits (Class 1E, associated Class 1E, or non-Class 1E) will not adversely affect I&C circuits. The design, therefore, meets the above defined requirements of GDC 4 and 17 of 10 CFR Part 50, Appendix A, and is acceptable. The staff's evaluation for the protection of circuits from EMI is addressed in Chapter 7.0 of this report. GE has included the above design commitments in Sections 8.3.3.6.1.1, 8.3.3.6.2.2.3, and 8.3.3.6.2.2.4 of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.3.2.6-1 related to the design description and the ITAAC for the separation of cables inside cabinets/panels. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

8.3.2.7 Separation of Cables Approaching and/or Exiting Cabinets and Panels

The response to Question 435.30 in SSAR Amendment 10, DSER (SECY-91-355) Open Item 40, stated that cable spreading areas do not apply to the ABWR and are not included in the plant layout because the majority of the I&C signals will be multiplexed to the control room. This response implied that the 0.3 m (1 ft) by 0.9 m (3 ft) separation guidelines allowed by Section 5.1.3 of IEEE 384-1974 (Section 6.1.3 of IEEE 384-1981) will not apply to the ABWR. The guidelines of Position C12 of RG 1.75 Revision 2, also will be irrelevant. The ABWR SSAR did not clearly address the criteria for the separation and protection of cables approaching or exiting cabinets and panels.

Electric Power Systems

In discussions with the staff and in Section 8.3.1.4 of the draft SSAR revision dated April 3, 1992, GE clarified its design commitments as follows:

- I&C and optical cables (including metallic and fiber-optic cables) will be treated the same with respect to separation and protection throughout the plant.
- Each division of Class 1E power, instrumentation, and control cables will be routed to the control room complex through a cable chase or other means, so that different divisional areas will be separated by a 3-hour fire-rated barrier.
- Each cable chase will be ventilated.
- Separation between Class 1E and non-Class 1E cables within the cable chase will be the same as separation of cables located outside cabinets and panels as described in Section 8.3.2.5 of this report.
- Class 1E, associated, or non-Class 1E power circuits routed in a cable chase serving the control room or in the control room area will be limited to those required to operate systems, equipment, or components located in the control room area (power cables will be not be permitted to traverse through from one side of the control room area to the other without being terminated in the control room area).
- Class 1E, associated, or non-Class 1E power circuits routed in a cable chase or the control room area will be routed inside rigid or flexible conduits that will be physically separated horizontally and vertically from any I&C cables by a minimum distance of 15.2 cm (6 in.) or by steel barriers or additional enclosures.
- Power cables may be routed in flexible metallic conduit under the raised floor of the control room.
- Separation between divisional and between divisional and non-divisional power, instrumentation, and control cables within the control room area will be separated in the same way as cables located outside cabinets and panels described in Section 8.3.2.5 of this report.
- Power, instrumentation, and control cables of different Class 1E divisions will enter cabinets and panels through separate apertures. Similarly, Class 1E and non-Class 1E power, instrumentation, and control cables will enter cabinets or panels through separate apertures.

- Cable chases and the control room area will be non-hazard areas (Section 6.1.3 of IEEE 384-1981 defines non-hazard areas).
- Cable chases and the control room area will not contain potential hazards such as high energy switchgear, power distribution panels, transformers or rotating equipment, potential sources of missiles, pipe failure hazards, or fire hazards.

GDC 4 of 10 CFR Part 50, Appendix A, requires that Class 1E systems, equipment, and components be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents. GDC 17 of 10 CFR Part 50, Appendix A, requires that the Class 1E systems, equipment and components have sufficient independence to perform their safety functions assuming a single failure. The staff concludes that a design which meets the above described commitments will reasonably ensure (1) that failure of Class 1E (or associated) cables in any one division (located in a cable chase or the control room area) will not cause failure of Class 1E (or associated) cables in a different safety division, (2) that failure of non-Class 1E cables (located in a cable chase or the control room area) will not adversely affect Class 1E (or associated) cables, and (3) normal operation and/or failure of power circuits (Class 1E, associated, or non-Class 1E) will not adversely affect I&C circuits. The design therefore meets the above defined requirements of GDC 4 and 17 of 10 CFR Part 50, Appendix A, and is acceptable.

Verification that GE provided the above design commitments in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.3.2.7-1. In response to this item, GE indicated that it was not their intent to commit to the routing of power circuits inside rigid or flexible conduits that will be physically separated from instrumentation and control cables of the same division by minimum horizontal and vertical distances of 15.2 cm (6 in.) or by steel barriers or additional enclosures. GE further indicated that the IEEE guideline "installed in enclosed raceways that qualify as barriers" has been interpreted to mean that for EMI considerations, power cables will be routed in metallic conduit wherever they come in close proximity with low level (V1) cables. For independence between power and I&C cables within the same division, the staff agreed with GE's interpretation. The staff concludes that a design which meets the above described commitments as clarified will reasonably ensure that normal operation and/or failure of power circuits (Class 1E, associated, or non-Class 1E) will not adversely affect I&C circuits. The design, therefore, meets the above defined requirements of GDC 4 and 17 of 10 CFR

Part 50, Appendix A, and is acceptable. The staff's evaluation for the protection of circuits from EMI is addressed in Chapter 7.0 of this report. GE has included the above design commitments in Sections 8.3.3.1, 8.3.3.6.1.1, 8.3.3.6.1.2, 8.3.3.6.2.2.3, and 8.3.3.8.2 of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

In addition, based on the design commitment to meet the guidelines of IEEE 384-1981, GE indicated that administrative control of operations and maintenance activities will be used to control and limit introduction of potential hazards into cable chases and the control room area. Verification that GE included in a future SSAR amendment the statement that the COL applicant include these administrative controls in appropriate plant procedures was DFSER (SECY-92-349) COL Action Item 8.3.2.7-1. GE included this action item in Section 8.3.4.26 of SSAR Amendment 32, which is acceptable.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.3.2.7-1 related to the design description and the ITAAC for the separation of cables approaching and/or exiting cabinets/panels. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

8.3.2.8 Independence and Physical Separation of Equipment

This discussion addresses DSER (SECY-91-355) Open Item 36.

GE indicated in the draft SSAR revision dated April 3, 1992, and in discussions with the staff that Class 1E power supply and distribution systems, equipment, and components from the power supply through the power distribution panels of different Class 1E divisions will be separated by a 3-hour rated fire barrier and a missile barrier when the potential for missiles exist.

GDC 17 of 10 CFR Part 50, Appendix A, requires that Class 1E systems, equipment, and components have sufficient independence to perform their safety functions assuming a single failure. The staff concludes that a design which meets the above described commitment will ensure that any failure of or within one division of the Class 1E power system or its load group will not cause a loss of function in another division of the Class 1E power system. The design, therefore, meets the above defined requirements of GDC 17 of 10 CFR Part 50, Appendix A, and is acceptable.

Verification that GE provided the above design commitments in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.3.2.8-1. (Section 8.3.3.5 of this report addresses acceptable redundant circuits independence and protection from distribution system power panels to connected equipment loads which are not separated by fire and/or missile barriers.) GE included this commitment in Section 8.3.3.6.2.2.2 of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.3.2.8-1 related to the design description and the ITAAC for the independence/physical separation of equipment. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

8.3.2.9 Equipment, Cable, and Raceway Identification

This section addresses the staff's evaluation of GE's response to DSER (SECY-91-355) Open Item 39.

Identification of Power, Instrumentation, and Control Equipment, Cables, and Raceways

GE indicated in Section 8.3.1.3 of the draft SSAR revision dated April 3, 1992, that the ABWR electrical system design related to identification of power, control, and instrumentation systems, equipment, and components will meet the following commitments:

- The background of the nameplate for a division's equipment will be the same color as the electrical cable jacket markers and the cable raceway markers affiliated with that division.
- All exposed Class 1E and associated circuit raceways will be marked with the division color at 4.5 m (15 ft) intervals on; straight sections, at turning points, at points of entry to and exit from rooms and enclosed areas, at discontinuities, at pull boxes, and at origins and destinations of equipment.
- Class 1E and associated circuit raceways will be marked before their cables are installed.
- Before or during installation of all cables, for Class 1E and associated circuits, will be marked with the division color at intervals of approximately 1.5 m (5 ft).

- During installation cables for Class 1B and associated circuits that are routed in conduits will be marked with the division color at; points of entrance to and exit from conduits, at pull boxes, equipment, or enclosures where cables will or can be exposed. Such cables may not be marked at 1.5 m (5 ft) intervals inside conduits.
- All equipment, cables, and raceways will be marked in a manner of sufficient durability to be legible throughout the life of the plant and to facilitate initial verification that the installation conforms with the design separation criteria.

- All cables will be tagged with a permanent marker at each end with a unique identifying number (cable number) in accordance with the design drawings or cable schedule.

- The method used for identification will readily distinguish between different divisions of Class 1B systems, between Class 1B and non-Class 1B divisions, and between associated cables of different divisions.

- Associated cables will be uniquely identified as such by a longitudinal stripe or other color-coded method.

- The color of the cable marker for associated cables will be the same color as the related Class 1B cable.

- Individual conductors (located inside panels or cabinets) exposed by stripping the jacket will be color coded or color-tagged at intervals not to exceed 0.3 m (1 ft) such that their division will still be discernable. Exceptions are permitted for individual conductors within cabinets or panels where all wiring is unique to a single division. Any non-divisional cable within such cabinets will be marked appropriately to distinguish it from the divisional cables.

- Class 1B wire bundles or cables (located inside panels or cabinets) will be identified in a distinct permanent manner at a sufficient number of points to readily distinguish between Class 1B wiring of different divisions and between Class 1B and non-Class 1B wiring.

- For a cabinet or compartment containing only Class 1B wiring of a single division, no distinctive identification will be required.

GDC 17 of 10 CFR Part 50, Appendix A, requires that Class 1B systems, equipment, and components have sufficient independence to perform their safety functions assuming a single failure. The staff concludes that a design which meets the above described commitments will

Certified Design Material

Verification that GB provided the above design commitments in a future SSAR amendment was DFRSR (SBCY-92-349) Confirmatory Item 8.3.2.9-1. GB has included these commitments in Section 8.3.3.5 of SSAR Amendment 32, which is acceptable. Therefore, DFRSR Confirmatory Item 8.3.2.9-1 is resolved.

DFSR (SBCY-92-349) Open Item 8.3.2.9-1 related to the design description and the ITAAC for the identification of power, instrumentation, and control equipment, cables, and raceways. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

Identification of Neutron-Monitoring, Scram Solenoid, and dc Emergency Lighting Cables/Raceways

Section 8.3.1.3 of SSAR Amendment 10 indicated that cables of the neutron-monitoring system will be run in their own divisional conduits and cable trays separate from all other power, instrumentation, and control cables. Scram solenoid and dc emergency lighting cables will be similarly routed in their own conduits or cable trays separate from other cables. In addition, scram solenoid cables will be run in separate conduits for each rod scram group.

The following unique voltage class designations and markings will be used to help distinguish the neutron-monitoring and scram solenoid cables from other cable types:

- Neutron monitoring cables will be marked with a "VN" designation.
- Scram solenoid cables will be marked with a "VS" designation.

The staff concluded earlier in the review that the proposed identification of raceways and cables defined above does not meet the guidelines for the identification of raceways and cables defined by Section 6.1.2 of IEBE 384-1981 and position C10 of RG 1.75 (Rev. 2). The design did not include permanent, color raceway and cable markings to ensure that neutron monitoring, scram solenoid, and dc

lighting cables will be installed in their associated raceways in accordance with design basis protection and independence requirements. This was DFSER (SECY-92-349) Open Item 8.3.2.9-2.

In response to this item, GE indicated that the "VN" or "VS" designation would be superimposed on the divisional color markings on the cable and raceway. Similarly, GE indicated a "DCL" designation would be superimposed on the color mark for dc emergency lighting cables and their dedicated raceways. The staff concludes that these designations and color coding meet the guidelines of RG 1.75 (Rev. 2). This aspect of the design therefore ensures that cables will be installed in their designated raceways in accordance with design basis requirements and is acceptable. GE included this design information in Sections 8.3.3.5.1.3 and 9.5.3 of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.3.2.9-3 related to the design description and the ITAAC for the identification of neutron-monitoring, scram solenoid, and dc emergency lighting cables/raceways. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

8.3.3 Protection of Electrical Systems, Equipment, and Components

Protection of Class 1E cable systems from non-Class 1E cable systems by spatial separation or barrier is addressed in Section 8.3.2. Protection of Class 1E cable systems by isolation devices is addressed in Section 8.3.4.

8.3.3.1 Protection of Electric Penetrations

This section addresses DSER (SECY-91-355) Open Item 43.

Item 7 of Section 8.3.1.4.1.2 of SSAR Amendment 10 indicated that power circuits passing through electric penetration assemblies are protected against overcurrent by redundant interrupting devices. In addition, GE's response to Question 435.31(b) of SSAR Amendment 10 indicated that the ABWR design requires that redundant interrupting devices be provided for electrical circuits passing through containment penetrations, if the maximum available fault current (including failure of upstream devices) is greater than the continuous current rating of the penetration.

Based on these design requirements, GE indicated that the proposed design will include redundant interrupting devices

on all I&C circuits as well as power circuits that pass through containment when these circuits can produce sufficient energy (maximum available fault current) to exceed the current carrying capability of containment penetrations. When calculating maximum available fault current at the penetration, GE further indicated that current limiting devices will not be used in the calculation. For example, the worst case failure or shorting of the upstream or current limiting devices will be assumed as a given in the calculation. The staff concluded that the proposed design will include redundant protective devices (that is, current limiting and/or current interrupting) on all containment penetration circuits in accordance with guidelines in RG 1.63 (Rev. 3) as discussed below.

Based on the above, discussions with GE, and information presented in Section 8.3.4.4 of the draft SSAR revision dated April 3, 1992, GE indicated that protection of electrical penetrations will meet the following commitments:

- The thermal capability of all electrical conductors within containment penetrations will be preserved and protected by two independent devices which meet requirements of IEEE 603-1980.
- The two independent devices will be located in separate panels or will be separated by barriers.
- The two independent devices will be independent such that failure of one will not adversely affect the other.
- The two independent devices will not depend on the same power supply to accomplish their safety-related function of protecting the containment penetration.
- Analysis will demonstrate that the maximum available fault current in the event of failure of either of two devices (that is short or open between input and output of a current limiting device or protective device fails open or closed) will be less than the maximum continuous current capacity of the conductor within the penetration and the maximum continuous current capacity rating of the penetration.
- Fault current clearing-time curves of the electrical penetrations' primary and secondary current interrupting devices plotted against the thermal capability (I^2t) curve of the penetration will show proper coordination.
- A simplified, one-line diagram will show the location of the protective or current limiting devices in the penetration circuit, the maximum available fault current of the circuit, and specific identification and location of

Electric Power Systems

power supplies used to provide external control power for tripping primary and backup electrical penetration breakers (if utilized).

- The devices will be capable of being functionally tested and calibrated.

In addition, Section 5.2 of IEEE 308-1980 indicates that components, equipment, or systems required to provide some protective action, such as containment integrity protection, are covered by safety system requirements defined in IEEE 603-1980. Based on GE's commitment to meet the guidelines of IEEE 308-1980, GE indicated that the devices used to protect containment integrity will be covered by all safety system requirements defined in IEEE 603-1980.

GDC 50 of 10 CFR Part 50, Appendix A, requires that the reactor containment structure, including penetrations, be designed so that the containment structure can, without exceeding the design leakage rate, accommodate the calculated pressure, temperature, and other environmental conditions resulting from any loss-of-coolant accident. The staff concludes that a design that meets above described commitments will provide protection of containment electrical penetration such that a failure of a circuit (i.e., the single failure during a design basis event) and single failure of a device providing protection to containment penetrations will not cause loss of containment integrity. The design, therefore, meets the above defined requirements of GDC 50 of 10 CFR Part 50, Appendix A, and is acceptable.

Verification that GE provided the above design commitments in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.3.3.1-1. GE has included these commitments in Sections 8.3.1.2(2)(c), 8.3.3.6.1.2, and 8.3.3.7 of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

In addition, GE indicated that the protective devices will periodically be tested to demonstrate their functional capability to perform their required safety functions. Verification that GE included in a future SSAR amendment a statement that the COL applicant will include periodic test and calibration of protective devices in appropriate procedures was DFSER (SECY-92-349) COL Action Item 8.3.3.1-1. GE included this action item in Section 8.3.4.4 of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.3.3.1-1 related to the design description and the ITAAC for the protection of

electric penetrations. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

8.3.3.2 Design/Qualification of Electrical Equipment

Section 8.3.1.2.4 of the draft SSAR revision dated April 3, 1992, in response to DSER (SECY-91-355) Open Item 45, indicated that all Class 1E equipment is designed to operate in its normal service environment as well as in the environment expected in the area in which it is located during and after any design-basis event. In addition, by committing to meet the guidelines of IEEE 308-1980 (Table 1.8-21 of SSAR Amendment 17), GE indicated compliance with the requirements of Section 5.9, Equipment Qualifications of IEEE 308-1980. Section 5.9 of IEEE 308-1980 requires that all Class 1E power system equipment shall be qualified in accordance with IEEE 323-1974 to substantiate that it will be capable of meeting the performance requirements specified in the design basis.

Based on this commitment, information presented in the draft SSAR revision dated April 3, 1992, and discussions with the staff, GE indicated that each type of Class 1E equipment will be

- qualified by; analysis, successful use under similar conditions, or by actual test to demonstrate its ability to perform its function under normal and design-basis event environmental and operational conditions
- designed and qualified to survive the combined effects of temperature, humidity, radiation, and other conditions associated with a LOCA or other design-basis event environments at the end of their qualified and/or design life
- qualified to IEEE 344-1987, "Recommended Practices for Seismic Qualifications of Class 1E Equipment for Nuclear Power Generating Stations"
- qualified by test and/or analyzed to demonstrate its ability to meet its performance requirements during and following the design-basis seismic event
- located in seismic Category I structures
- seismically supported
- designed and qualified to operate within allowable design basis limits; (for example, able to operate for predetermined time when subject to voltage below 90 percent, to operate for a predetermined time when

voltage is below 70 percent, or to operate continuously when subjected to voltage variations of ± 10 percent of nominal).

All structures, systems, equipment, components, pipes and loads that are not Class 1E and whose failure could possibly prevent Class 1E systems, equipment, components, and circuits including connected loads from performing their required safety function will be appropriately designed and qualified to not fail in the normal and design-basis event environment for which the structures, systems, equipment, components, pipes and loads will be expected to function. In addition, variations of voltage, frequency, and waveform in the Class 1E power systems, during any mode of plant operation, will not degrade the performance of any safety-related system load below an acceptable level. The dc system equipment and loads will be designed and qualified to perform their required safety-related function while operating with voltages between 100 to 140 volts at the dc system's 125-volt distribution panels.

GDC 2 and 4 of 10 CFR Part 50, Appendix A, requires that Class 1E systems, equipment, and components be designed (1) to withstand the effects of natural phenomena without loss of capability to perform their safety functions, (2) to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, and (3) be appropriately protected against dynamic effects. GDC 17 of 10 CFR Part 50, Appendix A, requires that the Class 1E systems, equipment, and components have sufficient independence to perform their safety functions assuming a single failure. The staff concludes that a design that meets the above commitments will provide protection to Class 1E systems, equipment, and components during design basis events such that there will be reasonable assurance that Class 1E systems will be capable of performing their required function. The electrical system design, therefore, meets the above defined requirements of GDC 2, 4, and 17 of 10 CFR Part 50, Appendix A, and is acceptable.

Verification that GE provided the above design commitments in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.3.3.2-1. In response to this item, GE further modified their design to indicate that the Class 1E systems, equipment, and components conform to seismic Category I requirements and are housed in seismic Category I structures in accordance with the above design information except for some control sensors associated with the reactor protection system and the leak detection system which are housed in the turbine building which is not a seismic Category I structure. For these exceptions, GE has indicated that the feeders between Class 1E systems located in seismic Category I structures

(safety class structures) and systems that are not located in seismic Category I structures will be provided with Class 1E protective devices (such as coordinated circuit breakers) located in a seismic Category I structure. By committing to meet the guidelines of IEEE 308-1980 (Table 1.8-21) of SSAR Amendment 17), GE indicated compliance with the requirements of Section 5.2 of IEEE 308-1980. Section 5.2 of IEEE 308-1980 requires that components, equipment, or systems utilized to provide isolation protection are covered by safety system design requirements defined in IEEE 603-1980. The staff concluded that the Class 1E classification for the protective devices utilized to provide an isolation protection function in the proposed modified design meets the guidelines of Section 5.2 of IEEE 308-1980. The staff concludes that a design that meets the above described commitments will also provide protection to Class 1E systems such that there will be reasonable assurance that Class 1E systems will be capable of performing their required function. The electrical system design, therefore, meets the above defined requirements of GDC 2, 4, and 17 of 10 CFR Part 50, Appendix A, and is acceptable. GE has included the above design commitments in Sections 3.11.2, 8.1.3.1.1.1, 8.3.1.1.5, 8.3.1.1.7, 8.3.2.1.3.1, and 8.3.3.4 of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.3.3.2-1 related to the design description and the ITAAC for the design/qualification of Class 1E electrical equipment. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

8.3.3.3 Seismic Qualification of Light Bulbs

In response to DSER (SECY-91-355) Open Item 45, GE provided draft SSAR revision dated April 3, 1992, Section 8.3.2.2.2, identifying an exception to the requirement that all Class 1E equipment is seismically qualified. GE indicated that the safety-related dc standby lighting system is powered from the Class 1E dc system and that the lighting system circuits from the Class 1E dc system power source to the lighting fixtures will be treated as Class 1E circuits (that is, these circuits will be classified as associated) and will be routed in seismic Category I raceways. The lighting fixtures themselves will not be seismically qualified, but will be seismically supported. The bulbs cannot be seismically qualified.

To justify this exception, GE indicated that bulbs can only fail open, and, therefore, do not represent a hazard to the Class 1E power source.

Electric Power Systems

Based on subsequent discussions with GE, the staff determined that lighting fixtures will be seismically qualified. Light bulbs may fail during and/or following a seismic event thereby extinguishing the light; however, the light bulbs will be replaceable. (The staff's evaluation of lighting requirements is in Section 8.3.5 of this report.) In addition, bulbs will not fail in a manner that could cause failure of other safety-related systems and will not become a hazard to personnel or safety-related equipment during or following a seismic event.

GDC 4 of 10 CFR Part 50, Appendix A, requires that Class 1E systems, equipment, and components be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents. GDC 17 of 10 CFR Part 50, Appendix A, requires that the Class 1E systems, equipment, and components have sufficient independence to perform their safety functions assuming a single failure. The staff concludes that lighting circuits that are treated as Class 1E except for the seismic qualification of light bulbs and which include seismically qualified light bulb fixtures will provide protection to Class 1E systems such that there will be reasonable assurance that the lighting systems will not prevent the Class 1E systems from performing their required function. Consequently, the design meets the above defined requirements of GDC 4 and 17 of 10 CFR Part 50, Appendix A, and is acceptable.

Verification that GE provided the above design commitments in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.3.3.3-1. In response to this item, GE indicated that lighting fixtures are seismically supported but not seismically qualified as indicated above. GE further indicated that overcurrent protective devices and their coordination would provide protection and isolation of the Class 1E power supply from possible failure of non-seismically qualified fixtures during a seismic event. The staff concluded that Class 1E systems will be adequately protected from failure of the non-seismically qualified fixtures as well as the non-seismically qualified light bulbs during a seismic event. By committing to meet the guidelines of IEEE 308-1980 (Table 1.8-21 of SSAR Amendment 17), GE indicated compliance with the requirements of Section 5.2 of IEEE 308-1980. Section 5.2 of IEEE 308-1980 requires that components, equipment, or systems utilized to provide isolation protection are covered by safety system design requirements defined in IEEE 603-1980. The staff concluded that the overcurrent protective devices and their coordination which are utilized to provide an isolation protection function to the Class 1E system from failure of the non-seismically qualified fixtures will meet the guidelines of Section 5.2 of IEEE 308-1980 and thus will

meet safety system design requirements defined in IEEE 603-1980.

The staff concludes that lighting circuits that are treated as Class 1E except for seismic qualification of light bulbs and fixtures and include protective devices which meet safety system requirements will provide protection to Class 1E systems such that there will be reasonable assurance that the lighting systems will not prevent the Class 1E systems from performing their required function. Consequently, the design meets the above defined requirements of GDC 4 of 10 CFR Part 50, Appendix A, and is acceptable. GE has included the above design commitments in Sections 8.3.2.2.2, 9.5.3.1.1, 9.5.3.2.2.1, and 9.5.3.2.3.1 of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.3.3.3-1 related to the design description and the ITAAC for the protection of Class 1E systems from the non-seismically qualified light fixtures and bulbs. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

8.3.3.4 Submergence

Item (6) of Section 8.3.1.4.2.3.2 of SSAR Amendment 10 stated that any RPS or ESF electrical equipment and/or raceway located in the suppression pool level swell zone will be designed to satisfactorily complete its function before being rendered inoperable due to exposure to the environment created by the level phenomena. In response to staff Question 435.36 of SSAR Amendment 10, GE identified electrical equipment that may be submerged as a result of suppression pool level swell phenomena or as a result of a LOCA. GE further indicated that the design specifications associated with this electric equipment would require that electrical terminations be sealed such that equipment operation would not be impaired by submersion. However, GE did not specifically address the qualification of this equipment in accordance with the guidelines of Section 4.7 of IEEE 308-1974.

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

The staff was concerned that equipment failure due to submergence could adversely affect the safe operation of

Electric Power Systems

design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

8.3.3.5 Redundant Class 1B Systems Subject to Common Design Basis Environments

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

The draft information provided by GE on September 4, 1991, indicated that cables and other electric equipment will not be designed and qualified to perform their safety function while being subjected to the direct impingement of fire suppressant. In justifying this lack of design and qualification, GE indicated that redundant divisions are provided. In the event that the cable system or equipment in one division fails due to fire suppressant impingement, and single failure occurs in a second division, safe shutdown of the plant (the required minimum safety function) can be achieved by the third division. This was DSEB (SECY-91-355) Open Item 47.

In the draft SSAR revision dated April 3, 1992, GE indicated that where fire suppressant impinged on cables of more than one division, each case had been analyzed and found to be acceptable for the worst case failure mode.

After reviewing information presented in Chapter 8 of the draft SSAR revision dated April 3, 1992, and Section 9A.5 through SSAR Amendment 20 to the SSAR, the staff was unable to reach conclusions as to the acceptability of the level of protection to be afforded Class 1B power systems due to failure of redundant Class 1B components that may be subjected to environments of the same design-basis event (including fire, fire suppressant, and non-seismic structures) for which they may not be designed or qualified. Information and design commitments presented in Chapter 8 and Section 9A.5 were found to be inconsistent. This was DSEB (SECY-92-349) Open Item 8.3.3.5-1.

In response to this item, GE provided the results of an analysis in SSAR Section 9A.5. These results assured that when common mode failure of redundant safety systems and their supporting Class 1B systems occurs because of (fire, fire suppressant impingement, or seismic event), sufficient remaining safety systems not affected by these

the plant and could adversely affect Class 1B power sources serving this equipment. This was DSEB (SECY-91-355) Open Item 46.

GE indicated in Section 8.3.1.4.2.3.2 of the draft SSAR revision dated April 3, 1992, that the only Class 1B equipment located in the suppression pool level well zone will be suppression pool temperature monitors, which have their terminations sealed such that their operation will not be impaired by submergence due to pool swell or a LOCA. Consistent with their Class 1B status, these devices will also be qualified to the requirements of IHB 323 for the environment in which they are located.

GE also indicated that all Class 1B equipment (including affiliated systems and component parts) subject to submergence in the suppression pool level well zone will either be designed and qualified to perform its required safety function without failing while submerged. If this is not possible, all Class 1B equipment will be appropriately protected from submergence and will be appropriately designed and qualified to perform its required safety function without failing in the normal and design-basis event environment for which the equipment is expected to operate.

GDC 4 of 10 CFR Part 50, Appendix A, requires that Class 1B systems, equipment, and components be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents. GDC 17 of 10 CFR Part 50, Appendix A, requires that the Class 1B systems, equipment, and components have sufficient independence to perform their safety function assuming a single failure. The staff concludes that a design which meets the above described commitments will provide reasonable assurance that the components subject to submergence will be capable of performing their required function. The design, therefore, meets the above defined requirements of GDC 4 and 17 of 10 CFR Part 50, Appendix A, and is acceptable.

Verification that GE provided the above design commitments in a future SSAR amendment was DSEB (SECY-92-349) Confirmatory Item 8.3.3.4-1. GE has included these commitments in Section 8.3.3.6.2.3.2(6) of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

Certified Design Material

DSEB (SECY-92-349) Open Item 8.3.3.4-1 related to the design description and the ITAAC for the submergence of Class 1B electrical equipment in the suppression pool well zone. The adequacy and acceptability of the ABWR

Electric Power Systems

same events are available to accomplish the required safety function. The results of this analysis, indicated that Class 1E power systems will be protected by circuit protective devices and their coordination or current limiting devices, in order to assure the continued operation of Class 1E systems in accordance with their required safety function. By committing to meet the guidelines of IEEE 308-1980 (Table 1.8-21 of SSAR Amendment 17), GE indicated compliance with the requirements of Section 5.2 of IEEE 308-1980. Section 5.2 of IEEE 308-1980 requires that components, equipment, or systems utilized to provide isolation protection are covered by safety system design requirements defined in IEEE 603-1980. The staff concluded that the overcurrent protective devices and their coordination or current limiting devices which are utilized to provide an isolation protection function to the Class 1E systems will meet the guidelines of Section 5.2 of IEEE 308-1980 and thus will meet safety system design requirements defined in IEEE 603-1980.

GDC 4 of 10 CFR Part 50, Appendix A, requires that Class 1E systems, equipment, and components be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents. GDC 17 of 10 CFR Part 50, Appendix A, requires that the Class 1E systems, equipment and components have sufficient independence to perform their safety function assuming a single failure. The staff concludes that a design that meets the above commitments will adequately protect Class 1E systems such that there will be reasonable assurance that redundant system will not fail. Consequently, the design meets the above defined requirements of GDC 4 and 17 of 10 CFR Part 50, Appendix A, and is acceptable. GE has included the results of this analysis and the above commitments in Sections 8.3.1.2(2)(c) and 9A.5 of SSAR Amendment 32, which is acceptable.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.3.3.5-2 related to the design description and the ITAAC for the protection of redundant Class 1E systems subject to common design basis environments. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

8.3.3.6 Associated Circuits

Based on discussions with GE and information presented in Section 8.3.1.1.5.1 of the draft SSAR revision dated April 3, 1992, in response to DSER (SECY-91-355) Open Item 38, GE indicated that the ABWR electrical system

design related to associated circuits will meet the following commitments:

- Associated circuits will remain with or be physically separated in the same manner as those Class 1E circuits with which they are associated;

or

Associated circuits, will remain with or be physically separated in the same manner as those Class 1E circuits with which they are associated, from the Class 1E equipment to and including an isolation device.

- Associated circuits (including their isolation devices or their connected loads without isolation devices) will be subject to all requirements placed on Class 1E circuits.
- Non-Class 1E circuits powered from a Class 1E power supply will be limited to power circuits related to the FMCRDs and lighting systems.

GDC 4 of 10 CFR Part 50, Appendix A, requires that Class 1E systems, equipment, and components be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents. GDC 17 of 10 CFR Part 50, Appendix A, requires that the Class 1E systems, equipment, and components have sufficient independence to perform their safety function assuming a single failure. The staff concludes that a design which meets the above described commitments will provide adequate protection and independence for Class 1E systems. The design, therefore, meets the above defined requirements of GDC 4 and 17 of 10 CFR Part 50, Appendix A, and is acceptable.

Verification that GE provided the above design commitments in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.3.3.6-1. GE has included the above design commitments in Sections 8.3.3.1 and 8.3.1.1.1 of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.3.3.6-1 related to the design description and the ITAAC for associated circuits. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

The staff determined that a commitment was required in the design description of electrical systems that states that non-Class 1E circuits connected to the Class 1E system

shall be limited to circuits in the FMCRD and lighting subsystems. This was Open Item F8.3.3.6-1 identified in the advance version of the SER. GE has included this commitment in the design description of Section 2.12.1 and 2.12.12 of the certified design material. The staff finds this acceptable. This resolved Open Item F8.3.3.6-1.

8.3.3.7 Diesel Generator Protective Relaying Bypass

Section 8.3.1.1.6.4 of SSAR Amendment 10 indicated that the following identified protective relaying will trip the diesel generator and will be retained under LOCA conditions. This relaying included the generator differential, bus differential, engine over speed, low diesel cooling water pressure (two out of two sensors), and low differential pressure of secondary cooling water (two out of two sensors). Other diesel generator protective trips will be bypassed during LOCA conditions.

GE responded to DSER (SECY-91-355) Open Item 49 in Section 8.3.1.1.6.4 of the draft SSAR revision dated April 3, 1992, where they indicated that only the generator differential relays and engine overspeed trip would be retained under accident conditions. Other protective relays, such as loss of excitation, antimooring (reverse power) overcurrent voltage restraint, low jacket water pressure, high jacket water temperature, and low lube oil pressure are automatically removed from the tripping circuits during LOCA conditions.

By committing to meet the guidelines of IEEE 308-1980 (Table 1.8-21 of SSAR Amendment 17), GE indicated compliance with the requirements of Section 5.2 of IEEE 308-1980. Section 5.2 of IEEE 308-1980 requires that components, equipment, or systems that provide some protection to Class 1E systems are covered by safety system design requirements defined in IEEE 603-1980. The staff concluded that the diesel generator protective relaying bypass circuitry which are utilized to provide protection to the Class 1E systems will meet the guidelines of Section 5.2 of IEEE 308-1980 and thus will meet safety system design requirements defined in IEEE 603-1980. By committing to meet the guidelines of RG 1.9 (Rev. 3) (Table 1.8-20 of SSAR Amendment 33), GE indicated compliance with the requirements of Position C-1.8 of the RG. Position C-1.8 requires that the diesel generator protective relay bypass system design include the capability for testing the status and operability of the bypass circuitry, for alarming in the control room for abnormal values of all bypass parameters, and for resetting the trip bypass function manually (automate reset is not acceptable). Based on the above commitment and the commitment to meet the guidelines of Position C-1.8 of RG 1.9 (Rev. 3) and Section 5.2 of IEEE 308-1980 contained in Section 8.3.1.2.1(2)(b) of the draft SSAR

revision dated April 3, 1992, and Table 1.8-21 of SSAR Amendment 17, information presented in the draft SSAR revision dated April 3, 1992, and discussions, GE's indicated that:

- the design of the bypass circuitry will meet all the requirements of IEEE 603-1980
- abnormal values of all bypassed parameters will be alarmed in the control room so that the control room operator can react appropriately to the abnormal condition on the diesel generator unit
- the trip bypass function will be capable of being reset manually (capability for automatic reset is not acceptable)
- the protective relaying and its bypass circuitry will have the capability to be tested periodically

GDC 4 of 10 CFR Part 50, Appendix A, requires that Class 1E systems, equipment, and components be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents. GDC 17 of 10 CFR Part 50, Appendix A, requires (1) that offsite and onsite power systems have sufficient capacity and capability to permit safety systems to perform their required safety function and (2) that the Class 1E systems, equipment and components have sufficient independence and testability to perform their safety functions assuming a single failure. GDC 18 of 10 CFR Part 50, Appendix A, requires that electric power systems important to safety be designed to permit appropriate periodic inspection and testing of important areas and features. The staff concludes that a design which meets the above described commitments will provide reasonable assurance that the protective relaying (to be installed on Class 1E diesel generators to protect the diesel generator from failure) will be bypassed during accident conditions so that the diesel generator will not be prevented from performing its required safety function under accident conditions due to operation or failure of the protective scheme. Consequently, the design meets the above defined requirements of GDC 4, 17, and 18 and is acceptable.

Verification that GE provided the above design commitments in a future SSAR amendment was DFER (SECY-92-349) Confirmatory Item 8.3.3.7-1. GE included this information in Section 8.3.1.1.6.4 of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

In addition, GE indicated that the protective relays and their bypass circuitry will be periodically tested.

Electric Power Systems

Verification that GE specified in a future SSAR amendment that the COL applicant will include periodic testing of protective relays and their bypass circuitry in appropriate procedures was DFSER (SECY-92-349) COL Action Item 8.3.3.7-1. GE included this action item in Section 8.3.4.22 of SSAR Amendment 32, which is acceptable.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.3.3.7-1 related to the design description and the ITAAC for the diesel generator protective relaying and their bypass. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, the above open item is resolved.

8.3.3.8 Thermal Overloads

GE's response to Question 435.60 in SSAR Amendment 10 indicated that thermal overload protection for Class 1E motor operated valves (MOVs) is in effect only when the MOVs are in test mode. The thermal overload protection is bypassed at all other times by means of closed contacts in parallel with the thermal overload contacts.

GE responded to DSER (SECY-91-355) Open Item 50, in Section 8.3.1.2.1(2)(g) and Section 8.3.2.2.2(2)(f) of the draft SSAR revision dated April 3, 1992, where it was indicated that the thermal overload protection for Class 1E MOVs will be in effect during normal plant operation but the overloads will be bypassed under accident conditions as specified by Position 1.(b) of RG 1.106 (Rev. 1).

By committing to meet the guidelines of IEEE 308-1980 (Table 1.8-21 of SSAR Amendment 17), GE indicated compliance with the requirements of Section 5.2 of IEEE 308-1980. Section 5.2 of IEEE 308-1980 requires that components, equipment, or systems that provide some protection to Class 1E systems are covered by safety system design requirements defined in IEEE 603-1980. Based on this commitment, the staff concluded that the thermal overload protection bypass circuitry for Class 1E MOVs (which is utilized to provide protection to motors of MOVs in the Class 1E system) will meet the guidelines of Section 5.2 of IEEE 308-1980 and thus the safety system design requirements defined in IEEE 603-1980. In addition, GE indicated that the thermal overload and its bypass circuitry will have the capability to be tested periodically.

GDC 4 of 10 CFR Part 50, Appendix A, requires that Class 1E systems, equipment, and components be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal

operation, maintenance, testing, and postulated accidents. GDC 17 of 10 CFR Part 50, Appendix A, requires (1) that offsite and onsite power systems have sufficient capacity and capability to permit safety systems to perform their required safety function and (2) that the Class 1E systems, equipment, and components have sufficient independence to perform their safety function assuming a single failure. GDC 18 of 10 CFR Part 50, Appendix A, requires that Class 1E power systems be designed to permit appropriate periodic inspection and testing of important areas and features. The staff concludes that a design which meets the above described commitments will provide reasonable assurance that the thermal overload protection to be installed on Class 1E motor operated valves will be bypassed during accident conditions so that the Class 1E valve motor will not be prevented from performing its required safety function under accident conditions due to operation or failure of the thermal overload devices. Consequently, the design meets the above defined requirements of GDC 4, 17, and 18 and is acceptable.

Verification that GE provided the above design commitments in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.3.3.8-1. GE has included these commitments in Sections 8.3.1.2(2)(g) and 8.3.2.2.2(2)(f) of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

In addition, GE indicated that the thermal overloads and their bypass circuitry will periodically be tested. Verification that GE specified in a future SSAR amendment that the COL applicant will include periodic testing of thermal overloads and their bypass circuitry in appropriate procedures was DFSER (SECY-92-349) COL Action Item 8.3.3.8-1. GE included this action item in Section 8.3.4.24 of SSAR Amendment 32, which is acceptable.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.3.3.8-1 related to the design description and the ITAAC for the thermal overloads. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

8.3.3.9 Breaker Coordination

Section 8.3.1.1.2.1 of SSAR Amendment 10 stated that tripping of the Class 1E bus feeder breaker is normal for faults that occur on its Class 1E loads. The staff disagreed with this statement. Class 1E load breakers should be coordinated with the Class 1E bus feeder breaker so that faults which occur on its Class 1E loads will, to the extent

possible, not cause the bus feeder breaker to trip. A design which utilizes coordinated breakers minimizes the potential for loss of safety-related systems. This was DSER (SECY-91-355) Open Item 51.

The draft information provided by GE on September 4, 1991, revised the SSAR to delete the statement that tripping of the bus supply breaker is normal for faults that occur on its Class 1E loads. GE further indicated that the Class 1E load and bus supply breakers will be coordinated.

GDC 4 of 10 CFR Part 50, Appendix A, requires that Class 1E systems, equipment, and components be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents. GDC 17 of 10 CFR Part 50, Appendix A, requires (1) that offsite and onsite power systems have sufficient capacity and capability to permit safety systems to perform their required safety function and (2) that the Class 1E systems, equipment, and components have sufficient independence to perform their safety function assuming a single failure. The staff concludes that a design which meets the above commitment will minimize to the extent practicable the effect of single Class 1E components or equipment failure. The design, therefore, meets the above defined requirements of GDC 4 and 17 of 10 CFR Part 50, Appendix A, and is acceptable.

Verification that GE provided the above design commitment in a future SSAR amendment was DFSE (SECY-92-349) Confirmatory Item 8.3.3.9-1. GE has included these commitments in Section 8.3.1.1.1 of SSAR Amendment 32, which is acceptable.

8.3.3.10 Protective Relaying

This section addresses DSER (SECY-91-355) Open Item 52.

Experience with protective relay applications has established that relay trip setpoints will drift with conventional relays. Setpoint drift at nuclear power plants has resulted in premature tripping of redundant, safety-related pump motors when they were required to be operative. While the staff recognizes the basic need for proper fault protection for feeders and equipment (and while such protection may be required for some design-basis events such as fire), the total non-availability of redundant safety systems due to spurious trips of protective relays is not acceptable. GE responded to this concern (Question 435.58 of SSAR Amendment 10) and indicated that loads, such as motors, will be designed with sufficient current carrying capability or overload margins so that setpoints of protective devices can be set sufficiently above

the operating current point of loads to allow for setpoint drift. The use of loads, such as motors, with sufficient overload margins resolves the staff's concern if one assumes the following:

- Specific design parameters clearly define the overload margin requirements with respect to; protective device trip setpoints, the margin between the trip setpoint and operating current point of loads, setpoint drift, and the margin between the trip setpoint and overload rating of loads.
- The protective device trip setpoint is periodically verified and calibrated.
- The protective device is periodically subjected to a functional test to demonstrate that it does not trip at its design rating (the normal operating current of load plus margin) and that it does trip when subjected to a fault current.

The staff was concerned earlier in the review that the ABWR design may not satisfy the above assumptions.

By committing to meet the guidelines of IEEE 308-1980 (Table 1.8-21 of SSAR Amendment 17), GE indicated compliance with the requirements of Section 5.2 of IEEE 308-1980. Section 5.2 of IEEE 308-1980 requires that components, equipment, or systems that have no direct safety function and are only provided to increase the availability or reliability of the Class 1E power systems shall meet those safety system design requirements defined in IEEE 603-1980 to assure that those components, equipment, and systems do not degrade the Class 1E power system below an acceptable level. Based on the design commitment to meet the guidelines of Section 5.2 of IEEE 308-1980, information presented in the draft SSAR revision dated April 3, 1992, and discussions, GE indicated that:

- protective relaying design will meet the above defined assumptions (that is, there will be protective device trip setpoint margin and capability to functionally test and calibrate the protective relaying)
- protective relaying—as well as all other components, equipment, and systems within the Class 1E power system (that have no direct safety function and are only provided to increase the availability or reliability of the Class 1E power systems) including the diesel generator protective relaying and thermal overload protective devices which are bypassed during accident conditions—will meet those requirements of IEEE 603-1980 that assure that the consequences of

Electric Power Systems

any operation or failure is acceptable to the Class 1E power system

GDC 4 of 10 CFR Part 50, Appendix A, requires that Class 1E systems, equipment, and components be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents. GDC 17 of 10 CFR Part 50, Appendix A, requires (1) that offsite and onsite power systems have sufficient capacity and capability to permit safety systems to perform their required safety function and (2) that the Class 1E systems, equipment, and components have sufficient independence and testability to perform their safety functions assuming a single failure. GDC 18 of 10 CFR Part 50, Appendix A, requires that Class 1E power systems be designed to permit appropriate periodic inspection and testing of important areas and features. The staff concludes that a design that meets the above commitments will assure that when these components, equipment, or systems are used that have no direct safety function their operation or failure will not significantly reduce the capability of the Class 1E power system from performing its safety function when required. The design, therefore, meets the above defined requirements of GDC 4, 17, and 18 of 10 CFR Part 50, Appendix A, and is acceptable.

Verification that GE provided the above design commitments in a future SSAR amendment was Confirmatory Item 8.3.3.10-1. GE has included these commitments in Sections 8.3.1.2(2)(c) and 8.3.2.2.2(2)(b) of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

In addition, GE indicated that the protective relaying will periodically be tested. Verification that GE specified in a future SSAR amendment that the COL applicant will include these periodic tests in appropriate plant procedures was DFSE (SECY-92-349) COL Action Item 8.3.3.10-1. GE included this action item in Section 8.3.4.27 of SSAR Amendment 32, which is acceptable.

Certified Design Material

DFSE (SECY-92-349) Open Item 8.3.3.10-1 related to the design description and the ITAAC for protective relaying. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

8.3.3.11 Fault Interrupting Capacity

Section 8.3.1.1.5.2(4) of SSAR Amendment 10 stated that the interrupting capacity of switchgear, load centers,

MCCs, and distribution panels is compatible with the short-circuit current available at the Class 1E buses. It was not clear whether the interrupting capacity of this equipment would be equal to or greater than the maximum available fault current to which it would be exposed for all modes of operation (for example, with the diesel generator operating in parallel with the grid).

Section 8.3.1.1.5.2(4) of the draft SSAR revision dated April 3, 1992, in response to DSER (SECY-91-355) Open Item 53, indicated that the interrupting capacity of switchgear, load centers, MCCs, and distribution panels will be equal to or greater than the maximum available fault current to which the equipment is exposed under all modes of operation.

GDC 4 of 10 CFR Part 50, Appendix A, requires that Class 1E systems, equipment, and components be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents. GDC 17 of 10 CFR Part 50, Appendix A, requires (1) that offsite and onsite power systems have sufficient capacity and capability to permit safety systems to perform their required safety function and (2) that the Class 1E systems, equipment, and components have sufficient independence and testability to perform their safety functions assuming a single failure. The staff concludes that a design which meets the above described design commitments will have sufficient capacity and capability to interrupt the worst case fault. The design, therefore, meets the above defined requirements of GDC 4 and 17 and is acceptable.

Verification that GE provided the above design commitments in a future SSAR amendment was DFSE (SECY-92-349) Confirmatory Item 8.3.3.11-1. GE has included these commitments in Section 8.3.1.1.5(4) of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

Certified Design Material

DFSE (SECY-92-349) Open Item 8.3.3.11-1 related to the design description and the ITAAC for the fault interrupting capacity of Class 1E protective equipment. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

8.3.3.12 Control of Design Parameters for Motor Operated Valves

Valve problems such as excess friction which may be caused by excessively tight packing can result in an operational condition where the current drawn will exceed

the design rating or capability of the insulation system used in the valve motor winding. Operating experience has shown that excessive current, if undetected during operation, can cause premature or unexpected failure when the valve is next operated. The ABWR SSAR did not present methods, design provisions, alarms, or procedures to ensure that the valve motor will not be operated with excessive currents without operator knowledge (or will always be operated within their design limits).

The draft information provided by GE on September 4, 1991, indicated that thermal overloads will provide protection at all times for non-Class 1E MOVs and will provide protection during testing or maintenance for Class 1E MOVs. At all other times, the Class 1E MOVs will not be protected. The staff was concerned by this lack of protection for Class 1E MOVs. This was DSER (SECY-91-355) Open Item 54.

Section 8.3.1.2.1(2)(g) of the draft SSAR revision dated April 3, 1992, indicated that Class 1E MOVs which are required to open and/or close to satisfy their safety function, will have the thermal overload protective device on the valves' motor in force during normal plant operation. The thermal overload protective device for these valves will be bypassed under accident conditions provided that safety function completion is not jeopardized or that other safety systems are not degraded as per Regulatory Position 1.(b) of Revision 1 of RG 1.106.

GDC 4 of 10 CFR Part 50, Appendix A, requires that Class 1E systems, equipment, and components be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents. GDC 17 of 10 CFR Part 50, Appendix A, requires (1) that offsite and onsite power systems have sufficient capacity and capability to permit safety systems to perform their required safety function and (2) that the Class 1E systems, equipment, and components have sufficient independence and testability to perform their safety functions assuming a single failure. The staff concludes that a design, which keeps the thermal overload in force during normal plant operation as well as during test and maintenance in accordance with the above described commitments, will provide reasonable assurance that the MOV will not be operated with excessive currents without operator knowledge (or will be operated within their design limits). The design, therefore, meets the above defined requirements of GDC 4 and 17 of 10 CFR Part 50, Appendix A, and is acceptable.

Verification that GE provided the above design commitments in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.3.3.12-1. GE

included these commitments in Sections 8.3.1.2(2)(g) and 8.3.2.2.2(2)(f) of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.3.3.12-1 related to the design description and the ITAAC for the control of design parameters for MOVs. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

8.3.3.13 Protection of Cable Systems from Internally Generated Fires

Section 8.3.3.2 of SSAR Amendment 10 indicated that spatial separation is used as a method of preventing the spread of fire between adjacent cable trays of different divisions (for example, inside primary containment). The design objective should be to separate cable trays of different divisions with structural fire barriers such as floors, ceilings, and walls. Where such barriers are not possible, divisional trays should be separated spatially by .9 m (3 ft) horizontally and 1.5 m (5 ft) vertically. Where this .9 by 1.5 m (3 by 5 ft) spatial separation is not possible, fire-rated barriers are used to separate divisional cable trays.

GDC 17 of 10 CFR Part 50, Appendix A, requires that the onsite electric distribution systems have sufficient independence to perform their safety function assuming a single failure.

For a fire initiated by a cable fault within one division, the staff concludes that a design which meets the above described commitments will provide reasonable assurance that a fire in one division will not propagate to a redundant division. The design, therefore, meets the above defined requirements of GDC 17 of 10 CFR Part 50, Appendix A, and is acceptable. This was DSER (SECY-91-355) Open Item 24.

In the draft SSAR revision dated April 3, 1992, GE revised Section 8.3.3.2 of SSAR Amendment 10 to indicate that separation will be achieved by using totally enclosed raceways separated by a least 2.54 cm (1 in.) when spacial separation is less than .9 by 1.5 m (3 by 5 ft). The staff concludes that a design which meets this commitment will provide reasonable assurance that a fire initiated in one division will not propagate to a redundant division. Consequently, the design meets the above defined requirements of GDC 17 of 10 CFR Part 50, Appendix A, and is acceptable.

Electric Power Systems

Verification that GE has provided the above design commitment in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.3.3.13-1. GE has included these commitments in Section 8.3.3.8.2 of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.3.2.5-1 related to the design description and the ITAAC for separation of raceways. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

8.3.3.14 Electrical Protection for Scram and MSIV solenoids Electrical Protection Assemblies (EPAs) (The need for EPAs was established as part of the staff's evaluation of I&C Systems. I&C Systems are addressed in Chapter 7 of this SER.)

A generic letter issued to all operating BWRs on September 24, 1980, requires two independent EPAs on the output of RPS power supplies. Two EPAs are required to satisfy the single-failure criterion for non-fail-safe type failures, which may be caused by under-voltage, over-voltage, and under-frequency conditions.

GE's response to Question 435.7 included in SSAR Amendment 10 indicated that EPAs will not be used in the ABWR design because of special design features. These special features included voltage and frequency monitoring, automatic transfer of power supply input sources when the voltage or frequency exceeds pre-established limits, control room alarm for abnormal conditions, operator action in response to alarm of abnormality, and design and qualification of equipment to preclude failure after operation for a period of time under the allowable abnormality of voltage and frequency.

The staff determined that these special features should provide reasonable assurance that any abnormality in voltage and frequency (which can cause failure of fail-safe-type equipment) will be promptly disconnected by alarms and operator action. The special features, however, do not meet the single failure criterion. Failure of the special features to alarm or failure of the operator to take prompt appropriate action are single failures which may cause a non-fail-safe type failure. The capability to scram the reactor could thus be compromised. This was DSER (SECY-91-355) Open Item 55.

Based on discussions, GE indicated that one EPA will be installed in each of the distribution circuits between the CVCF power supply and the RPS scram and main steam isolation valve (MSIV) solenoid valves (the fail-safe-type equipment). The CVCF abnormality in voltage or frequency alarm will be a Class 1E circuit and the CVCF alarm system and EPAs will be designed with the capability of being tested periodically.

The staff concludes that single failure of the EPA or the Class 1E CVCF power supply will not cause a non-fail-safe type failure of RPS scram or MSIV solenoid valves and is acceptable.

Verification that GE provided the above design commitments in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.3.3.14-1. GE included this information in Sections 8.3.1.1.4.2.1 and 8.3.1.1.4.2.2 of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

In addition, GE indicated that the CVCF alarm system and the EPA will be tested periodically. Verification that GE specified in a future SSAR amendment that the COL applicant will include these periodic tests in appropriate plant procedures was DFSER (SECY-92-349) COL Action Item 8.3.3.14-1. GE included this action item in Section 8.3.4.28 of SSAR Amendment 32, which is acceptable.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.3.3.14-1 was related to the design description and the ITAAC for the electrical protection for scram and MSIV solenoids. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

8.3.3.15 Safety Bus Grounding

On every bus shown in Figure 8.3-1 of the draft SSAR revision dated April 3, 1992, there is shown one circuit connected to ground through a circuit breaker. The circuit breaker or bus grounding device provides a safety ground on buses during maintenance operations. The bus grounding device includes the following interlocks

- under-voltage relays must be actuated
- related breakers must be in the disconnect position
- voltage for bus instrumentation must be available

The staff agreed that the proposed grounding device should be included in the design because it may be an important protection enhancement for personnel performing maintenance.

nance on Class 1E buses. The staff was concerned, however, that the proposed interlocks may not be sufficient to prevent inadvertent closing of the device during non-maintenance operation. This was DSER (SECY-91-355) Open Item 44.

GE indicated that annunciation will be provided in the design to alarm in the control room whenever the breakers are racked in for service. The staff concludes that a design which meets the above described commitments, together with administrative control and annunciation of the bus grounding system, will be sufficient to prevent inadvertent actuation of the grounding system. Consequently, this design is acceptable.

Verification that GE provided the above design commitments in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.3.3.15-1. GE included this information in Section 8.3.1.1.1 of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

In addition, GE indicated that there will be administrative controls to keep these circuit breakers racked out (that is in the disconnect position) whenever corresponding buses are energized. Verification that GE specified in a future SSAR amendment that the COL applicant will include these administrative controls in appropriate plant procedures was DFSER (SECY-92-349) COL Action Item 8.3.3.15-1. GE included this action item in Section 8.3.4.14 of SSAR Amendment 32, which is acceptable.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.3.3.15-1 related to the design description and the ITAAC for the safety bus grounding systems. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

8.3.3.16 Control of Access to Class 1E Power Equipment

By committing to meet the guidelines of IEEE 308-1980 (Table 1.8-21 of SSAR Amendment 17), GE indicated compliance with the requirements of Section 5.12 of IEEE 308-1980. Section 5.12 of IEEE 308-1980 requires that the plant design permit the administrative control of access to Class 1E power equipment. Based on this design commitment, information included in the draft SSAR revision dated April 3, 1992, and discussions, GE indicated that Class 1E power supplies and distribution equipment (including diesel generators, batteries, battery

chargers, CVCF power supplies, 6.9 kV switchgear, 480-volt load centers, and 480-volt MCCs) will be located in areas with access doors that can be administratively controlled. In addition, ac and dc distribution panels will be located in the same or similar areas as Class 1E power supplies and distribution equipment; otherwise, the distribution panels will be designed to be locked so that access to circuit breakers located inside the panel can be administratively controlled. The plant physical design of the ABWR will permit the administrative control of access to Class 1E power equipment areas.

GDC 4 of 10 CFR Part 50, Appendix A, requires that Class 1E systems, equipment, and components be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents. The staff concludes that a design which meets the above described design commitments will permit control of the conditions that equipment may be subjected to during normal operation, maintenance, testing, and postulated accidents. The design, therefore, meets the above defined requirements of GDC 4 and is acceptable.

Verification that GE provided the above design commitments in a future SSAR amendment was DFSER (SECY-92-349) 8.3.3.16-1. GE has included these commitments in Section 8.3.3.6.1.1(5) of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

In addition, GE indicated that there will be administrative control of access to Class 1E power equipment areas and/or distribution panels. Verification that GE specified in a future SSAR amendment that the COL applicant will include these administrative controls in appropriate plant procedures was DFSER (SECY-92-349) COL Action Item 8.3.3.16-1. GE included this action item in Section 8.3.4.19 of SSAR Amendment 32, which is acceptable.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.3.3.16-1 related to the design description and the ITAAC for control of access for Class 1E power equipment. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

8.3.4 Electrical Independence

Based on discussions with GE and information included in the draft SSAR revision dated April 3, 1992, GE indicated

Electric Power Systems

that electrical independence is addressed in the design as follows:

- The protective actions (that is, the initiation of a signal with the sense and command features, or the operation of equipment within the execute features, for the purpose of accomplishing a safety function) of each load group will be independent of the protective actions provided by redundant load groups.
 - Each onsite Class 1E power supply (for example, the diesel generator) will have provisions for automatic connection to one Class 1E load group, but will have no automatic connection to any other redundant Class 1E or non-Class 1E load group. If nonautomatic (manual) interconnecting means are furnished, provisions that prevent paralleling of the redundant onsite Class 1E power supplies will be included.
 - The ABWR electrical system design will not include provisions for the manual connection of the onsite Class 1E power supply of one Class 1E divisional load group to any other redundant Class 1E divisional or non-Class 1E non-divisional load group (except for the spare battery chargers)
 - + The ABWR design will include provisions to allow one spare battery charger to be connected to either of two divisions and another spare battery charger to be connected to either of two other divisions.
 - + The spare chargers for the dc power supply may be manually connected to either of two designated divisions, but only when their loads are switched to the same division. Key interlocks will mechanically ensure that these standby chargers can only be used in one division at a time.
 - The ABWR electrical system design will not have interconnections between redundant Class 1E divisions except as noted in Sections 8.2.2.3 and 8.3.4.1 of this report.
 - The divisional battery charger will normally be fed from its assigned Class 1E divisional 480-volt MCC bus.
 - Each standby power system division includes the diesel generator, its auxiliary systems, and the distribution of power to various Class 1E loads through the 6.9 kV and 480-volt systems. Each of these divisions will be segregated and separated from the other divisions. No automatic interconnection will be provided between the Class 1E divisions. Each diesel generator set will operate independently of the other sets.
 - Control power (for the Class 1E 480-volt auxiliaries) will be from the Class 1E 125-volt dc power system of the same division.
 - Each Class 1E dc system load group will have its own battery charger with no provision for automatic interconnection with other redundant Class 1E load groups.
 - There will be no provision for automatically interconnecting redundant dc system load groups.
 - No provision will be made for automatically transferring loads between Class 1E dc power sources.
 - The ABWR design will not have manual interconnections between redundant Class 1E divisions of the dc system except those that involve the battery chargers.
 - Each Class 1E battery will be independent of other redundant battery supplies.
 - Each Class 1E battery charger will be independent of other redundant battery chargers.
 - The ac and dc switchgear power circuit breakers in each division will receive control power from their respective load groups to provide the following assurances
 - loss of one Class 1E 125-volt dc system division will not jeopardize the Class 1E power supply to the Class 1E buses of the other load groups
 - the differential relays in one division and all the interlocks affiliated with these relays will be from one 125-volt Class 1E dc system division. There will be no cross connections between the redundant dc system divisions through protective relaying
- GDC 17 of 10 CFR Part 50, Appendix A, requires that the Class 1E systems, equipment, and components have sufficient independence and redundancy to perform their safety functions assuming a single failure. The staff concludes that a design which meets the above described commitments will include sufficient independent and redundant systems such that their will be reasonable assurance that a single failure of one set of systems (that is, one Class 1E division) will not prevent the remaining sets of interconnected system components, modules, and equipment from accomplishing the minimum required

safety function. The design, therefore, meets the above defined requirements of GDC 17 of 10 CFR Part 50, Appendix A and is acceptable.

Verification that GE provided the above design commitments in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.3.4-1. In response to this item, GE further clarified their design to indicate that, (in addition to the manual transfer capability of the battery chargers discussed above) the ABWR electrical design will permit the manual transfer capability of the FMCRDs motors from a non-divisional to a divisional supply as well as the manual transfer capability of any one of the Class 1E diesel generators to any one of the Class 1E divisional or non-divisional load groups by backfeeding power from the diesel generator through the combustion turbine bus. For these design clarifications, the staff also concludes that a design which meets the above described commitments as clarified will include sufficient independent and redundant systems such that there will be reasonable assurance that a single failure of one set of systems (that is, one Class 1E division) will not prevent the remaining sets of interconnected system components, modules, and equipment from accomplishing the minimum required safety function. The design, therefore, meets the above defined requirements of GDC 17 of 10 CFR Part 50, Appendix A and is acceptable.

GE included the above design commitments in Sections 8.3.3.1, 8.3.1.1.8.1, 8.3.1.1.2.1, 8.3.1.2(4)(b), 8.3.2.1.3, 8.3.2.1.3.1, and 8.3.2.2.1 of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

In addition, GE indicated that the keys for the above described interlocks, manual interconnections, and manual transfer of power supplies will be administratively controlled by the COL applicant. Verification that GE specified in a future SSAR amendment a statement that the COL applicant will include these administrative controls in appropriate plant procedures was DFSER (SECY-92-349) COL Action Item 8.3.4-1. GE included this action item in Sections 8.3.4.15 and 8.3.4.18 of SSAR Amendment 32, which is acceptable.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.3.4-1 related to the design description and the ITAAC for electrical independence. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

8.3.4.1 Interconnections

Chapter 7 of this SER addresses design requirements for I&C system isolation devices. These devices are to be used to maintain independence between Class 1E and non-Class 1E circuits (such as annunciators or data loggers and computer circuits) and between redundant Class 1E trip channels.)

Figure 8.3-8 of SSAR Amendment 10 showed two interconnections between redundant safety divisions:

- (1) The Division III 480-volt bus is connected to the Division I 480-volt bus through circuit breakers and a mechanical interlock. Section 8.3.2.1 of SSAR Amendment 10 indicates that this interconnection is used to transfer the 250-volt dc normal battery charger between Division I and III load centers.
- (2) The Division III 480-volt MCC is connected to Division I 480-volt MCC through battery chargers, breakers, and key interlocked breakers. Section 8.3.2.1 of SSAR Amendment 10 indicates that this interconnection is used for selection of the normal or standby battery chargers.

A staff concern over this issue was discussed in DSER (SECY-91-355) as Open Item 56.

In Section 8.3.2.1.4 and Figure 8.3-7 of the draft SSAR revision dated April 3, 1992, GE eliminated the interconnection between Divisions III and I (item 1 above), which was to be used to transfer the 250-volt dc battery charger between Class 1E divisions. In the new proposed design, power for the non-safety-related 250-volt dc battery charger is supplied from either the non-safety-related load group A or C turbine building load centers. With the elimination of the interconnection, this item was considered resolved.

Verification that GE provided the above design information in an SSAR amendment was one aspect of DFSER (SECY-92-349) Confirmatory Item 8.3.4.1-1. GE included this information in Section 8.3.2.1.3.3 of SSAR Amendment 32, which is acceptable.

In regard to the interconnection described in item 2 above, Section 8.3.2.1.2 and Figure 8.3-7 of the draft SSAR revision dated April 3, 1992, indicates that electrical interconnections will continue to exist between Divisions I and II and between Divisions I and III, so that two redundant divisions can share one standby charger. Similarly, Division I, and Division II, Division III, and Division IV, and Division I and Division III can be interconnected through the standby charger.

Electric Power Systems

GDC 17 of 10 CFR Part 50, Appendix A requires that the Class 1E systems, equipment, and components have sufficient independence to perform their safety functions assuming a single failure. To meet this GDC 17 requirement with respect to electrical interconnection between redundant Class 1E divisions, it is the staff position that two independent open disconnect links, locked open breakers, or other equivalent open devices shall be maintained between the redundant Class 1E divisions. To meet this staff position, GE indicated that key interlocks will be installed as part of their electrical system design which will mechanically ensure that two open devices are always maintained between redundant divisions in accordance with the above staff position. (Also see Section 8.3.4 of this report.)

The staff concludes that the proposed key interlock design which meets the above described design commitments will maintain independence between them by using two open devices. Failure of one device will not challenge or cause failure of the remaining redundant divisions. Therefore, this design meets the above defined requirement of GDC 17 of 10 CFR Part 50, Appendix A, and is acceptable.

Verification that GE provided the above design commitments in a future SSAR amendment was the second aspect of DFSER (SECY-92-349) Confirmatory Item 8.3.4.1-1. GE has included these commitments in Section 8.3.2.1.3.1 of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

In addition, GE indicated that the keys for the interlock described above will be administratively controlled. Verification that GE specified in a future SSAR amendment that the COL applicant will include these administrative controls in appropriate plant procedures was DFSER (SECY-92-349) COL Action Item 8.3.4.1-1. GE included this action item in Section 8.3.4.18 of an SSAR markup dated March 31, 1993, which is acceptable.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.3.4.1-1 related to the design description and the ITAAC for interconnections between redundant divisions. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

8.3.4.2 Constant Voltage Constant Frequency Power Supplies

Section 8.3.1.1.4.2 of SSAR Amendment 10 indicated that each of the four independent trip systems of the reactor

protection logic and control system is powered by four CVCF control power buses (one each for Divisions I, II, III, and IV). This section stated that each of these buses is independently supplied from an inverter which, in turn is supplied from one of four independent and redundant ac and dc power supplies. Subsequent sections and Figure 8.3-6 of SSAR Amendment 10, however, indicate that the ac supply for Divisions I and IV originates from a single 480-volt MCC (C14). A single 480-volt MCC is not "independent and redundant" as stated in Section 8.3.1.1.4.2. This was DSER (SECY-91-355) Open Item 57.

Based on discussions with and draft information provided by GE on September 4, 1991, November 26, 1991, and April 3, 1992, GE indicated that ac power to Divisions I and IV is supplied from the 6.9 kV Division I bus through a single 6.9 kV to 480-volt ac transformer and MCC to the vital ac system's CVCF power supplies and the dc system's battery charger power supplies for Divisions I and IV. GE indicated that Divisions I and IV ac and dc systems may be subject to a single common failure of the 6.9 kV to 480-volt transformer. In addition, GE indicated that they would revise the SSAR to indicate that there are four independent and redundant dc systems and three (versus four) independent and redundant ac systems.

Verification that GE provided the above design information in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.3.4.2-1. GE included this information in Section 8.3.1.1.4.2.1 of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

Subsequently, GE revised their design so that the ac supply for Division IV originates from a Division II (versus Division I) 480-volt MCC.

Because ac power to Divisions II and IV is supplied from a single 6.9 kV Division II bus through a single 6.9 to 480-volt ac transformer and MCC, Divisions II and IV ac and dc systems may be subject to common failure due to the single failure of the 6.9 kV to 480-volt transformer. The issue related to the lack of independence between Divisions II and IV of I&C equipment is resolved in Chapter 7 of this report.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.3.4.2-1 related to the design description and the ITAAC for constant voltage, constant frequency power supplies. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

8.3.4.3 Power Supply Circuits for Safety/Relief Valves

Section 19E.2.1.2.2.2 of SSAR Amendment 10 indicated that portions of each safety relief valve (SRV) control circuit use non-safety grade power. This power is taken from the Class 1E dc system through dc to dc converters or isolation devices connected to each of the four redundant and independent Class 1E dc system buses. Section 19E.2.1.2.2.2 implied that control power for each SRV comes from a minimum of two different Class 1E power source divisions. One source directly from the Class 1E dc bus, and the other from a different Class 1E dc bus through the dc-to-dc converter. The staff was concerned that the proposed design for powering the SRV may not provide sufficient independence between the redundant dc power sources, as required by GDC 17. This was DSER (SECY-91-355) Open Item 58.

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

GE indicated that SRVs will be powered only from Class 1E sources and that there will be no electrical interconnection between power supplies. This design will ensure electrical independence between the redundant power supplies, and is acceptable.

Verification that GE provided the above design commitments in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.3.4.3-1. GE included this information in Section 19E.2.1.2.2.2 of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

8.3.4.4 Isolation Between Class 1E Buses and Loads Designated as non-Class 1E

Section 8.3.1.1.2.1 of SSAR Amendment 10 indicated that isolation breakers will be provided between the Class 1E and non-Class 1E buses. In addition zone-selective interlocking was provided between each isolation breaker and its upstream Class 1E bus feeder breaker. Section 8.3.1.2.1 of SSAR Amendment 10 indicated that even though the isolation breaker is only fault current actuated and does not meet the guidelines of Position 1 of RG 1.75 (Rev. 2), the zone selective interlocking technique met the intent of this RG. GE therefore concluded that this design met the recommendations of RG 1.75 (Rev. 2).

With respect to protecting Class 1E systems from failure of non-Class 1E systems and components, the staff agreed with GE that coordinated breakers with zone selective interlocking met the intent of Position 1 of RG 1.75 (Rev. 2), as well as the protection requirements of GDC 2 and 4. However, with respect to the independence requirement of GDC 17, the staff disagreed with GE's assessment (DSER (SECY-91-355) Open Item 48). Figure 8.3-5 of SSAR Amendment 10 showed non-safety-related computers and transient recorder loads with provisions included in their power supply design for automatically transferring these loads from Class 1E Divisions I to Division III and from Class 1E Divisions II to Division III. In addition, it appeared that the power supply design may have included provision for automatic transfer of loads between Divisions I and II. This design did not meet the guidelines of RG 1.6 or the intent of Position 1 of RG 1.75 (Rev. 2).

Subsequently, in Section 8.3.1.1.1 of the draft SSAR revision dated April 3, 1992, GE eliminated automatic transfer of loads between redundant divisions by indicating that only Class 1E Division I will have a non-safety-related load and this is acceptable. Verification that GE provided the above design information in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.3.4.4-1. GE included this information in Section 8.3.1.1.1 of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

By SSAR markup dated March 31, 1993, GE revised their design described in Section 8.3.1.1.1 of the draft SSAR revision, dated April 3, 1992, for supplying power to the Non-Class 1E FMCRD motors from the Class 1E system. The following evaluation addresses GE response to DFSER (SECY-92-349) Confirmatory Item 8.3.4.4-2.

GE indicated that the non-Class 1E loads, to be connected to the Class 1E power system through isolation devices, will consist of three separate groups of non-Class 1E FMCRD motors. The rod drive motors are considered non-Class 1E but important to safety because of their backup scram function. In addition, GE indicated that these non-Class 1E loads will be restricted to Division I and will be isolated from Class 1E systems by a Class 1E fault-actuated breaker, a zone-selective interlock, and a design restriction that the circuits on the load side of the transfer switch be classified non-Class 1E (so that these circuits will not be routed as associated circuits with cables of any Class 1E division). Upon loss of power from the Class 1E source, the non-Class 1E FMCRD loads will automatically be transferred between the Class 1E Division I power supply and a non-Class 1E power supply. The design will not allow automatic transfer of this non-Class 1E load back from the non-Class 1E to the Class 1E

Electric Power Systems

power supply. This non-Class 1E load can only be manually transferred from the non-Class 1E to the Class 1E power supply.

GE indicated in Section 8.3.1.1.1 of the SSAR markup dated March 31, 1993, the following

- the fault interrupt capability of breakers supplying Class 1E loads, including the Class 1E breakers supplying the Division I non-Class 1E FMCRD loads, will be coordinated with the fault interrupting capability of each load's upstream supply breaker so that failure of a greater part of the Class 1E division due to the single failure of a load will be minimized to the extent feasible
- the Class 1E load breakers for the non-Class 1E FMCRD loads will have zone selective interlocks with the Class 1E supply breaker to provide additional assurance that failure of a portion of a Class 1E division due to the failure of the non-Class 1E load will be minimized to the extent feasible
- the FMCRD motor circuits from the output of the 6.9 kV switchgear through the transfer switch will be classified as associated
- the FMCRD motor circuits from the associated transfer switch through the load and the feeder circuits from the non-Class 1E bus to the transfer switch will be classified non-Class 1E
- the fault interrupt capability of all Class 1E breakers, fault interrupt coordination between the supply and load breakers for each Class 1E load and each Division I non-Class 1E load, the zone selective interlock feature of the breaker for the non Class 1E loads, and the transfer of power from Class 1E to non-Class 1E sources will have the capability of being tested
- the Division I Class 1E onsite power supplies, the non-Class 1E offsite power supplies, and the Division I distribution system will have sufficient capacity and capability with margin to supply all Class 1E loads and the additional non-safety loads during all modes of plant operation
 - each FMCRD power train has current limiting features that are Class 1E to limit the FMCRD motor fault current
 - continuous operation of the FMCRD motors at the limiting fault current will not degrade operation of any Class 1E loads

- the Division I diesel generator has sufficient capacity margin to supply overload currents up to the trip setpoint of the Class 1E feeder breaker to FMCRDs

In addition, by committing to meet the guidelines of IEEE 308-1980 (Table 1.8-21 of SSAR Amendment 17), GE indicated compliance with the requirements of Section 5.2 of IEEE 308-1980. Section 5.2 of IEEE 308-1980 requires that components, equipment, or systems utilized to provide isolation protection are covered by all the safety system design requirements defined in IEEE 603-1980. The staff concluded that the overcurrent protective devices and their coordination together with the zone select interlocks which are utilized to provide an isolation protection function to the Class 1E systems will meet the guidelines of Section 5.2 of IEEE 308-1980 and thus will meet safety system design requirements defined in IEEE 603-1980. By committing to meet the guidelines of IEEE 384-1981 (Table 1.8-21 of SSAR Amendment 33) and RG 1.75 (Rev. 2) (Table 1.8-20 of SSAR Amendment 33), GE indicated compliance with the guidelines of Sections 5.5.2(1) and 5.5.2(2) and Position 4 of RG 1.75, Revision 2. These guidelines require that associate circuits should be subject to all requirements placed on Class 1E circuits. The staff concluded that the non-Class 1E circuit that have been classified as associated will meet safety system design requirements defined in IEEE 603-1980. Also, by committing to the guidelines of IEEE 384-1981, GE indicated compliance with Section 7.1.2.1 of IEEE 384-1981. Section 7.1.2.1 requires that the circuit breaker being used for isolation shall be coordinated with upstream breakers, requires that the isolation breakers and their coordination be testable and be periodically tested, and requires that the Class 1E power supply to the loads being isolated have sufficient capacity and capability to supply fault current. The staff concluded that GE's proposed design for isolating non-Class 1E circuits includes design commitments which are in accordance with these guidelines of IEEE 384.

GDC 4 of 10 CFR Part 50, Appendix A, requires that Class 1E systems, equipment, and components be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents. GDC 17 of 10 CFR Part 50, Appendix A, requires (1) that offsite and onsite power systems have sufficient capacity and capability to permit safety systems to perform their required safety function and (2) that the Class 1E systems, equipment and components have sufficient independence to perform their safety functions assuming a single failure.

The staff concludes that an isolation system design which meets the above described commitments will adequately

protect the Class 1E power system from the failure of the non-Class 1E rod control loads, provides sufficient capability and capacity from either the non-Class 1E offsite or the Class 1E onsite power systems to supply Class 1E loads in addition to the non-Class 1E rod control loads, and provides sufficient independence between redundant Class 1E divisions. The proposed ABWR design, for isolating the non-Class 1E FMCRD motor loads from the Class 1E power system, therefore, meets the above defined requirements of GDC 4 and 17 of 10 CFR Part 50, Appendix A and is acceptable. GE included the above design commitments in Section 8.3.1.1.1 of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

In addition, GE indicated that the design permits periodic calibration and testing of the fault interrupt capability of all Class 1E breakers, fault interrupt coordination between the supply and load breakers for each Class 1E load and the Division I non-Class 1E load, and the zone selective interlock feature of the breaker for the non-Class 1E load. Verification that GE specified in a future SSAR amendment that the COL applicant must include these periodic calibrations and functional tests in appropriate plant procedures was DFSER (SECY-92-349) COL Action Item 8.3.4.4-1. GE included this action item in Section 8.3.4.29 of SSAR Amendment 32, which is acceptable.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.3.4.4-1 related to the design description and the ITAAC for the isolation between Class 1E buses and non-Class 1E loads. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

8.3.5 Lighting Systems

In its DSER (SECY-91-355), the staff identified 17 concerns in Open Item 59. GE provided a response in the draft SSAR revision dated April 3, 1992. In that document GE indicated that there will be four lighting systems: the normal ac lighting system, the standby ac lighting system, the emergency dc lighting system, and the guide lamp lighting system.

The Normal Lighting System

The normal ac lighting system will be used to provide up to 50 percent of the lighting needed for operation, inspection, and repairs during normal plant operation. Normal lighting will be installed throughout the plant in areas containing non-safety-related equipment. Normal

lighting will not be installed in passageways and stairwells. The normal lighting system will be part of the plant's non-safety-related system and as such will be supplied by the non-safety-related power system buses and will be energized as long as power from an offsite power source is available. Normal lighting will not be available following a LOPP event.

The ac Standby Lighting System

The ac standby lighting system will be comprised of two parts. The non-Class 1E ac standby system and the associated ac standby system. The Class 1E associated ac standby system will serve the safety-related equipment areas and their passageways. The non-Class 1E ac standby system will serve both safety and non-safety-related equipment areas and their passageways.

The associated ac standby system will be comprised of three subsystems. Each subsystem will be supplied from a different Class 1E standby divisional power supply (diesel generator). Each subsystem will supply a minimum of 50 percent of the lighting needs for the areas containing safety-related equipment in its respective division. Each subsystem will also supply 50 percent of the lighting in passageways and stairwells leading to its respective equipment areas. In addition, the subsystem associated with Division II will supply a minimum of 50 percent of the lighting needs for areas containing Division IV safety-related equipment (the Division IV battery room and other Division IV I&C areas). The subsystems associated with Divisions II and III will each supply a minimum of 50 percent of the lighting needs of the main control room. Each of the subsystem circuits will be treated as Class 1E (that is, they will be classified as associated) and as such will meet all requirements of a Class 1E circuit. In regard to the lighting system's fixtures and bulbs, GE has taken exception to the Class 1E requirement for seismic qualification. The staff's evaluation of this exception is addressed in Section 8.3.3.3 of this report.

The non-Class 1E ac standby system will be comprised of three non-Class 1E load groups. Each load group will be supplied from a different plant investment protection bus, which can be connected to the non-Class 1E standby power supply CTG. The non-Class 1E ac standby lighting system will supply a minimum of 50 percent of the lighting needed for its affiliated equipment areas and will supply 100 percent (50 percent from each of two different plant investment protection buses) of the lighting needs in passageways and stairwells leading to equipment areas containing non-safety-related equipment. In addition, the non-Class 1E ac standby lighting system will supply up to 50 percent of the lighting needs in areas containing safety-related equipment and in passageways and stairwells

AIM

Association for Information and Image Management

1100 Wayne Avenue, Suite 1100
Silver Spring, Maryland 20910
301/587-8202

Centimeter

Inches

MANUFACTURED TO AIM STANDARDS
BY APPLIED IMAGE, INC.

5 of 10

Electric Power Systems

leading to them. The non-Class 1E lighting in the areas containing safety-related equipment and the passageways and stairwells leading to them will be supplied from the same unit auxiliary transformer power source as the safety-related equipment in the area. (Each unit auxiliary transformer can supply a Class 1E and non-Class 1E distribution system and load group. For example, the unit auxiliary transformer that supplies power to Division I of the Class 1E distribution system and load group will also supply a non-Class 1E distribution system and load group.) The non-Class 1E lighting system's fixtures in areas containing safety-related equipment and in the passageways and stairwells leading to them will be seismically supported and will be designed with appropriate grids or diffusers such that broken material from either the fixture or bulb will be contained and will not become a hazard to personnel or safety equipment during or following a seismic event.

The dc Emergency Lighting System

The dc emergency lighting system will provide dc powered backup lighting to prevent total blackout in areas which are or may be occupied during periods when ac lighting is lost until the normal or standby lighting systems are energized. The dc emergency lighting system will be comprised of two parts. The non-Class 1E dc emergency lighting system and the associated dc emergency lighting system. The non-Class 1E dc emergency lighting system supplies the lighting needed in plant areas containing non-safety-related equipment. The associated dc emergency lighting system supplies the lighting needed in plant areas containing safety-related equipment.

Plant areas containing safety-related equipment and associated dc emergency lighting include the main control room, the remote shutdown panel rooms, the diesel generator areas and associated control rooms, the safety-related electrical equipment rooms, and dc electric equipment rooms (including battery rooms). Electrical power for the associated dc emergency lighting system in these safety-related equipment areas (except the main control room) is supplied from the Class 1E 125-volt dc system in the same division as the equipment in the area (for example, Class 1E dc Division I will supply power to the associated dc emergency lighting system in those rooms containing Division I safety-related equipment). Electrical power for the associated dc emergency lighting system in the main control room will be supplied from the Divisions II and III Class 1E 125-volt dc systems. Each of the dc emergency lighting circuits will be treated as Class 1E (that is, they will be classified as associated) and as such will meet all requirements of a Class 1E circuit. In regard to the lighting system's fixtures and bulbs, GE has taken exception to the Class 1E requirement for the

seismic qualification. The staff's evaluation of this exception is addressed in Section 8.3.3.3 of this report.

Plant areas containing non-safety-related equipment and non-Class 1E dc emergency lighting include the radwaste building control room, the CTG area and control room, and the non-safety-related electrical equipment areas (both ac and dc). Electrical power for the non-Class 1E dc emergency lighting system in the radwaste building control room will be supplied from the non-Class 1E 250-volt dc system. Electrical power for the non-Class 1E dc emergency lighting system in electrical equipment rooms containing non-safety-related equipment will be supplied from the same non-Class 1E 125-volt dc system that supplies power to equipment in the room. Electrical power for the non-Class 1E dc emergency lighting in the non-Class 1E CTG area and control room will be supplied from one of the three non-Class 1E 125-volt dc systems. The non Class 1E dc emergency lighting system circuits providing the needed lighting in non-safety-related equipment areas will be classified as non-Class 1E. (By letter dated March 31, 1993, GE modified their design to remove security lighting from their scope of design.)

The Guide Lamp Light System

The guide lamp light system will illuminate stairways, exit routes, and major control areas such as the main control room and remote shutdown panel areas. Each guide lamp unit will have a lighting fixture with two incandescent sealed-beam lamps and a self-contained battery pack unit containing a rechargeable battery with an 8-hour capacity. Each guide lamp unit will also contain a charger and an initiating switch, which energizes the fixture from the battery in the event of loss of the ac power supply and de-energizes the fixture upon return of ac power to the standby light following a time delay of 15 minutes. The guide lamp units will be supplied ac power from the same power source that supplies the associated standby lighting system in the area in which they are located. The guide lamp light system will be seismically qualified and will meet Class 1E requirements in plant areas containing Class 1E equipment.

GE provided information pertaining to the design of the normal, standby, emergency, and guide lamp lighting systems to demonstrate that the lighting is adequate in plant areas containing safety-related equipment as well as passageways to and from these areas. GE indicated that the ABWR design for each of these lighting systems will meet or exceed the lighting level requirements of the Illuminating Engineering Society Lighting Handbook and will have the capability of being functionally tested on a periodic basis.

The staff concluded that a lighting system that meets the above described commitments will provide adequate levels of light to permit the required operation and maintenance of equipment in safety-related equipment areas, and passageways to and from these areas, under normal operating conditions. This design is, therefore, acceptable. Verification that GE provided the above design commitments in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.3.5-1. GE included this information in Section 9.5.3 of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

In addition, GE indicated that the ac standby and dc emergency lighting systems which are normally energized will be periodically inspected and bulb replacement will be performed. In addition, the guide lamp system will be inspected and tested periodically to ensure operability of lights and switching circuits. Verification that GE has specified in a future SSAR amendment the requirement that the COL applicant must include these periodic tests, inspections, and bulb replacement in appropriate plant procedures was DFSER (SECY-92-349) COL Action Item 8.3.4.4-1. GE included this requirement in Section 8.3.4.25 of SSAR Amendment 32, which is acceptable.

For other off-normal conditions, it was not clear that the design adequately considers lighting needs for areas containing safety-related equipment and for passageways to and from these areas where plant operations are or may be required by emergency procedures. For example, under certain failures, it appears that the main control room may have only a portion (50 percent) of its lighting. Therefore, GE needed to further address the adequacy of the lighting in areas containing safety-related equipment under off-normal conditions. This was DFSER (SECY-92-349) Open Item 8.3.5-1.

GE indicated that because of the redundancy provided by the four lighting systems described above, the complete loss of lighting in any of the critical areas is not credible. The standby lighting and emergency lighting systems provide totally independent low level illumination in areas vital to safe shutdown of the reactor. In addition, the safety-related control systems will automatically bring the plant to safe-shutdown conditions. The control systems do not require the lighting system to perform their safety function of bringing the plant to a safe-shutdown condition.

The staff concluded that a lighting system design which meets the above described commitments will provide reasonable assurance that sufficient illumination will be available to perform emergency procedures as may be required during off normal conditions. The lighting system design is, therefore, acceptable. GE included the

above commitments in Section 9.5.3.2.5 of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.3.5-2 related to the design description and the ITAAC for the lighting system. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

8.3.6 Design Control

8.3.6.1 Control of the Electrical Design Process

Recently, a number of problems have been identified with the electrical system design at nuclear power plants. Although the majority of these problems arose as a result of modifications performed after plant licensing, some were (and all could have been) the result of poor original design. GL 88-15 dated September 12, 1988, addressed a number of these problems that have occurred primarily as a result of inadequate control over the design process. These problems have occurred in areas of electrical system design which have historically well-established, comprehensive design criteria and guidelines available for the design engineer such as circuit breaker coordination and fault current interruption capability. The adequacy of the design is a function of the designers proper exercise of the well-established design criteria and guidelines.

In Sections 8.3.4.17 and 8.3.5 of draft SSAR revision dated April 3, 1992, GE indicated that purchase specification for both Class 1E and non-Class 1E equipment will contain a list of appropriate common industrial standards to ensure quality manufacturing. Based on this commitment, the staff concluded that this concern (DSER SECY-91-355 Open Item 60) is resolved. Verification that GE provided the above design information in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.3.6.1-1. GE included this information in Sections 8.3.4.17 and 8.3.5 of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.3.6.1-1 related to the design description and the ITAAC for the control of the electrical design process. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

Electric Power Systems

8.3.6.2 Control of the Electrical Design Bases

The design bases described and presented in the ABWR SSAR should, for the most part, be useable as the bases by which the NRC issues a plant combined operating license. However, a review of the bases presented in Chapter 8 and other related chapters, revealed numerous inconsistencies as noted in DSER (SECY-91-355) Open Item 61. Consequently, it appears that GE used a deficient process to control the design bases presented in the ABWR SSAR.

At the time the DFSER was issued, GE had indicated that a formal engineering review and update of the SSAR was in progress to identify and correct inconsistencies within and between sections of the SSAR. Verification that GE revised the design bases in a future SSAR amendment to eliminate conflicting information was DFSER (SECY-92-349) Confirmatory Item 8.3.6.2-1. Staff review of SSAR Amendment 32 did not show these that inconsistencies had been resolved. Subsequently, the staff has reviewed the SSAR through Amendment 34 and found no significant inconsistencies which affect safety findings in this chapter. Therefore, Confirmatory Item 8.3.6.2-1 is resolved.

8.3.7 Testing and Surveillance

This section addresses DSER (SECY-91-355) Open Item 62. By committing to meet the guidelines of IEEE 308-1980 (Table 1.8-21 of SSAR Amendment 17), GE indicated compliance with the requirements of Section 7 of IEEE 308-1980. Section 7 of IEEE 308-1980 defines the guidelines for surveillance and preoperational and periodic equipment and system tests for electrical systems. Based on this commitment to Section 7 of IEEE 308-1980, information presented in the draft SSAR dated April 3, 1992, and discussions, GE indicated the following:

- The ABWR electrical system design will provide controls and indicators in the main control room.
- The design will include provisions for control and indication outside the main control room for
 - circuit breakers that switch Class 1E buses between the preferred and the standby power supply
 - the standby power supply
 - circuit breakers and other equipment as required for safety systems that must function to bring the plant to a safe-shutdown condition.

- Operational status information will be provided for Class 1E power systems.
- Class 1E power systems required to be controlled from outside the main control room will also have operational status information provided outside the central control room at the equipment itself, at its power supply, or at an alternate central location.
- The operator will be provided with accurate, complete, and timely information pertinent to the status of the execute features in the control room.
- Indication of protective actions and execute features unavailability, will be provided in the control room.
- Electric power systems and equipment will have the capability of being periodically tested.
- Testability of electrical systems and equipment will not be so burdensome operationally that required testing at intervals of 1, 2, or 3 months cannot be included in the TS if deemed necessary.

GDC 17 of 10 CFR Part 50, Appendix A, requires that Class 1E systems, equipment, and components have sufficient testability to perform their safety functions assuming a single failure. GDC 18 of 10 CFR Part 50, Appendix A requires that Class 1E power systems be designed (1) to permit appropriate periodic inspection and testing of important areas and features (such as wiring, insulation, connections, and switchboards) to assess the continuity of the systems and the condition of their components, (2) the capability to test periodically the operability and functional performance of the components of the systems, and (3) the capability to test periodically the operability of the systems as a whole and (under conditions as close to design as practical) the full operation sequence that brings the systems into operation.

Except as noted below, the staff concludes that an electrical system design which meets the above commitments will be testable. The design, therefore, meets the above defined requirements of GDC 17 and 18 of 10 CFR Part 50, Appendix A and is acceptable.

Verification that GE provided the above design commitments in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.3.7-1. GE has included these commitments in Section 8.1.3.1.1.3 of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

GE also indicated that the electrical systems and equipment will be periodically tested. Verification that GE specified

in a future SSAR amendment that the COL applicant will include this periodic testing of electrical systems and equipment in appropriate plant procedures was DFSER (SECY-92-349) COL Action Item 8.3.7-1. GE included this action item in Section 8.3.4.30 of SSAR Amendment 32, which is acceptable.

Section 8.3.1.1.5.3, of SSAR Amendment 10 indicated that the ABWR design of Class 1E equipment permits periodic testing of the chain of system elements from sensing devices through driven equipment. This testing ensures that Class 1E equipment is functioning according to design requirements. This section also implies that the requirements of the single-failure criterion described in IEEE 379-1977, "Standard Application of the Single Failure Criterion to Nuclear Power Generating Station Safety Systems," will be met with respect to testing of Class 1E equipment.

The staff interpreted these statements to mean that one complete electrical system division may be taken out of service for maintenance, testing, and/or repair during any mode of plant operation and still leave the remaining electrical systems in compliance with the single-failure criterion. The staff, therefore, concluded that the design provision for testability of electrical systems (as interpreted) meets the sufficient testability requirement of GDC 17 and is acceptable.

To confirm this interpretation, the staff further evaluated the capability of the electric power system to be tested during normal plant operation while meeting single failure requirements with remaining systems for any design-basis event.

GE's design commitment that the ABWR design meets Section 5 of IEEE 338-1977, which is endorsed by RG 1.118 (Rev. 2), reflected the following:

- On-line testing will be greatly enhanced by the design which utilizes three independent divisions, any one of which can safely shutdown the plant. However, the design will not meet the single-failure criterion with respect to all required safety-related systems for all design-basis events with one of the three electrical system divisions out of service.
- An acceptable level of reliability for the remaining operable safety systems will exist when one train is taken out of service for a specified period of time for preplanned or unplanned maintenance while the single-feature criterion is not met.

- When one Class 1E division is out of service, an acceptable level of reliability will be established by GE for the remaining operable safety systems by a PRA.

Based on the information presented in the ABWR SSAR, the staff was unable to determine what constitutes an acceptable level of reliability. This was DFSER (SECY-92-349) Open Item 8.3.7-1.

GE revised Sections 8.2.2.1(3) and 8.3.1.1.5.3 of SSAR Amendment 21 to indicate that all equipment can be tested, as necessary, to assure continued and safe operation of the plant. For equipment which will not be tested during operation, GE indicated that the equipment's reliability will be such that testing can be performed during plant shutdown. Based on the above commitment as revised, the staff concludes that all systems and component parts of an electrical system design which met the above commitments as revised will have the capability of being tested. In addition, the reliability assurance program as discussed in Chapter 17.3 of this report will ensure that important reliability assumptions of the probabilistic risk assessment will be considered throughout the plant life. The design therefore meets the above defined requirements of GDC 17 and 18 of 10 CFR Part 50, Appendix A, and is acceptable. GE included the above design commitments in Sections 8.2.2.1(3) and 8.3.3.2 of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.3.7-2 related to the design description and the ITAAC for the testing and surveillance of electrical equipment. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

8.3.8 Capacity and Capability

8.3.8.1 Non-Class 1E dc Power Systems

In response to DSER (SECY-91-355) Open Item 64, regarding interaction between safety and non-safety-related systems, GE provided in Sections 8.3.2.1.3 and 8.3.2.1.4 of the draft SSAR revision dated April 3, 1992, a description of the non-Class 1E 125- and 250-volt dc systems. The 125-volt dc non-Class 1E system will provide power to non-Class 1E switchgear, valves, converters, transducers, controls, and instrumentation. The non-Class 1E 125-volt dc system will have three load groups with one battery, charger, and bus per load group. The 250-volt dc

Electric Power Systems

non-Class 1E system will provide power for non-safety-related computers and the turbine turning gear motor. The 125- and 250-volt dc systems will provide power only to non-safety-related loads and will be physically and electrically independent of the Class 1E ac and dc systems.

Based on the design commitments included in the draft SSAR revision, staff concerns relating to interactions between safety and non-safety-related systems associated with the previously proposed common dc power supply have been resolved.

Verification that GE provided the above design commitments in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.3.8.1-1. GE included this information in Sections 8.3.2.1.3.2 and 8.3.2.1.3.3 of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.3.8.1-1 related to the design description and the ITAAC for the independence between Class 1E and non-Class 1E dc power systems. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

8.3.8.2 Capacity of the Class 1E 125-volt dc Battery Supply

This section addresses the staff's evaluation of GE's response to DSER (SECY-91-355) Open Item 65.

Based on information presented in Section 8.3.2.1.1.1 of the draft SSAR revision dated April 3, 1992, GE indicated that each of the four Class 1E 125-volt batteries will

- be capable of starting and operating its required steady state and transient loads
- be immediately available during both normal operations and following the loss of power from the alternating current system
- have sufficient stored energy to provide an adequate source of power for starting and operating all required LOCA and/or LOPP loads and circuit breakers for two hours with no ac power
- have sufficient stored energy to provide power in excess of the capacity of the battery charger when needed to supply transient loads

- be sized in accordance with industry recommended practice defined in IEEE 485-1983
- have a capacity design margin of 5 to 15 percent to allow for less than optimum operating conditions
- have a 25-percent capacity design margin to compensate for battery aging
- have a 4-percent capacity design margin to allow for the lowest expected electrolyte temperature of 21 °C (70 °F)
- have a number of battery cells that matches the battery-to-system voltage limitations
- base the first minute of the batteries' duty cycle on the sum of all momentary, continuous, and noncontinuous loads that can be expected to operate during the one minute following a LOCA and/or LOPP
- be installed in accordance with IEEE 484-1987
- meet the recommendations of Section 5 of IEEE 946-1985
- be designed so that each battery's capacity can periodically be verified

In addition to having sufficient stored energy to operate all required LOCA and/or LOPP loads and circuit breakers for 2 hours, GE indicated that the Division I battery will have sufficient stored energy to provide an adequate source of power to start and operate all required loads and circuit breakers for approximately 8 hours with no ac power. Further, the heating/ventilation system will maintain battery electrolyte temperature above 21 °C (70 °F).

GDC 17 of 10 CFR Part 50, Appendix A, requires that the Class 1E 125-volt dc battery supply have sufficient (1) capacity and capability to permit safety systems to perform their required safety function and (2) testability to perform their safety function assuming a single failure. GDC 18 of 10 CFR Part 50, Appendix A requires that the Class 1E 125-volt dc battery supply systems be designed (1) to permit appropriate periodic inspection and testing of important areas and features (such as wiring, insulation, connections, and switchboards) to assess the continuity of the systems and the condition of their components, (2) the capability to test periodically the operability and functional performance of the components of the systems, and (3) the capability to test periodically the operability of the systems as a whole and (under conditions as close to design as practical) the full operation sequence that brings the systems into operation. The staff concludes that Class 1E

125-volt battery supply design that meets the above commitments will have sufficient capacity and capability to supply required loads following a LOCA and/or LOPP and a station blackout event and will be testable. The design, therefore, meets the above defined requirements of GDC 17 and 18 of 10 CFR Part 50, Appendix A, and is acceptable.

Verification that GE provided the above design commitments in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.3.8.2-1. In response to this item, GE modified the above commitments to indicate that the batteries' capacity design margin to allow for the lowest expected electrolyte temperature was changed from 21 °C (70 °F) to 10 °C (50 °F). This change was based on the capability of the heating and ventilation system to maintain room temperature above 10 °C (50 °F). The capability of the heating and ventilation system to maintain a temperature range of 10 to 40 °C (50 °F to 104 °F) is addressed in Chapter 9.0 of this SER.

The staff concludes that a Class 1E 125-volt battery supply design that meets the above commitments as modified, will also have sufficient capacity and capability to supply required loads following a LOCA and/or LOPP and a station blackout event and will be testable. The design, therefore, meets the above defined requirements of GDC 17 and 18 of 10 CFR Part 50, Appendix A, and is acceptable. GE has included the above design commitments in Sections 8.3.2.1.1.1, 8.3.2.1.3.1, and 8.3.2.2(5)(c) of SSAR Amendment 32, which is acceptable.

In addition, GE indicated that the capacity and capability of the dc system batteries and the capability of the batteries to supply power to their connected loads will periodically be tested in accordance with the recommendation of IEEE 450-1985. Verification that GE specified in a future SSAR amendment that the COL applicant will include these periodic tests in appropriate plant procedures was DFSER (SECY-92-349) COL Action Item 8.3.8.2-1. GE included this action item in Sections 8.3.4.32 and 8.3.4.33 of SSAR Amendment 32, which is acceptable.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.3.8.2-1 related to the design description and the ITAAC for the Class 1E 125-volt dc battery supply. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

8.3.8.3 Use of Silicon Diode in the dc System

Figure 8.3-7 and GE's response to Question 435.51 in SSAR Amendment 10 indicated that a silicon diode would be installed in series with the output of the battery and battery chargers to create a voltage drop of 10 volts. During normal operation (that is, when battery charger output voltage is set at 140 volts to equalize the charge of battery cells) the switch in parallel with the silicon diode will be open so that the voltage from the battery charger to the dc bus will remain at 130 volts (140 volts minus the 10-volt drop across the silicon diode), while 140 volts is supplied to the battery to equalize the charge of battery cells.

In response to DSER (SECY-91-355) Open Item 66, GE provided the draft SSAR revision dated April 3, 1992, which removed the use of the silicon diode from the ABWR design and restated the commitment to design the dc system distribution equipment, component, and loads to function at 140 volts during equalization charge.

This item is, therefore, considered resolved. Verification that GE revised the SSAR in a future amendment, indicating the removal of the silicon diode from the ABWR design was DFSER (SECY-92-349) Confirmatory Item 8.3.8.3-1. GE included this revision in SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

8.3.8.4 Class 1E ac Standby Power System (Diesel Generator)

This section addresses the staff's evaluation of GE's response to DSER (SECY-91-355) Open Item 67.

Based on discussions with GE and information presented in the draft SSAR revision dated April 3, 1992. GE indicated that each standby (diesel generator) power source will

- be capable of starting, accelerating to rated speed, and supplying, in the required sequence, all the required safety system loads
- be capable of attaining rated frequency (i.e., full speed) and voltage within 20 seconds after receipt of a start signal
- have a continuous load rating of 6.25 megavolt amps (MVA) @ 0.8 power factor

Electric Power Systems

- have a short time rating (operation at this rating does not limit the use of the diesel generator unit at its continuous rating) of 110 percent of the continuous load rating for a 2-hour period out of any 24-hour period, without exceeding the manufacturer's design limits and without reducing the maintenance interval established for the continuous rating
 - be available following the loss of the preferred power supply within a time consistent with the requirements of the safety function under normal and accident conditions
 - have stored energy (fuel) at the site in its own storage tank with the capacity to operate the standby diesel generator power supply, while supplying post-accident power requirements to a unit for seven days
 - have stored energy (fuel) at the site in its own day tank with the capacity to operate the standby diesel generator power supply while supplying post-accident power requirements for 8 hours
 - have a fuel transfer system with the capability of automatically replenishing the day tank from the storage tank such that the 8-hour fuel capacity of the day tank is maintained
 - be capable of operating in its service environment during and after any design-basis event, without support from the preferred power supply
 - be capable of starting, accelerating, and being loaded with the design load, within an acceptable time
 - from the diesel engine's normal standby condition
 - with no cooling available, for a time equivalent to that required to bring the cooling equipment into service with energy from the diesel generator unit
 - on a restart with an initial engine temperature equal to the continuous rating full load engine temperature
 - be capable of accepting design load following operation at light or no load for a period of 4 hours
 - be capable of maintaining voltage and frequency at the generator terminals within limits that will not degrade the performance of any of the loads comprising the design load below their minimum requirements, including the duration of transients caused by load application or load removal
 - be capable of carrying its continuous load rating for 22 hours following 2 hours of operation at its short time rating
 - start from each automatic and remote manual signal and then accelerate to rated voltage and frequency, and then properly sequence its loads if there is no offsite power available or operate at no load if offsite power is available
 - start but not sequence its loads by a local manual start signal
 - be capable of being manually started without ac external electric power
 - be capable of automatic acceleration to rated voltage and frequency without ac external electric power
 - be capable of allowing the bus to be manually energized without ac external electric power
- GE also indicated that
- the maximum loads expected to occur for each division (according to nameplate ratings) will not exceed 90 percent of the continuous power output rating of the diesel generator
 - each diesel generator's air receiver tanks will have sufficient capacity for five starts without recharging
 - following one unsuccessful automatic start of the diesel generator with and without ac external power, each diesel generator's air receiver tanks will have sufficient air remaining for three more successful starts without recharging
 - automatic load sequence will begin at ≤ 20 seconds and will end at ≤ 65 seconds
 - following application of each load during load sequencing, voltage will not drop more than 25 percent from nominal voltage measured at the bus
 - following application of each load during load sequencing frequency will not drop more than 5 percent from nominal frequency measured at the bus
 - frequency will be restored to within 2 percent of nominal, and voltage will be restored to within 10 percent of nominal within 60 percent of each load sequence time interval

- during recovery from transients caused by step load increases or resulting from the disconnection of the largest single load, the speed of the diesel generator unit will not exceed the nominal speed plus 75 percent of the difference between nominal speed and the overspeed trip setpoint or 115 percent of nominal, whichever is lower
- the transient following the complete loss of load will not cause the speed of the unit to attain the overspeed trip setpoint.
- bus voltage and frequency will recover to $6.9 \text{ kV} \pm 10\text{-percent}$ at $60 \pm 2\text{-percent Hz}$ within 10 seconds following trip and restart of the largest load
- each of the above design commitments will have the capability of being periodically verified

GDC 17 of 10 CFR Part 50, Appendix A, requires that the Class 1E ac standby power supply (the diesel generator power supply) have sufficient (1) capacity and capability to permit safety systems to perform their required safety function and (2) testability to perform their safety functions assuming a single failure. GDC 18 of 10 CFR Part 50, Appendix A, requires that the Class 1E ac standby electric power system be designed (1) to permit appropriate periodic inspection and testing of important areas and features (such as wiring, insulation, connections, and switchboards) to assess the continuity of the systems and the condition of their components, (2) the capability to test periodically the operability and functional performance of the components of the systems, and (3) the capability to test periodically the operability of the systems as a whole and (under conditions as close to design as practical) the full operation sequence that brings the systems into operation. The staff concludes that a diesel generator power supply design that meets the above described commitments will have sufficient capacity and capability to supply required loads and will be testable. The design, therefore, meets the above defined requirements of GDC 17 and 18 of 10 CFR Part 50, Appendix A, and is acceptable. Verification that GE has revised the SSAR to include the above design commitments in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.3.8.4-1. GE has included these commitments in Sections 8.1.3.1.1.3, 8.3.1.1.8.2, 8.3.1.1.8.3, and 8.3.1.1.8.6 of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

GE indicated that each of the above design commitments will be verified periodically and that testing or analysis will be performed periodically to demonstrate the capability of the diesel generator to supply the actual full design basis load current for each sequenced load step. Verification

that GE specified in a future SSAR amendment that the COL applicant will include these periodic tests and analysis in appropriate plant procedures was DFSER (SECY-92-349) COL Action Item 8.3.8.4-1. GE included this action item in 8.3.4.36 of SSAR Amendment 32, which is acceptable.

Start time for the diesel generators

As part of DSER (SECY-91-355) Open Item 67, the staff identified inconsistencies between Sections 8.3.1.1.8.2 and 8.3.1.1.8.3 of SSAR Amendment 10 with regard to the design capability of the diesel generator to start and attain rated voltage and frequency. Section 8.3.1.1.8.2 of SSAR Amendment 10 indicated a 13-second design capability, while Section 8.3.1.1.8.3 indicated a 20-second capability.

GE's SSAR Amendment 17 corrected the inconsistency by changing the 13-seconds start time for the diesel generator to 20 seconds, with the sequence start times for loads changing accordingly. GE indicated that this change was made to be consistent with EPRI/advanced light water reactor requirements.

GE indicated that the accident analysis requires the residual heat removal (RHR) and HPCF injection valves to be open 36 seconds after the receipt of a high drywell or low reactor vessel level signal. Since the motor operated valves are not tripped off the buses, they start to open (if requested to do so by their controls) when power is restored to the bus at 20 seconds. This gives them an allowable travel time of 16 seconds.

Changing the allowable start time for the diesel generator from 13 to 20 seconds has changed the allowable travel time for the RHR and HPCF injection valves to move from the close to open position from 23 to 16 seconds. GE indicated that this reduction to a 16 second travel or opening time is attainable for the RHR and HPCF injection valves. The staff therefore concluded that the longer start time for the diesel generator is within the accident analysis limits and is acceptable. GE included this information in Section 8.3.1.1.8.2 of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

Load Sequencing on the Diesel Generator following a LOCA with a delayed LOPP

This section addresses, in part, the staff's evaluation of GE's response to DSER (SECY-91-355) Open Item 28 and DFSER (SECY-92-349) Confirmatory Item 8.2.3.2-1.

As noted in Section 8.2.3.2 of this report, the diesel generator may be overloaded if there is a LOCA with a delayed LOPP. Section 8.3.1.1.7(3) of SSAR

Electric Power Systems

Amendment 21, LOPP following LOCA, indicated that all loads that have been connected to the Class 1E bus and are operating from the offsite power supply in response to the LOCA will be connected, in one block load, to the standby diesel generator (operating at no load due to the LOCA) if there is a LOPP. The staff was concerned that the diesel generator may not have sufficient capacity to supply, in one block, the loads which may be operating on the Class 1E bus following a LOCA with a delayed LOPP.

Subsequently, GE indicated that the diesel generator breaker will not close until after large motors loads are tripped. Thus, loads will be reloaded onto the bus in a controlled sequence which will not overload the diesel generator power supplies.

If there is a LOCA with a delayed LOPP, if there is a coincident LOCA and LOPP, or if there is a LOPP with a subsequent LOCA, the staff concludes that a diesel generator power supply design that meets the above commitments (as clarified) will have sufficient capacity and capability to supply required loads for all modes of plant operation and will be testable. The design, therefore, meets the above defined requirements of GDC 17 and 18 of 10 CFR Part 50, Appendix A, and is acceptable. GE has included the above clarification in Section 8.3.1.1.7 of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.3.8.4-1 was related to the design description and the ITAAC for the capacity and capability of the Class 1E ac standby power system. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

8.3.8.5 Constant Voltage Constant Frequency Power Supply Capacity

Based on information presented in the draft SSAR revision dated April 3, 1992, and discussions; GE indicated that each of the four redundant Class 1E CVCF power supplies will have a capacity based on the largest combined demands of the various continuous loads, plus the largest combination of noncontinuous loads that would likely be connected to the power supply simultaneously during normal or accident plant operation, whichever is higher. The design will also permit periodic verification of this capacity for each of the CVCF power supplies.

GDC 17 of 10 CFR Part 50, Appendix A, requires that the Class 1E CVCF power supplies have sufficient (1) capacity and capability to permit safety systems to perform their

required safety function and (2) testability to perform their safety functions assuming a single failure. GDC 18 of 10 CFR Part 50, Appendix A, requires that the Class 1E CVCF power supply systems be designed (1) to permit appropriate periodic inspection and testing of important areas and features (such as wiring, insulation, connections, and switchboards) to assess the continuity of the systems and the condition of their components, (2) the capability to test periodically the operability and functional performance of the components of the systems, and (3) the capability to test periodically the operability of the systems as a whole and (under conditions as close to design as practical) the full operation sequence that brings the systems into operation. The staff concludes that a CVCF power supply design that meets the above commitments will have sufficient capacity and capability to supply required loads and will be testable. The design, therefore, meets the above defined requirements of GDC 17 and 18 of 10 CFR Part 50, Appendix A, and is acceptable.

Verification that GE provided the above design commitments in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.3.8.5-1. GE has included these commitments in Section 8.3.1.1.4.2.1 of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

GE also indicated that the above capacity commitment for the CVCF power supplies will periodically be verified. Verification that GE specified in a future SSAR amendment that the COL applicant will include these periodic tests in appropriate plant procedures was DFSER (SECY-92-349) COL Action Item 8.3.8.5-1. GE included this action item in Section 8.3.4.34 of SSAR Amendment 32, which is acceptable.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.3.8.5-1 related to the design description and the ITAAC for the capacity of the CVCF power supplies. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

8.3.8.6 Battery Charger

Based on information presented in the draft SSAR revision dated April 3, 1992, and discussions, GE indicated that each of the four redundant Class 1E dc battery chargers will have sufficient capability and will be testable. The battery chargers will have a capacity based on the largest combined demands of the various continuous steady-state loads, plus charging capacity to restore the battery from the design minimum charge state to the fully charged state

within the time stated in the design-basis, regardless of the status of the plant during which these demands occur. The dc battery chargers will have a disconnecting device in the incoming ac power feed and in its dc-power output circuit for isolating the charger. The dc battery chargers will be designed to prevent the ac power supply from becoming a load on the battery. The dc battery chargers will also have provisions to isolate transients from the ac system from affecting the dc system and; conversely, provisions will be included to isolate transients from the dc system from affecting the ac system. The dc battery charger system will be sized in accordance with the guidelines defined in Section 6.1 of IEEE 946-1985 for establishing the required rating for battery chargers. The design of the dc system will include the capability to periodically verify the required capacity for each of the battery charger power supplies.

GDC 17 of 10 CFR Part 50, Appendix A, requires that the Class 1E battery chargers have sufficient (1) capacity and capability to permit safety systems to perform their required safety function and (2) testability to perform their safety functions assuming a single failure. GDC 18 of 10 CFR Part 50, Appendix A, requires that the Class 1E dc system battery chargers be designed (1) to permit appropriate periodic inspection and testing of important areas and features (such as wiring, insulation, connections, and switchboards) to assess the continuity of the systems and the condition of their components, (2) the capability to test periodically the operability and functional performance of the components of the systems, and (3) the capability to test periodically the operability of the systems as a whole and (under conditions as close to design as practical) the full operation sequence that brings the systems into operation. The staff concludes that a dc system battery charger design that meets the above described commitments will have sufficient capacity and capability to supply required loads and will be testable. The design, therefore, meets the above defined requirements of GDC 17 and 18 of 10 CFR Part 50, Appendix A, and is acceptable.

Verification that GE provided the above design commitments in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.3.8.6-1. GE has included these commitments on Figure 8.3-4 and in Sections 8.3.2.1.1 of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

GE also indicated that the above capacity commitment for the battery charger power supplies will be periodically verified. Verification that GE specified in a future SSAR amendment that the COL applicant will include this periodic testing of the battery capacity in appropriate plant procedures was DFSER (SECY-92-349) COL Action

Item 8.3.8.6-1. GE included this action item in Section 8.3.4.35 SSAR Amendment 32, which is acceptable.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.3.8.6-1 related to the design description and the ITAAC for the capacity of the battery charger. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

8.3.8.7 Distribution Systems

Based on information presented in the draft SSAR revision dated April 3, 1992, and discussions, GE indicated that each Class 1E distribution circuit will be capable of transmitting sufficient energy to start and operate all required loads in that circuit for all plant conditions described in the design basis. GE also indicated that the design will also permit periodic verification of this required capacity for each distribution circuit.

GDC 17 of 10 CFR Part 50, Appendix A, requires that the Class 1E distribution system have sufficient (1) capacity and capability to permit safety systems to perform their required safety function and (2) testability to perform their safety functions assuming a single failure. GDC 18 of 10 CFR Part 50, Appendix A, requires that the Class 1E distribution systems be designed (1) to permit appropriate periodic inspection and testing of important areas and features (such as wiring, insulation, connections, and switchboards) to assess the continuity of the systems and the condition of their components, (2) the capability to test periodically the operability and functional performance of the components of the systems, and (3) the capability to test periodically the operability of the systems as a whole and (under conditions as close to design as practical) the full operation sequence that brings the systems into operation.

The staff concludes that a Class 1E distribution system design that meets the above described commitments will be capable of supplying sufficient energy to Class 1E safety system loads for their operation, and will be capable of being tested. The design, therefore, meets the above defined requirements of GDC 17 and 18 of 10 CFR Part 50, Appendix A, and is acceptable.

Verification that GE provided the above design commitments in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.3.8.7-1. GE has included these commitments in Sections 8.3.1.1.5(2) and

Electric Power Systems

8.3.3.2 of SSAR Amendment 32, which is acceptable. Therefore, this item is resolved.

GE also indicated that the above capacity commitment for each distribution circuit will periodically be verified. Verification that GE has specified in a future SSAR amendment that the COL applicant will include this periodic testing in appropriate plant procedures was DFSER (SECY-92-349) COL Action Item 8.3.8.7-1. GE included this action item in Section 8.3.4.30 of SSAR Amendment 32, which is acceptable.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.3.8.7-1 related to the design description and the ITAAC for capacity of distributions systems. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report. Therefore, this item is resolved.

8.3.9 Station Blackout (SBO)

In SECY-90-016, dated January 12, 1990, the staff recommended that the Commission approve its position that a diverse, AAC source, be provided for SBO in evolutionary LWRs. In its June 26, 1990, SRM, the Commission approved the staff's position.

In response to the above staff position and DFSER (SECY-92-349) Confirmatory Items 8.3.9.1-1, 8.3.9.2-1, and 8.3.9.3-1 and open items 8.3.9.1-1, 8.3.9.2-1, and 8.3.9.3-1, GE indicated that they planned to change their commitment for meeting the SBO rule from a design that met the SBO rule through the use of coping to a design that meets the SBO rule through the use of an alternate ac (AAC) power source in accordance with NRC Policy Issue SECY-90-016.

In an SSAR draft version dated July 2, 1993, GE indicated in accordance with RG 1.115, that the AAC power source will

- be a CTG
- be capable of automatically starting, accelerating to rated speed, of reaching nominal voltage, and to begin accepting load within two minutes of receipt of its start signal
- be capable of being manually connected to Class 1E 6.9 kV buses and safety-related loads
- be capable of being manually reconfigured such that non safety investment protection loads can be shed and safety-related shutdown loads can be connected (via any one of the Class 1E 6.9 kV buses) from the main control room within ten minutes
- be a self-contained unit equipped with its own auxiliary control and support systems such that external ac power is not required for its operation
- be physically and electrically independent and diverse from the Class 1E standby diesel generators such that weather-related failures, common cause failures, or single point vulnerabilities are minimized to the extent practicable
- be electrically isolated from each 6.9 kV bus by two normally open circuit breakers in series (one Class 1E and one non-Class 1E)
- require DC control power from the Class 1E and non-Class 1E dc power systems so that the AAC power source can be connected to the Class 1E 6.9 kV bus (to close the two normally open circuit breakers) from the control room within the 10 minute time requirement
- be capable of operating during and after a station blackout without any ac support systems powered from the preferred power supply or the blacked-out units Class 1E power sources affected by the event
- have sufficient capacity and capability to power one safety-related bus within 10 minutes of the onset of a station blackout, such that the plant safety systems will be capable of maintaining core cooling and containment integrity
- not normally or automatically be connected to the off-site power sources or the on-site Class 1E 6.9 kV buses thus minimizing the possibility of a common cause failure
- be designed to assume non-safety plant investment protection (PIP) loads automatically, to shed loads manually, and assume safety system loads manually while maintaining voltage and frequency within design requirements of safety system loads
- be capable of powering required shutdown loads with margin
- be capable of powering HVAC systems, chillers, battery chargers, and other support/auxiliary equipment during the station blackout event such that the environment during and following a station blackout event will not exceed the environment for which the

equipment is designed and the habitable environment for personnel

- undergo factory testing, similar to the Class 1E standby diesel generator, to demonstrate its ability to reliably start, accelerate to rated speed, voltage, and supply power within two minutes
- be subject to site acceptance testing, periodic preventive maintenance, inspection, testing and operational reliability assurance program goals
- be designed to quality assurance requirements commensurate with its importance to safety
- be located above the maximum flood level in the turbine building
- be provided with an oil storage and transfer system that will be physically and mechanically independent of the Class 1E standby diesel generator oil storage and transfer system
- have its fuel sampled and analyzed consistent with applicable standards
- have sufficient fuel oil stored on-site to support 7 days operation
- be capable of being periodically inspected, tested, and maintained

In addition, GE indicated that required plant core cooling and containment integrity during the station blackout duration (10 minutes) will not depend on any ac power sources.

Section 50.63(c)(2) of 10 CFR Part 50 requires (1) the time required for startup and alignment of the AAC to required shutdown equipment be demonstrated by test, (2) the ACC source have sufficient capacity and capability to permit required shutdown equipment to perform their required function, and (3) the AAC can be started and aligned within 10 minutes to the buses that distribute power to required shutdown equipment.

The staff concludes that a design, which meets the above described commitments, will have sufficient capacity, capability, and testability to provide power to required shutdown loads within specified station blackout time limitations. The design, therefore, meets the above defined requirements of Section 50.63(c)(2) of 10 CFR Part 50 and is acceptable. GE has included the above design commitments in Section 8.3.1.1.7(9) of SSAR Amendment 32, which is acceptable.

GE also indicated that the COL applicant must include periodic testing and/or analysis to verify the adequacy of the CTG to meet AAC requirements for station blackout in appropriate plant operating procedures. GE included this requirement in Section 9.5.13.19 of SSAR Amendment 32, which is acceptable.

Certified Design Material

The adequacy and acceptability of the ABWR design descriptions and ITAAC for the AAC are evaluated in Section 14.3 of this report.

9 AUXILIARY SYSTEMS

9.1 Fuel Storage and Handling

9.1.1 New Fuel Storage

The staff reviewed the new fuel storage capability in accordance with Standard Review Plan (SRP) Section 9.1.1. Staff acceptance of the new fuel storage facility design is based on the design meeting the requirements of General Design Criteria (GDC) 2 as it relates to the ability of structures housing the facility and the facility itself to withstand the effects of natural phenomena such as earthquakes; the position in Regulatory Guide (RG) 1.29, "Seismic Design Classification," Revision 3, Section C, which lists the systems which need to be designed to operate following a safe shutdown earthquake (SSE); GDC 5 as it relates to shared structures, systems, and components (SSCs) important to safety being capable of performing required safety functions; GDC 61 as it relates to the facility design for fuel storage; and GDC 62 as it relates to the prevention of criticality.

New fuel storage is provided in the new fuel storage vault located in the reactor building. The vault will contain storage racks for up to 40 percent of one full core fuel load. New fuel normally will be in dry storage; however, the storage racks can be used in either a wet or dry mode. The storage racks, vault, and the reactor building that houses the facilities are designed to seismic Category I criteria. The reactor building also is designed to provide protection against flooding and tornado missiles as evaluated in Sections 3.4.1 and 3.5.2 of this report, respectively. Therefore, the design satisfies the requirements of GDC 2 and the guidelines of Position C.1 of RG 1.29, Revision 3 because the reactor building provides the required protection.

The ABWR is designed as a single-unit facility. Therefore, the requirements of GDC 5 regarding the sharing of SSCs are not applicable. An application for a multi-unit facility will require review of the design for compliance with GDC 5.

The facility is designed to store unirradiated, low-emission fuel assemblies. Accidental damage to the fuel would release only minor amounts of radioactivity. These amounts would be treated by the standby gas treatment system (SGTS) which will limit any possible release of radioactivity from a fuel-handling accident to below acceptable values. Thus, the design meets the requirements of GDC 61 regarding appropriate containment, confinement, and filtering systems for a fuel storage facility.

The new fuel racks are not located near any high-energy lines, rotating machinery, or non seismic Category I SSCs

and thus are protected from internally-generated missiles and the effects of pipe breaks by physical separation. Applicants referencing the ABWR design will be required to provide features which prevent placement of a new fuel assembly in other than its prescribed location. Because the racks are freestanding (i.e., no supports above the base of the racks), the racks and rack support structure provide the necessary dynamic stability for the racks, thus preventing them from tipping.

The new fuel racks are designed to ensure that k_{eff} does not exceed 0.95 under all normal and abnormal conditions. The standard safety analysis report (SSAR) states that the new fuel storage racks are purchased equipment. In the draft safety evaluation report (DSER) (SECY-91-235), interface requirements were identified for the combined license (COL) applicant to provide:

- An analysis of design details that prevent inadvertent placement of a fuel assembly in other than prescribed locations.
- Confirmatory dynamic and impact analyses. The input excitation for these analyses should utilize the horizontal and vertical response spectra provided in SSAR Section 3A.10.2.
- An analysis showing that the design of the new fuel storage racks will be such that k_{eff} will not exceed 0.98 with a fuel load of the highest anticipated reactivity, assuming optimum moderator conditions (foam, small droplets, spray, or fogging), as described in SRP Section 9.1.1.

In the DSER, the staff identified these requirements as Interface Information Items 28 and 29. They were later reclassified as Open Item 9.1.1-1 in the draft final safety evaluation report (DFSER). Upon further evaluation, the staff determined that these requirements can be accomplished by COL actions and are not needed for design certification. SSAR Sections 9.1.6.1 and 9.1.6.2 state that the COL applicant shall provide a confirmatory criticality analysis and dynamic and impact analyses of the new fuel storage racks. This is acceptable as information which will be evaluated during the COL review. On the basis of the above, DFSER Open Item 9.1.1-1 and DFSER Interface Information Items 28 and 29 are resolved. Therefore, the staff concludes that design commitments in the SSAR meet GDC 62 as it relates to prevention of inadvertent criticality for fuel storage facilities.

In the DFSER, the staff requested GE to provide adequate design description and inspections, tests, analyses, and acceptance criteria (ITAAC) for the new fuel storage facility. This was part of DFSER Open Item 9.1.2-2. GE

Auxiliary Systems

submitted a revised set of design description and ITAAC. The adequacy and acceptability of the design description and the ITAAC are evaluated in Section 14.3 of this report. On the basis of the above, the applicable part of DFSE Open Item 9.1.2-2 is resolved.

The new fuel facility includes the fuel assembly storage racks and the concrete storage vault that contains the storage racks. The staff reviewed the applicant's proposed design criteria, design bases, and seismic classification for the new fuel storage facility related to the provisions necessary to maintain a subcritical array. The staff concludes that the design of the new fuel storage facility and supporting systems is acceptable and meets the requirements of GDC 2 as it relates to the ability of structures housing the facility (the reactor building) and the facility itself (designed to seismic Category I) to withstand the effects of natural phenomena such as earthquakes and GDC 61 as it relates to the facility design for fuel storage. GE adequately incorporated a COL action item requiring the applicant referencing the ABWR to provide information to ensure compliance with GDC 62 during the COL review. Therefore, the staff concludes that the design and related commitments comply with the guidelines of SRP 9.1.1 and are acceptable.

9.1.2 Spent Fuel Storage

The staff reviewed the ABWR spent fuel storage facility in accordance with the guidelines of SRP Section 9.1.2 to verify that it meets the requirements of GDC 2 as it relates to structures housing the facility and the facility itself being capable to withstand the effects of natural phenomena; GDC 4 as it relates to structures housing the facility and the facility itself being capable of withstanding the effects of environmental conditions, missiles, pipe whip, and jet impingement forces associated with pipe breaks; GDC 5 as it relates to shared SSCs important to safety being capable of performing required safety functions, GDC 61 as it relates to the facility design for fuel storage and handling of radioactive materials, GDC 62 as it relates to prevention of criticality, and GDC 63 as it relates to monitoring systems to detect conditions that could result in the loss of decay heat removal capabilities, to detect excessive radiation levels, and to initiate appropriate safety actions.

The spent fuel storage facility consists of fuel storage racks contained in the spent fuel storage pool in the reactor building. The spent fuel pool can store 270 percent of one full core load, in accordance with American Nuclear Society (ANS) 57.2 for the capacity of a single-unit facility. As stated in SSAR Section 9.1.4.2.8, defective fuel assemblies are placed in special fuel storage containers which are stored in equipment storage racks. Both the containers and the racks are designed for defective fuel.

The reactor building housing the facility is designed to seismic Category I criteria, as are the storage racks and other fuel storage facilities, including the gates between the spent fuel pool and other areas. In the DSER (SECY-91-235), the staff commented on the lack of information in the SSAR regarding the seismic classification of the spent fuel pool liner. This was DSER Open Item 44. Table 3.2-1 of the SSAR states that the liner will be seismic Category I. This meets the guidelines of SRP Section 9.1.2 and resolved DSER Open Item 44.

Applicants referencing the ABWR design will be required to provide features which prevent placement of a spent fuel assembly in other than its prescribed location. Because the racks are freestanding (i.e., no supports above the base of the racks) the racks and rack support structure provide the necessary dynamic stability for the racks, thus preventing them from tipping.

The reactor building is also designed to provide protection against flooding and tornado missiles as discussed in SSAR Sections 3.4.1 and 3.5.2 and evaluated in Sections 3.4.1 and 3.5.2 of this report. Therefore, the design satisfies the requirements of GDC 2 for protection of safety-related SSCs from natural phenomena and the guidelines of Position C.3 of RG 1.13, "Spent Fuel Storage Facility Design Basis," Revision 1, Positions C.1 and C.2 of RG 1.29, and Positions C.1 through C.3 of RG 1.117, "Tornado Design Classification," Revision 1.

The spent fuel pool is not located near any high-energy lines, rotating machinery, or non seismic Category I SSCs, and thus is protected from internally-generated missiles and the effects of pipe breaks by physical separation. The reactor building provides protection against externally generated missiles. Therefore, the design of the spent fuel facility satisfies GDC 4 and Positions C.1 through C.3 of RG 1.117.

The ABWR is designed as a single-unit facility. Therefore, the requirements of GDC 5 regarding the sharing of SSCs are not applicable. An application for a multi-unit facility will require review of the design for compliance with GDC 5.

Accidental damage to the fuel could release radioactive material. These amounts would be treated by the SGTS to limit any possible release of radioactivity from a fuel-handling accident to below acceptable values. Thus, the design meets the requirements of GDC 61 regarding appropriate containment, confinement, and filtering systems.

The fuel storage racks are designed to meet design requirements specifying that the fuel assemblies will be stored in

an array that limits k_{eff} to 0.95 or less under all normal and abnormal conditions, such as a dropped fuel assembly, earthquake, or stuck fuel assembly. The SSAR specifies the design requirements.

In the DSER (SECY-91-235), the staff included interface requirements, that the COL applicant's purchase specification for the spent fuel storage racks should require the vendor to submit:

- Confirmatory criticality analysis, including the uncertainty value and associated probability and confidence level for the k_{eff} value determined by the analysis.
- Confirmatory load drop analysis, including the free fall of a fuel assembly and its associated handling tool.

These were identified as Interface Information Item 30 in the DSER and later reclassified as Open Item 9.1.2-1 in the DFSER. Upon further evaluation, the staff determined that these requirements can be accomplished by COL actions and are not needed to be met for design certification. SSAR Sections 9.1.6.3 and 9.1.6.4 state that the COL applicant shall provide a confirmatory criticality analysis and load drop analysis of the spent fuel storage racks. This is acceptable. On the basis of the above, DFSER Open Item 9.1.2-1 and DSER Interface Information Item 30 are resolved. Therefore, the staff concludes that the design and associated commitments meet GDC 62 as it relates to prevention of inadvertent criticality.

In the DSER, the staff determined that the design of the storage pool includes the provisions for radiation monitoring systems described in SSAR Section 11.5 that satisfy, in part, the requirements of GDC 63 regarding the monitoring of spent fuel. In the DSER, the staff identified, as Open Item 45, a lack of information regarding pool level indication. Subsequently, GE provided additional information which states that the pool level will be monitored and alarmed (on high and low level) locally and in the control room. Leakage flow detectors in the pool drains and pool liners are also provided and alarmed in the control room. The staff concludes that these provisions will provide adequate monitoring capability and meet the requirements of GDC 63. Therefore, DSER Open Item 45 is resolved.

In the DFSER, the staff requested GE to provide adequate design description and the ITAAC relating to the spent fuel storage facility. This was part of DFSER Open Item 9.1.2-2. Subsequently, GE provided a revised set of design description and ITAAC. The adequacy and acceptability of the design description and the ITAAC are evaluated in Section 14.3 of this report. On the basis of

the above, the applicable part of DFSER Open Item 9.1.2-2 is resolved.

The spent fuel storage facility includes the spent fuel storage racks and the spent fuel storage pool that contains the racks. The staff reviewed the design criteria, design bases, and safety classification for the spent fuel storage facility and the provisions necessary to maintain a subcritical array. The staff concludes that the design and related commitments of the spent fuel storage facility and supporting systems conform with the Commission's regulations as set forth in GDC 2, 4, 5, 61, and 63. GE adequately incorporated a COL action item requiring the applicant referencing the ABWR to submit information to ensure compliance with GDC 62.

The staff concludes that the spent fuel storage design and commitments meet the guidelines of SRP 9.1.2 and the requirements of GDC 2 as it relates to structures housing the facility and the facility itself being capable of withstanding the effects of natural phenomena; GDC 4 as it relates to structures housing the facility and the facility itself being capable of withstanding the effects of environmental conditions, missiles, pipe whip, and jet impingement forces associated with pipe breaks; GDC 5 as it relates to shared SSCs important to safety being capable of performing required safety functions; GDC 61 as it relates to the facility design for fuel storage and handling of radioactive materials; GDC 62 as it relates to prevention of inadvertent criticality; and GDC 63 as it relates to monitoring systems to detect conditions that could result in the loss of decay heat removal capabilities, to detect excessive radiation levels, and to initiate appropriate safety actions. The design and related commitments, therefore, are acceptable.

9.1.3 Spent Fuel Pool Cooling and Cleanup System

The staff reviewed the spent fuel pool cooling and cleanup (FPC) system and its makeup system in accordance with SRP Section 9.1.3 to verify that they meet the requirements of GDC 2 as it relates to structures housing the system and the system itself being capable of withstanding the effects of natural phenomena; GDC 4 as it relates to structures housing the system and the system itself being capable of withstanding the effects of external missiles; GDC 5 as it relates to shared SSCs important to safety being capable of performing required safety functions; GDC 44 as it relates to the system's ability to reliably transfer heat loads from safety-related SSCs (including suitable redundancy and isolability); GDC 45 and 46 as they relate to inspection and functional testing, respectively; GDC 61 as it relates to the facility design for fuel storage and handling of radioactive materials; GDC 63 as it relates to monitoring systems provided to detect

Auxiliary Systems

conditions that could result in the loss of decay heat removal capabilities, to detect excessive radiation levels, and to initiate appropriate safety actions; and 10 CFR Part 20 as it relates to keeping radiation doses as low as reasonably achievable (ALARA).

The FPC system is a non-safety-related system designed to remove decay heat generated by spent fuel assemblies in the spent fuel storage pool; to maintain water quality and clarity; and to remove corrosion products, fission products, and other impurities from the pool water. The system consists of all components and piping from the system inlet at the fuel pool to the system outlet at the fuel pool, piping used to carry the fuel pool makeup water, and the cleanup filter/demineralizers (shared with the suppression pool cleanup (SPCU) system) to the point of discharge to the radwaste system. Specifically, the system includes two 50-percent capacity circulating pumps, two 50-percent capacity heat exchangers, two filter/demineralizers, one postdemineralizer strainer, two skimmer surge tanks, piping, valves, controls, and instrumentation. The pool water is circulated by means of overflow through skimmers around the periphery of the pool and a scupper at the end of the transfer canal; the overflow is collected in the surge tanks. Normally, one of the two circulating pumps draws water from these tanks and discharges it through a common header to either of two filter/demineralizers connected in parallel. System flow then passes through two heat exchangers in parallel, cooled by water from the reactor building cooling water (RCW) system, and returns to the spent fuel storage pool. The system includes a bypass line around the cleanup portion of the system (the filter/demineralizers). Circulating pumps can be powered from the combustion turbine generators (CTGs) if normal power is not available.

The requirements of GDC 5 regarding the sharing of SSCs do not apply because the ABWR is designed as a single-unit facility. An application for a multi-unit facility will require review of the design for compliance with GDC 5.

Filtering and ion exchange together maintain clarity and purity of pool water. The filter/demineralizers maintain total corrosion product metals at 30 parts per billion (ppb) or less with a pH range of 5.6 to 8.6 at 25 °C (77 °F) for compatibility with fuel storage racks and other equipment. Conductivity is maintained at less than 1.2 micro siemens (μS)/cm (3.0 micro Mho/in.) at 25 °C and chlorides less than 20 ppb. Maintaining the pool water purity within these levels will ensure that there are no adverse chemical interactions with the fuel storage racks and other pool equipment supplied by the applicant. Each filter unit in the filter/demineralizer subsystem has adequate capacity to maintain the desired purity of the pool under normal operating conditions.

The FPC system is housed in the reactor building (secondary containment area), a seismic Category I structure designed to protect against flood and tornado missiles. Thus, the system meets Position C.2 of RG 1.13 for protection of this support system for the fuel storage facility from natural phenomena such as tornados, winds, and externally-generated missiles.

Normal makeup water is supplied to the storage pool by the non-safety-related makeup water (condensate) system (MUWC). Backup to the normal makeup will also be available from the nonsafety-related SPCU system. Additionally, an emergency safety-related, seismic Category I makeup water source to the spent fuel pool will be provided by FPC connections to the residual heat removal (RHR) system, which draws water from a safety-related water source, that is, the suppression pool. FPC system piping from the spent fuel pool to the RHR system is classified as safety related. In a letter dated July 23, 1990, GE stated that the FPC system, except for the filter demineralizers, is designed to seismic Category I, Quality Group (QG) C standards. Although the entire system is not classified as seismic Category I, QG C standards, the nonsafety-related portions of the system that could affect any SSCs important to safety if they failed during a seismic event, are designed to ensure their integrity under seismic loading. Thus, the system meets Positions C.1 and C.2 of RG 1.29 with respect to the seismic design of the safety-related and nonsafety-related portions of the system, respectively.

There are no connections to the spent fuel pool that would cause the pool water to be drained below a safe shielding level. All lines that connect to the pool and extend below the safe level of the pool are equipped with siphon breakers, check valves, or other suitable devices to prevent inadvertent pool draining. Drainage paths are interconnected behind the liner welds. Leakage through the pool liner is collected in a drain system and transferred to an equipment drain tank. No piping connections penetrate the fuel pool liner to the fuel storage pool. The FPC system is designed so that no single failure, malfunction, or misoperation of the active components will uncover the stored fuel. Thus, the system meets Position C.6 of RG 1.13.

As mentioned above, the system includes a seismic Category I makeup water source that uses a safety-related piping segment of the FPC system return line. Check valves in the return lines that lead to the submerged FPC water return diffusers protect against backflow. Also, the system includes redundant safety-related isolation devices to facilitate isolation of the safety-related makeup portion of the system from the nonsafety-related SPCU system. The SPCU system makeup uses the same FPC system

safety-related piping segment that the RHR makeup uses, and is, therefore, equally vulnerable to a passive failure. In the DSER (SECY-91-235), the staff noted that such a failure will also affect the decay heat removal capability of both the FPC and the RHR systems, because both of these use the same safety-related piping segment of the FPC system return line. Subsequently, GE indicated that the pipe segments are protected from the effects of the failure of other components and systems. GE also submitted the results of an analysis that show that given a failure of this pipe segment, alternative cooling and makeup capabilities exist that will maintain the pool level and temperature within acceptable limits. The SSAR Section 9.1.6.10 clarifies the common piping through which FPC, RHR, and SPCU flow by stating that the COL applicant will ensure that the RHR system connections are adequately protected from the effects of pipe whip, internal flooding, internally-generated missiles, and the effects of a moderate energy pipe rupture in the vicinity. This is acceptable. Furthermore, responding to an early staff request (request for additional information (RAI) Question 410.37), GE stated that fire hoses will also be available to supply makeup water to the pool. This is also acceptable. The staff concludes that the FPC system design meets Position C.8 of RG 1.13.

Acceptance Criteria II.1.a and b of SRP Section 9.1.3 state that GDC 2 and 4 need not apply to the FPC system design if the design includes a safety-related makeup water system, including its source, and a safety-related fuel pool area ventilation and filtration system designed in accordance with the guidelines of RG 1.52. As noted above, a major portion of the FPC system is not safety-related and, therefore, does not meet Position C.1 of RG 1.13. The design relies instead on the SGTS, a safety-related ventilation and filtration system for the fuel pool area and relies on a safety-related makeup water system. In Section 6.5.1 of the DSER (SECY-91-235), the staff stated that the design of the SGTS had been identified as Open Item 30 in SECY-91-153 because GE had not demonstrated its compliance with all the applicable positions of RG 1.52 (e.g., redundancy of filters in the SGTS). The staff stated its concerns about safety-related makeup capability for the fuel pool and the SGTS design in the DSER (SECY-91-235) as part of Open Item 46 (Section 1.8 of the DSER (SECY-91-235) grouped four concerns of DSER Section 9.1.3 as Open Item 46.).

GE subsequently committed to revise the design of the SGTS to include a second redundant filter train. Because the safety-related makeup for the pool is not totally independent, but relies on the safety-related portion of the FPC system, GDC 2 and 4 are applicable for that portion of the FPC system. As stated above, the system is protected against the effects of adverse natural phenomena

by virtue of its location in the reactor building. Also, SSAR Sections 3.5.1 and 3.6.1, and SSAR Table 3.6-2, include spent fuel pool cooling for protection against the effects of missiles and piping failures. The staff noted that GE's submittal dated June 2, 1989, further stated that the major components of the FPC system are located in separately-shielded rooms and that barriers, restraints, and equipment compartments protect fuel pool cooling components against failure of high-energy piping systems. In the DSER (SECY-91-235), the staff stated, as part of Open Item 46, the need for GE to confirm that the safety-related portion of the system is protected from moderate-energy piping failures in the vicinity. The SSAR states that the safety-related portions of the system are protected from moderate-energy pipe breaks. This resolved this part of DSER (SECY-91-235) Open Item 46. GE also committed to modify the design of the SGTS to resolve the part of DSER (SECY-91-235) Open Item 46 associated with the SGTS design, and include the required ventilation and filtration system required as an alternative to designing the entire FPC system as a safety-grade system. Incorporation of the identified changes to the SGTS into the SSAR was DFSER Confirmatory Item 9.1.3-1. The staff reviewed the SSAR and concluded that the SGTS has been modified to include two redundant safety-grade filter trains and that the system complies with RG 1.52. This modification allows the ABWR FPC system to meet the alternate design guidance provided in SRP Sections 9.1.3.II.1.a and b. The SSAR clarifies the common piping through which FPC, RHR, and SPCU flow. SSAR Section 9.1.6.10 notes that the COL applicant will provide the protective features for this section of piping. The staff finds this approach and commitment acceptable. Therefore, DFSER Confirmatory Item 9.1.3-1 is resolved.

Based on the discussions above, the staff concludes that the design of the FPC system and the structures housing the system meet the criteria in GDC 2 and 4.

The staff based its evaluation of the capability of the FPC system to handle the decay heat load in the fuel pool on the guidelines in SRP Section 9.1.2, Paragraphs II.1.d.(4) and III.1.h. Each of the two FPC system heat exchangers is rated at $6.91\text{E}+9$ J/h ($6.55\text{E}+6$ BTU/hr) at the design temperature of 52°C (125°F). Based on an independent staff calculation of the spent fuel heat load using Branch Technical Position (BTP) ASB 9-2, the maximum normal heat load at 150 hours after shutdown is approximately $18\text{E}+9$ J/h ($1.7\text{E}+7$ BTU/hr). This heat load is based on the decay heat generated by one refueling load (35 percent of the core) at equilibrium conditions 150 hours after shutdown, plus a refueling load at 365 days, and one at 400 days after shutdown in accordance with SRP Section 9.1.3.III.1.h.iv for pools with storage capacity greater than 1.3 cores. The FPC heat exchangers are inadequately

Auxiliary Systems

sized to handle the decay heat load in the fuel pool at 150 hours after shutdown as outlined by SRP Section 9.1.3, Paragraphs II.1.d.(4) and III.1.h, to meet GDC 44. The heat load beyond the FPC heat exchanger capacity is to be handled by the RHR system, which includes a segment of the non-safety-related suction portion of the FPC system and a safety-related piping segment of the FPC system return line for performing its decay heat removal function. The RHR system also supplements the FPC system capabilities to maintain pool water temperature below 66 °C (150 °F) under the maximum abnormal heat load, that is, a full core offload. GE has based the FPC heat exchanger design on the heat load at 21 days after shutdown, the time at which the fuel transfer canal can be closed without requiring supplemental cooling. Independent calculations of the heat load at this time confirm the GE-calculated value of $12.0E+9$ J/h (1.14×10^7 BTU/hr). At this time after shutdown, the FPC system is adequately sized to handle the decay heat load of the fuel pool.

In the DSER (SECY-91-235), the staff stated that the FPC system design does not accommodate any single active failures. A single heat exchanger has insufficient heat removal capability to maintain the fuel pool temperature at an acceptable level at all times. GE also indicated that for some active single failures, using the RHR system may be necessary to limit the temperature of the fuel pool to less than 60 °C (140 °F). As a result, in addition to this concern, the DSER (SECY-91-235) identified as part of Open Item 46, the apparent undersizing of the FPC heat exchangers for the ABWR. The staff also identified as a concern the sole reliance on the RHR system to supplement the FPC system's normal maximum spent fuel heat load removal capability during certain situations (e.g., all times preceding 21 days after shutdown and after 21 days when there is a single active failure coincident with a loss of all offsite power). To address this concern, the SSAR provides additional commitments regarding the design of the RHR connection and alternative makeup capabilities so that the connection will be protected from the effects of high-energy and moderate-energy pipe breaks, and any other hazards to limit the likelihood of failure of the SPCU and RHR spent fuel pool makeup and cooling capability. The SSAR also addresses the capability to provide alternative makeup within an acceptable time limit. Fire water makeup to the spent fuel pool will be available within 30 minutes after the failure of the fuel pool cooling capability. GE's analysis showed that the spent fuel pool would not reach 60 °C (140 °F) under the maximum normal heat load at 21 days in less than this time. It concluded that over 6 hours would be required for the pool to heat up to this level. Under conditions of a full core off-load at 21 days after shutdown, the time to reach boiling was estimated to be over 16 hours. The staff's

independent calculations verify that the results are representative of the times available to provide additional makeup. The available time to realign the fire water system appears to be sufficient to make the configuration changes. Incorporation of this information into the SSAR was DFSER Confirmatory Item 9.1.3-2. The SSAR provides this information which adequately addresses the part of DSER Open Item 46 (SECY-91-235) regarding the use of the RHR system as an integral part of the FPC system from 150 hours after shutdown to 21 days after shutdown, and under the conditions of a loss of offsite power (LOOP) and a single active failure. This resolved the applicable part of DSER Open Item 46 and DFSER Confirmatory Item 9.1.3-2. Therefore, the staff concludes that the FPC system design complies with GDC 44 regarding the system's ability to transfer heat from systems important to safety under both normal and accident conditions and also complies with the RHR capability and coolant inventory maintenance requirements of GDC 61.

The pumps, heat exchangers, and the filter/demineralizers are each located in separate shielded rooms within the secondary containment portion of the reactor building. Individual components can be isolated from the rest of the system during operation. The system design allows for isolation and accessibility of the components. This complies with the requirements of GDC 61 for shielding, containment, and confinement.

One division of the FPC system (pump, heat exchanger, and filter-demineralizer) is normally in operation. The second pump will be operated periodically to ensure its operability or to allow the operating pump to be removed from service. Periodically, the FPC system components will be visually inspected. From this discussion, the staff concludes that the requirements of GDC 45 and 46 regarding the inspection and testing of cooling water systems are satisfied for the components in the safety-related portion of the FPC system. The design also complies with the testing and inspection requirements of GDC 61.

The components of the FPC system are accessible and can be isolated to allow for periodic testing. The FPC system is located within the secondary containment portion of the reactor building, thus providing adequate containment. The FPC system contains design features to prevent reduction in fuel storage coolant inventory under accident conditions in accordance with Position C.6 of RG 1.13. The decay heat removal capabilities of the FPC system are designed to meet the heat load requirements of the fuel pool at 21 days after shutdown, not the requirements at 150 hours after shutdown. Use of the RHR system is required to meet the heat load requirements of the FPC

system during the period between 150 hours and 21 days after shutdown. Additionally, as identified in the DSER, the FPC system is not capable of handling the decay heat load assuming both a LOOP and a single active failure. Again the RHR system, in conjunction with the operable portions of the FPC system, is the system used to meet the heat load requirements. As indicated previously, GE has provided information justifying this system arrangement. The design includes measures to ensure RHR system operability in the spent fuel cooling mode and alternative makeup and cooling capabilities. Therefore, the staff concludes that the FPC system meets all requirements of GDC 61.

The control room includes instrumentation to monitor the fuel pool level and temperature and area radiation for the fuel storage and handling areas. The control room also includes high and low fuel pool level and high-radiation alarms. Interconnected drainage paths behind the liner welds include instrumentation to detect and measure liner leaks. Several FPC system parameters are also displayed and recorded in the control room and locally. These include pump suction pressure and pump flow and system water discharge temperature. The pump suction pressure and flow are also alarmed in the control room. These monitoring and alarming devices enable plant personnel to quickly begin appropriate safety actions such as erecting temporary shields to reduce radiation, supplying makeup water to the pool through a remotely operated valve, and actuating the SGTS on a high radiation signal on the refueling floor). Indications of high system temperature allow the operator to manually bypass system flow from the filter-demineralizers to protect the resins from damage. Thus, the system instrumentation meets the requirements of GDC 63 regarding the monitoring of fuel storage facilities and the initiation of appropriate safety actions.

Based on its review as discussed above, the staff concludes the following:

1. Designing the FPC system for the ABWR as a non-safety-related system, except as stated above, meets the applicable acceptance criteria of SRP Section 9.1.3.
2. The cooling portion of the FPC system complies with the requirements of GDC 44 and 61 for decay heat removal, inspection and testing, containment and confinement of stored fuel, maintenance of fuel storage coolant inventory, and shielding requirements.
3. The safety-related portion of the FPC system complies with the requirements of GDC 4 for its protection from the effects of missiles and high- and moderate-energy piping failures in the vicinity.

4. The safety-related portion of the FPC system complies with the requirements of GDC 2 for its protection from the effects of adverse natural phenomena.
5. The cooling portion of the FPC system complies with the requirements of GDC 45, 46, 61, and 63, for inspection, testing, and monitoring.

In the DSER (SECY-91-235), the staff stated that conclusions 1, 2, and 3 are subject to:

1. GE's confirmation of protection of the system's safety-related portions from the effects of moderate-energy piping failures in the vicinity. The SSAR includes a commitment that the COL applicant will include features to protect the common portion of FPC piping from pipe and equipment failures. This commitment is acceptable.
2. Resolution of the staff's concerns relating to (a) the system's decay heat removal capability, and (b) the SGTS design as it pertains to the requirement for redundant filters. The information provided in the SSAR regarding the capability of the operators to align the fire water system to supply pool makeup and cooling and the temperature response of the pool water after the loss of all cooling, adequately addresses the staff's concern about the system's decay heat removal capability. The staff's concern regarding the design of the SGTS was identified as Confirmatory Item 9.1.3-3 in the DFSE. Subsequently, the SGTS design was modified (see SSAR Section 6.5.1 for a full discussion of the SGTS and Section 6.5.1 of this report for the staff evaluation of the SGTS). As a result of the design modifications, the staff concludes that the SGTS provides adequate redundancy. Therefore, Confirmatory Item 9.1.3-3 is resolved.

In the DSER, the staff raised the issue of the technical specification (TS) implications of using the RHR system as an integral part of the FPC system, as part of Open Item 46. In the DFSE, the staff stated that GE's response did not explain how the RHR system can be used for the maximum normal heat load removal without violating the TS requirements for availability of the system for other purposes in Mode 5 (refueling mode). This was DFSE Open Item 9.1.3-1. Subsequently, GE submitted information in which it explained that, during refueling, containment is not required to be operable unless fuel is being moved. Therefore, suppression pool cooling and wetwell/drywell spray is not required while in the refueling mode. In a situation where augmented fuel pool cooling may be required, the shutdown cooling function is not affected since in this mode, the pool gates are removed and the reactor and spent fuel pool are common. RHR in the

Auxiliary Systems

low-pressure flooders (LPFL) mode may be needed during refueling. In this case, TS require only two emergency core cooling system (ECCS) divisions to be operable. Five divisions are available in the ABWR design to meet this need (two divisions of high-pressure core flooders (HPCF) and three divisions of RHR). TS consider an RHR division aligned in the shutdown cooling mode to be considered operable if it can be manually realigned to the LPFL mode. The staff notes that the ABWR design allows this reconfiguration. Therefore, based on the evaluation above, the staff concludes that the ABWR design is capable of providing adequate availability of RHR for all required heat removal functions during all modes. This resolved DFSER Open Item 9.1.3-1.

The ACRS inquired about the ability of the ABWR design to initiate and/or maintain spent fuel pool cooling during and following an accident, including a LOCA. This question arose as a result of design concerns identified at an operating reactor in a 10 CFR Part 21 notification. Specifically, it was postulated that if a LOCA occurred, the resulting environmental conditions in the secondary containment would render the fuel pool cooling and cleanup system (FPC) equipment inoperable while, at the same time, preventing implementation of recovery actions. Should this occur, water in the spent fuel pool would boil, resulting in excessive condensation leading to flooded conditions. In addition, boiling conditions could render the standby gas treatment system (SGTS) inoperable, so that any radioactive material resulting from failed fuel in the pool (due to lack of pool cooling) could be released to the environment without first being processed by the SGTS.

The staff discussed the postulated scenario with GE and asked whether the ABWR design would be able to initiate and/or maintain spent fuel pool cooling during and following an accident, including a LOCA, in accordance with GDC 44 and 61. Subsequently, GE stated that the ABWR design can accommodate this accident for the following reasons:

- (1) GE determined that a break in the reactor water cleanup system (CUW) presents the most limiting environmental conditions inside secondary containment for all design-basis accidents, including a LOCA. Safety-related equipment located inside the secondary containment is environmentally qualified to remain functional given a CUW pipe failure inside the secondary containment. Therefore, this equipment would remain available to perform its safety-related functions following an accident.
- (2) In Section 9.1.3 of the SSAR, GE provided the results of a pool heatup analysis. The results

showed that, should pool cooling fail with the maximum abnormal heat load in the pool, it would take approximately 16 hours for the pool to reach boiling conditions. This allows sufficient time for operators to implement manual recovery actions (manipulation of manual valves, etc.) well before boiling conditions develop. The system connections to each of the two residual heat removal (RHR) system divisions connected to the FPC utilizes a motor-operated valve (MOV) on the FPC inlet from the RHR system and a manual valve on the FPC outlet to the RHR system. The inlet valves are remotely operated; the outlet valves, however, require the operator to manually manipulate the valves. GE states in the SSAR that these manual valves will be accessible following an accident in sufficient time to permit an operator to align the RHR system to prevent the SFP from boiling.

- (3) Two of the three safety-related RHR divisions are available to provide cooling water to the spent fuel pool. Either division is sufficient to cool the spent fuel pool given the worst-case heat load in the pool. Therefore, given a single failure of an RHR division, one remaining division can provide spent fuel pool cooling while the other provides cooling for the reactor.
- (4) In the unlikely event that pool boiling were to occur, safety-related makeup water can be provided from the suppression pool through either of the MOVs on the FPC inlet from RHR. The resulting environmental conditions are bounded by the worst-case environmental conditions postulated in the CUW line break discussed earlier. Each division of safety-related equipment inside the secondary containment is physically and electrically separated to prevent flood conditions from affecting more than one division of safety-related equipment.
- (5) The system is not safety related except for those portions from RHR to the FPC system and from the FPC system to the RHR system. However, the FPC system is designed to seismic Category I standards (with the exception of the filter-demineralizers) to ensure it remains functional during seismic events. Alternative means of power to the system is available through the combustion turbine generator (CTG) which is the alternate ac (AAC) power source for the ABWR design. Should a loss of preferred power (LOPP) or station blackout (SBO) occur, the CTG can be used to provide power to the FPC system. Therefore, the system will be available to provide spent fuel pool cooling under both normal and accident conditions.

On the basis of this information, the staff concludes that the ABWR design ensures that cooling can be initiated and/or maintained during both normal and accident conditions, including a LOCA.

In the DFSER, the staff requested GE to submit adequate design description and the ITAAC relating to the FPC system. This was DFSER Open Item 9.1.3-2. The adequacy and acceptability of the design description and the ITAAC are evaluated in Section 14.3 of this report. On the basis of the above, DFSER Open Item 9.1.3-2 is resolved.

The staff's review of the FPC system included all components and piping of the system from the inlet to and the exit from the storage pool, the seismic Category I makeup source and piping used for the fuel pool makeup, the cleanup system filter/demineralizers, and the regenerative process to the point of discharge to the radwaste system. The scope of the FPC system review included layout drawings, process flow diagrams, piping and instrumentation diagrams, and descriptive information for the system and supporting systems that are essential to safe operation. The cooling portion of the system and the emergency primary makeup system are designed to seismic Category I, QG C requirements, since they are necessary to remove decay heat from the spent fuel and to prevent fuel damage that could lead to unacceptable releases of radioactivity. The staff concludes that the design of the FPC system and its makeup systems meet the guidelines of SRP 9.1.3 and the requirements of GDC 2, 4, 5, 44, 45, 46, 61, and 63, subject to incorporation of information regarding protection of the filter-demineralizer resins.

9.1.4 Light Load-Handling System (Related to Refueling)

The light load-handling system (LLHS) provides the means of transporting, handling and storing fuel (both new and spent fuel) in the reactor building. The staff reviewed the fuel-handling system, in accordance with the guidelines of SRP Section 9.1.4. Staff acceptance of the LLHS is based on meeting the requirements of GDC 2 as it relates to structures housing the system, and the system itself being capable of withstanding the effects of natural phenomena; GDC 5 as it relates to shared SSCs important to safety being capable of performing required safety functions; GDC 61 as it relates to radioactivity release as a result of fuel damage, and the avoidance of excessive personnel radiation exposure; and GDC 62 as it relates to prevention of inadvertent criticality.

The transfer of new fuel assemblies between the uncrating area and the new fuel inspection stand and/or the new fuel storage vault is accomplished using a 4500-kg (5-ton)

auxiliary hook on the reactor building crane equipped with a suitable grapple. A 450-kg (1000-lb) auxiliary hoist on the reactor building crane is used with an auxiliary fuel grapple to transfer new fuel from the new fuel vault to the fuel storage pool. From there, the fuel is handled by the telescoping grapples on the refueling platform or auxiliary hoists.

The refueling machine is a gantry crane used to transport fuel and reactor components to and from pool storage and the reactor vessel. The platform spans the fuel storage and vessel pools. A telescoping mast and grapple suspended from a trolley system is used to lift and orient fuel bundles for placement in the core or storage rack. The fuel grapple hoist has a redundant load path so that no single component failure can result in a fuel bundle drop. Interlocks on the platform (1) prevent hoisting a fuel assembly over the vessel with a control rod removed, (2) prevent collision with fuel pool walls or other structures, (3) limit travel on the fuel grapple, (4) interlock grapple hook engagement with hoist load and hoist up power, and (5) ensure correct sequencing of the transfer operation in the automatic or manual mode.

The refueling machine also has two (4.71 kN (1060 lbf) and 9.87 kN (2200 lbf)) auxiliary hoists. The hoists can be used normally with appropriate grapples to handle control rods, guide tubes, fuel support pieces, sources, and other internals of the core.

The refueling machine is designed structurally to seismic Category I standards and the entire system is housed within the reactor building, which is a seismic Category I structure designed to protect against flood and tornados.

In the DSER (SECY-91-235), the staff identified as Open Item 47, the impact of the failure of the fuel inspection stand during a safe-shutdown earthquake (SSE) because the stand was not classified as seismic Category I, and the radiological and criticality impact of the fall of the new fuel stand was not satisfactorily addressed in the SSAR. The SSAR commits the new fuel inspection stand to be firmly attached to a wall so that it does not fall or dump personnel into the fuel pool during an earthquake. This is acceptable and resolved DSER Open Item 47.

The final inspection stand design meets the requirements of GDC 2 and the guidelines of Positions C.1 and C.6 of RG 1.13, and Positions C.1 and C.2 of RG 1.29, relating to the protection of safety-related equipment and spent fuel from the effects of earthquakes, because it is housed and attached to a seismic Category I structure.

The reactor building crane main hook is used to move the spent fuel cask, and the auxiliary hook is used to move

Auxiliary Systems

new fuel from the new fuel vault to the spent fuel storage pool. Interlocks and procedures prevent the main hook of the reactor building crane from traversing over the spent fuel pool or the new fuel storage vault while carrying a heavy load. This is acceptable.

The requirements of GDC 5 for sharing SSCs do not apply because the ABWR is designed as a single-unit facility. An application for a multi-unit facility will require review of the design for compliance with GDC 5.

In Section 9.1.2 of the DSER (SECY-91-235), the staff stated that the COL applicant's purchase specification should include an interface requirement for the vendor to submit a confirmatory analysis of the consequences of dropping a fuel assembly and its associated handling tool from above the spent fuel storage racks. The staff determined that this requirement can be accomplished by COL actions as resolved in Section 9.1.2 of this report.

This additional information will ensure compliance with GDC 62 regarding protection of the fuel from inadvertent criticality.

The fuel-handling machine grapple retracts to a maximum position which ensures that the fuel will not be lifted beyond a safe-shielding height. Various electrical and mechanical interlocks prevent excessive cable load, movement over the core with a control rod removed, withdrawal of a control rod with fuel of the core, and collisions of fuel with surrounding structures. The refueling machine has redundant load paths and is designed to retain its load during a seismic event. Finally, the ABWR design includes instrumentation to monitor radiation levels on the refueling floor and initiated protective actions on a high-radiation level (realignment of heating, ventilation, and air conditioning (HVAC) systems and SGTS initiation). The staff concludes that the design of the light load-handling system meets the guidelines of Position C.3 or RG 1.13 and ANS 57.1, and the requirements of GDC 61 for the release of radioactivity resulting from fuel damage and protection of personnel from excessive radiation exposure.

In the DFSE, the staff requested GE to submit adequate design description and the ITAAC for the LLHS. This was DFSE Open Item 9.1.4-1. GE submitted a revised set of design description and ITAAC. The adequacy and acceptability of the design description and the ITAAC are evaluated in Section 14.3 of this report. On the basis of the above, DFSE Open Item 9.1.4-1 is resolved.

The LLHS includes all components and equipment used in moving fuel and other related light loads between the receiving area, storage areas, and reactor vessel. Based on

the review of the applicant's proposed design criteria and design bases for the LLHS, and the requirements for the safe operation of the LLHS, as discussed above, the staff concludes that the design of the LLHS and supporting systems are in conformance with the guidelines of SRP 9.1.4, and the Commission's regulations as set forth in GDC 2 (protection from natural phenomena), GDC 5 (sharing of SSCs), and GDC 61 (fuel storage handling and radioactivity control) and will conform to GDC 62 (protection of criticality in fuel storage handling), and are acceptable.

9.1.5 Overhead Heavy Load-Handling System

Inadvertent operations and equipment malfunctions caused by a load drop during critical load-handling operations could cause (1) a release of radioactivity to the environment above acceptable limits, (2) a criticality accident, (3) the inability to cool fuel within the reactor vessel or spent fuel pool, or (4) the inability to achieve safe shutdown of the reactor when needed. Therefore, critical load-handling equipment and operations are required to prevent these problems by built-in design features and operating procedures. Additionally, safe handling of loads includes design considerations for maintaining occupational radiation exposures as low as practicable during transportation and handling. The overhead heavy load-handling system (OHLHS) consists of all components and equipment used in moving all heavy loads, that is, loads weighing more than one fuel assembly and its associated handling device.

The staff reviewed the system in accordance with SRP Section 9.1.5. Staff acceptance of the OHLHS design was based on meeting the requirements of GDC 2 as it relates to the ability of structures, equipment, and mechanisms to withstand the effects of natural phenomena; GDC 4 as it relates to protection of safety-related equipment from the effects of dropped loads; GDC 5 as it relates to sharing of equipment and components important to safety; and GDC 61 as it relates to the safe handling and storage of fuel. In addition, the acceptance criteria for the OHLHS includes meeting the guidelines of ANS 57.1, "Design Requirements for Light Water Reactor Fuel Handling System," and 57.2, "Design Objectives for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Plants." The staff used the guidelines contained in the SRP Section III titled "Review Procedures" and in NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants."

The OHLHS includes equipment necessary for the safe disassembly and reassembly of the reactor vessel head and internals during refueling operations and for the safe handling of the spent fuel cask. The reactor building crane

will handle heavy loads in the containment and fuel-handling area of the reactor building. This single-failure-proof crane will handle the reactor vessel head, shroud head, steam separator, steam dryer, pool gates, new fuel from the reactor building entry hatch to new fuel storage, spent fuel shipping cask, some reactor internal pump (RIP) components during pump servicing, and other miscellaneous loads during operation and outage. Also, the OHLHS includes reactor servicing equipment, upper and lower drywell servicing equipment, and main steam tunnel (MST) area servicing equipment, which are used for safe handling of main steam isolation valves (MSIVs), safety/relief valves (SRVs), RIP motors, heat exchangers and pump components, control rod blades and guide tubes, and fine motion control rod drive (FMCRD) components during their removal and reinstallation or replacement. This servicing equipment includes among other things, monorail and its hoist, transportation carts, equipment hatchway hoist, steam tunnel crane and its hoist, servicing platform, refueling platform, equipment platform, and lower drywell RIP hoist.

The OHLHS equipment described above is housed in the reactor building in which the associated heavy load-handling operations are performed. The building is a flood-protected and tornado-protected structure and thus, the OHLHS described above is protected against the effects of adverse natural phenomena. The spent fuel storage facility and the new fuel storage vault are seismic Category I. Also, as stated in Section 9.1.3 of this report, the spent fuel storage facility meets Position C.6 of RG 1.13, "Spent Fuel Storage Facility Design Basis." The OHLHS equipment and components are not seismic Category I except for the upper and lower drywell servicing equipment, refueling platform, equipment platform, and reactor service platform. Though the load-handling equipment need not be seismic Category I, the non seismic Category I equipment that can adversely impact SSCs important to safety should they fail during a seismic event, have to meet Position C.2 of RG 1.29. SSAR Table 3.2-1 states that the reactor building and refueling platform cranes are designed to hold their positions during an SSE. The table further indicates that the non seismic Category I system equipment that can adversely impact safety-related SSCs during an SSE, are designed to ensure their integrity under seismic loading resulting from an SSE. The staff identified in the DSER (SECY-91-235), as part of Open Item 48, (Section 1.8 of the DSER grouped eight concerns of DSER Section 9.1.5 as Open Item 48. Five are addressed in the items that follow. The remaining three are addressed in subsequent paragraphs.) several concerns pertaining to compliance of the OHLHS with GDC 2 as it relates to protection of safety-related equipment from the effects of adverse natural phenomena:

1. In the DSER (SECY-91-235), the staff noted that Note x for SSAR Table 3.2-1 needed to be clarified to show that the reactor building and refueling platform cranes will hold their loads under the dynamic effects of an SSE.

Note x of Table 3.2-1, SSAR Amendment 20, stated that the cranes are capable of holding their loads under operating-basis earthquake (OBE) conditions but did not address the SSE. This part of DSER Open Item 48 was not resolved and was reclassified as Open Item 9.1.5-1 in the DFSER. In the SSAR, Note x of Table 3.2-1, states that the cranes are designed to hold their load and maintain their position during an SSE. This is acceptable and resolved DFSER Open Item 9.1.5-1.

2. In the DSER (SECY-91-235), the staff noted that the above criterion should be applied for all other applicable (i.e., affecting safe-shutdown equipment) non seismic Category I load-handling equipment.

In response, GE indicated that the use of the steam tunnel handling equipment is limited to times when the reactor is shut down. Failure of the handling equipment at this time would not result in the loss of safety functions (identified in SSAR Amendment 17). The rationale for not requiring this equipment to have the capability of holding its loads during an SSE is acceptable. However, sufficient justification for the lack of the application of a seismic criterion, such as that identified in item 1 above, has not been provided for the remaining devices in the control building, secondary containment, and clean zone of the reactor building. This part of DSER Open Item 48 was not resolved and was reclassified as Open Item 9.1.5-2 in the DFSER. GE later added Amendment 24 to the SSAR which stated that, with the exception of containment, the MST, and the refueling floor, no safety-related components of one division shall be routed over another safety-related division. Therefore, a dropped load cannot adversely effect more than one safety-related division.

GE also indicated that the amount of safety-related equipment present inside primary containment, and the divisional equipment is widely separated (120 degrees) around the containment. Therefore, the likelihood of a dropped load damaging more than one train of safety-related equipment is minimized. Load-handling equipment in the MST is operated only during shutdown conditions. During this condition, systems located in the tunnel are isolated and not required to be operable. In addition, isolation valves inside the tunnel have redundancy with valves inside containment.

Auxiliary Systems

Therefore, a dropped load in the MST could damage components inside the tunnel, but the redundant components inside the containment will maintain the safety function. Finally, regarding the refueling floor, the reactor building crane meets the single-failure-proof design requirements of NUREG-0554 and is interlocked to prevent movement over the spent fuel storage pool. The SSAR incorporates the above information. The staff reviewed the information and concluded that the design features for the load-handling equipment in all certified buildings ensure that both new and spent fuel, as well as safety-related equipment, will not be damaged to such an extent as to inhibit or prevent safety functions. This is acceptable and resolved DFSER Open Item 9.1.5-2.

3. In the DSER (SECY-91-235), the staff noted that terms such as "refueling bridge crane," "automatic refueling machine," and "spent fuel handling crane" were used in the SSAR to represent possibly the same load-handling device. Also, it was not clear whether the ABWR used a jib crane and whether the fuel-handling platform was different from the refueling platform. Further, SSAR Tables 3.2-1 and 9.1-2 gave different seismic classifications for the refueling platform crane (automatic refueling machine) and the jib crane.

In the DFSER, the staff stated that many of the discrepancies had been corrected. However, neither the jib crane nor the refueling platform crane were referred to in the text of SSAR Section 9.1, although both were listed in SSAR Table 3.2-1. While the term "Refueling Bridge," was used only in Table 3.2-1, the term "Refueling Platform," was used throughout Section 9.1. The text of Section 9.1 appeared to be internally consistent. References to a refueling platform consistently referred to the fuel-handling equipment. The staff requested GE to clarify the remaining discrepancies in the nomenclature in SSAR Table 3.2-1 and Section 9.1. The staff reclassified this part of DSER Open Item 48 as Open Item 9.1.5-3 in the DFSER. Subsequently, GE provided clarifying information which corrected the remaining discrepancies and changed "refueling platform" to "refueling machine." This is acceptable and resolved DFSER Open Item 9.1.5-3.

4. The staff noted in the DSER (SECY-91-235) that SSAR Section 9.1.5.2.1 implies that the load-handling equipment for steam tunnel servicing (MSIVs and SRVs) is housed in the reactor building, while SSAR Table 3.2-1 shows the equipment to be located in "any other location." The staff requested GE to correct the location of the subject equipment, as appropriate and, if the location is "any other location," to state why

housing the equipment in a non seismic Category I structure is acceptable.

In response, SSAR Section 9.1.5.2.2.3, Amendment 17, stated that the MST servicing equipment is a permanently installed monorail system. Note c in SSAR Table 3.2-1, which indicated the MST tunnel servicing cranes and hoists are in "any other location," is interpreted to mean that the monorail is not in any of the locations specifically identified in Note c, and does not contradict the statement in SSAR Section 9.1.5. The clarification shows that the equipment is housed in a seismic Category I structure. This is acceptable and resolved this part of DSER Open Item 48.

5. The staff stated in the DSER (SECY-91-235) that SSAR Table 3.2-1 showed load-handling equipment for special service rooms being housed in the non seismic Category I radwaste and turbine buildings, and requested GE to clarify why such locations are acceptable.

In response, GE indicated that the load-handling equipment in the turbine and radwaste buildings will handle only the equipment in those structures. In the DFSER, the staff requested GE to incorporate this information into the SSAR by reclassifying it as Confirmatory Item 9.1.5-1 in the DFSER. The SSAR adds information in Sections 9.1.5.3 and 9.1.6.6 clarifying that all load-handling devices, including those in the radwaste building and turbine buildings, must be designed and procedures must be used which ensure that the loads they carry will not damage more than one division of a safety-related system. Further, the load-handling devices in the turbine and radwaste buildings are intended to handle only equipment in these buildings and that the loads will not be moved over safety-related equipment. Therefore, failure of this equipment or dropping of any loads moved by this equipment will not adversely affect safety-related SSCs. The staff finds this additional clarification acceptable. Therefore, DFSER Confirmatory Item 9.1.5-1 is resolved.

The OHLHS meets Positions C.1 and C.2 of RG 1.29, and Positions C.1 and C.6 of RG 1.13 for the spent fuel storage facility and therefore, complies with the requirements of GDC 2 with regard to protection of the system from the effects of adverse natural phenomena.

The spent fuel cask pool is separated from the spent fuel storage pool by a water-tight gate. Redundant safety interlocks and limit switches in the reactor building crane prevent transport of any heavy load, including the spent

fuel cask, over the spent fuel pool. Administrative control and a coverage area prevent transport of any heavy load over the new fuel storage vault. A dropped cask cannot, therefore, result in any fuel damage. Additionally, as mentioned above, the reactor building crane and the refueling platform crane are single-failure-proof cranes. The staff in the DSER (SECY-91-235) identified several concerns, as part of Open Item 48, pertaining to compliance of the OHLHS with the requirements of GDC 4 as it relates to protection of safety-related equipment from the effects of internally-generated missiles (i.e., dropped loads) during load-handling operations:

1. The staff stated in the DSER (SECY-91-235) that SSAR Section 9.1.4.3 incorrectly stated that the spent fuel-handling crane's lifting height for the spent fuel cask is limited to 30 feet on the operating floor, and that this should be corrected to refer to the reactor building crane.

GE subsequently modified the SSAR to eliminate inconsistent terms. This modification resolved this part of DSER Open Item 48.

2. The staff noted in the DSER (SECY-91-235) that SSAR Amendment 7 (response to RAI 410.43) gives contradictory statements: "While carrying heavy loads, such as the spent fuel cask, the reactor building crane is prohibited from moving the heavy load over the spent fuel pool," and "No other heavy loads other than the spent fuel cask, need to be carried above the top of the spent fuel storage pool." The staff requested GE to correct this as appropriate.

In response, GE committed to revise the response to RAI 410.43 to eliminate the contradictory statements regarding movement of the spent fuel cask. Incorporation of the revisions to the response to RAI 410.43 was identified as Confirmatory Item 9.1.5-2 in the DFSER. SSAR Section 9.1.5.2.1 states that the reactor building crane will be interlocked to prevent movement over the spent fuel pool while carrying heavy loads. This is acceptable and resolved DFSER Confirmatory Item 9.1.5-2.

3. The staff noted in the DSER (SECY-91-235), as part of DSER Open Item 48, that GE had not identified how safety-related equipment would be protected during all heavy load-handling operations involving non-single-failure-proof load lifting devices (e.g., MSIVs, SRVs, RIP motors and hoists, load-handling operations in control, radwaste, and turbine buildings).

In response, SSAR Amendment 20 identified the criteria to assess the safety impact of the failure of the

load-handling devices. The lower drywell load-lifting device does not carry loads over safety-related equipment. The failure of the upper drywell load-lifting device, used only during maintenance at shutdown, can affect only the one division of an ECCS. Safe-shutdown conditions can be maintained with the failure of the one ECCS division. The MST equipment is used only during shutdown, and its failure will not impact any safety-related equipment needed to maintain safe-shutdown conditions. Load-handling equipment in the turbine and radwaste buildings is not used near safety-related equipment because such equipment is restricted to use in these structures. However, the use of heavy load-handling equipment in the control building was not addressed. As discussed in the DFSER, all items identified in this portion of DSER Open Item 48 are resolved with the information provided by SSAR Amendment 20 except for the concern regarding the control building. Identification of how safety-related equipment would be protected during all heavy load-handling operations involving non-single-failure-proof load lifting devices in the control building was identified as Open Item 9.1.5-4 in the DFSER. Subsequently, GE provided clarifying information in SSAR Sections 9.1.5.3, 9.1.5.4, 9.1.5.5, and 9.1.6.6. The design of all heavy load-handling equipment will meet the guidelines of NUREG-0612, including the guidelines of NUREG-0554 regarding single-failure-proof cranes. Based on the additional information provided in these SSAR subsections, the staff concludes that safety-related equipment will be adequately protected during all heavy load-handling operations. Therefore, DFSER Open Item 9.1.5-4 is resolved.

Based on the above information, the staff concludes that the OHLHS design will meet the guidelines of Positions C.3 and C.5 of RG 1.13, with respect to protection of the spent fuel storage facility from the effects of internally-generated missiles and for the safe handling and storage of fuel, and further concludes that safety-related equipment is protected from the effects of internally-generated missiles during load-handling operations. Therefore, the OHLHS design for the ABWR will comply with GDC 4 as it relates to protection of safety-related equipment from internally-generated missiles.

The requirements of GDC 5 regarding the sharing of SSCs do not apply because the ABWR is designed as a single-unit facility. An application for a multi-unit facility will require review of the design for compliance with GDC 5.

As mentioned above, the reactor building and refueling platform cranes are single-failure-proof cranes. SSAR Section 9.1.5.5 further states that all the cranes, hoists, and related lifting devices for handling heavy loads either

Auxiliary Systems

satisfy the single-failure-proof guidelines of NUREG-0612, Section 5.1.6, including NUREG-0554, or evaluations are made to demonstrate compliance with guidelines of NUREG-0612, Section 5.1, including Sections 5.1.4 and 5.1.5. GE stated that the applicable components of the OHLHS will comply with the guidelines of the following industry standards: American National Standards Institute (ANSI) N14.6, ANSI B30.9, ANSI B30.10, ANSI B30.2, ANSI B30.16, ANSI B30.11, and Crane Manufacturers Association of America (CMAA) Specification 70. NUREG-0554 and NUREG-0612 recommend the standards specified in ANSI N14.6, ANSI B30.2, ANSI B30.9, and CMAA 70 for the design and performance of a nuclear power plant OHLHS. Additionally, GE states that the design of special lifting devices and slings will comply with the guidelines of NUREG-0612, Sections 5.1.1(4) and 5.1.1(5).

GE submitted a general description of the inspection, operation, maintenance, service, and test requirements for the OHLHS equipment. The COL applicant is responsible for supplying operating, maintenance, and test procedures and instruction manuals that will comply with NUREG-0612, Sections 5.1.1(2) and 5.1.1(6) for heavy load-handling equipment components (e.g., cranes, hoists, refueling platform). In accordance with the requirements of 10 CFR Part 50, Appendix B, the COL applicant will subject the OHLHS equipment to qualification load and performance tests, dimensional inspection, and nondestructive examination before the equipment is accepted by the applicant's quality assurance group. The applicant will also ensure that lifting components are appropriately inspected and tested before shipment, after receipt at the site, before use, and at periodic intervals. The tests will be conducted in accordance with the requirements of ANSI B30.2 and NUREG-0612, Section 5.1.1(6). For each item of equipment requiring servicing, the applicant will develop an interface control diagram (ICD) delineating the space around the equipment required for servicing. The ICD will include pull space for internal parts and access space for tools, handling equipment, and alignment requirements. The ICD will specify the weights of large removable parts, show the locations of their centers of gravity, and describe installed lifting accommodations such as eyes and trunnions. The COL applicant will develop safe load paths and routing plans for each heavy load to be handled, which will show, among other things, frequency of transportation and usage of the route. The safe load paths/routing will comply with NUREG-0612, Section 5.1.1(1), guidelines. These items are DFSER COL Action Item 9.1.5-1.

The staff stated in the DSER (SECY-91-235) that the information provided in SSAR Section 9.1.5 (also Section 9.1.4, which is cross-referenced in Section 9.1.5),

lacked details with respect to OHLHS compliance with the applicable guidelines of NUREG-0612. The staff's concerns, identified as DSER Open Item 49 (Section 1.8 of the DSER grouped four concerns of DSER Section 9.1.5 as Open Item 49.) in this regard, are as follows:

1. In the DSER (SECY-91-235), the staff noted that GE had not identified all the hoists for the reactor building, refueling platform, and steam tunnel cranes and any other crane, nor the load-handling capacity for all the hoists, including the monorail hoist, equipment hatchway hoist, equipment platform/lower drywell RIP hoist. In the DFSER, the staff further noted that GE had not provided this information, therefore, this part of DSER Open Item 49 was reclassified as Open Item 9.1.5-5 in the DFSER. Upon further review, the staff determined that the issue can be addressed by COL actions. The SSAR, Section 9.1.6.6, states that the COL applicant will provide this information. This is acceptable and resolved DFSER Open Item 9.1.5-5.
2. In the DSER (SECY-91-235), the staff noted that GE had not identified which of the hoists and related lifting devices associated with all heavy load-handling systems meet single-failure-proof criteria and which of these components meet the alternative criteria specified in applicable sections of NUREG-0612. The staff indicated that GE should identify the specific alternative criterion applicable to the chosen heavy load-handling device, or if neither of the above criteria is applicable for some heavy load-handling devices (because they do not affect required safety-related equipment). GE should include justification for such a conclusion for the applicable devices.

As previously noted, SSAR Amendment 20 identified the criteria to be applied to each heavy load-handling system except for those in the control building. With the exception of this area as discussed separately in item (4) below, this information resolved the applicable part of DSER Open Item 49.

3. In the DSER (SECY-91-235), the staff noted that GE did not identify the specific limit and safety devices (e.g., interlocks and limit switches) provided for automatic and manual operation of all the heavy load-handling equipment (within ABWR scope) under both normal and emergency conditions. Further, the staff noted that GE did not submit a failure modes and effects analysis for the OHLHS instrument and control system to demonstrate that the control system will adequately limit the loads or limit the crane load movement, assuming a single failure, without affecting the function of safety-related equipment or causing the release of radioactivity.

Section 9.1.5.5 of SSAR Amendment 20 states that limit switches and safety interlocks are provided to prevent transporting heavy loads over spent fuel. However, the amendment does not include the level of detail requested by the staff in the DSER (SECY-91-235) and does not specify this information as the action of the COL applicant. In the DFSER, the staff stated that GE should either submit this information or designate this as the action of the COL applicant. The staff reclassified this part of DSER Open Item 49 as DFSER Open Item 9.1.5-6. SSAR Section 9.1.6.6 states that the COL applicant will submit this information. This is acceptable and DFSER Open Item 9.1.5-6 is resolved.

4. The staff stated in the DSER (SECY-91-235) that GE had not indicated the heavy load-handling operations which may involve areas other than the reactor building which are within ABWR scope (e.g., control building). Specifically, GE had not identified the heavy load-handling equipment, its handling capacity, the load required to be handled, when such operations have to be performed, and how safety-related equipment is protected during such operations.

In the DFSER, the staff stated that GE did not submit the information requested in the DSER (SECY-91-235) and did not state guidelines for load-lifting systems outside the scope of the ABWR (e.g., the reactor service water (RSW) pump house). The staff further noted that this information should include the identification of the load-handling systems, the system characteristics, and the ability to meet the guidelines of NUREG-0612. The staff reclassified as DFSER Open Item 9.1.5-7 the submission of the previously requested information and the identification of the requirements for load lifting systems outside the ABWR scope. Upon further evaluation, the staff determined that the issue can be resolved by COL actions. SSAR Section 9.1.6.6 clarifies that the COL applicant will submit design and operational details for all load-handling devices both in and out of the design scope. The staff concludes that this clarification will ensure that the design and operational details associated with the movement of heavy loads will be provided. This is acceptable and resolved DFSER Open Item 9.1.5-7.

The staff finds that the OHLHS design and commitments meet the guidelines of NUREG-0612 and requirements of GDC 61 for the safe handling and storage of fuel.

In the DFSER, the staff requested GE to submit adequate design description and the ITAAC for the OHLHS. This was DFSER Open Item 9.1.5-8. GE submitted a revised set of design description and ITAAC. The adequacy and

acceptability of the design description and the ITAAC are evaluated in Section 14.3 of this report. On the basis of the above, DFSER Open Item 9.1.5-8 is resolved.

The staff concludes that the design of the OHLHS for the ABWR conforms with the requirements of GDC 2, 4, 5, and 61 for protection against natural phenomena, protection of safety-related equipment from the effects of internal missiles, the sharing of important systems, and the safe handling and storage of the fuel. The design also conforms with the guidelines of Positions C.1, C.3, C.5, and C.6 of RG 1.13 and Positions C.1 and C.2 of RG 1.29. The staff further concludes that the OHLHS equipment within the ABWR scope conforms with the guidelines of NUREG-0612. However, since a major portion of NUREG-0612 is outside the scope of the ABWR standard design (the part which deals with crane operation, operator training, operating and maintenance procedures, and physical marking of safe load paths), the staff stated in the DFSER that the COL applicant will provide the specifics of compliance and final implementation of NUREG-0612 guidelines on a plant-specific basis. This was DFSER COL Action Item 9.1.5-2. SSAR Section 9.1.6 states that the COL applicant will submit this information.

The OHLHS includes all components and equipment used to handle all heavy loads at the plant site over the lifetime of the facility. Based on the review of the SSAR design criteria and design bases for the OHLHS, the staff concludes that the design of the OHLHS, as discussed in detail above, is in conformance with the guidelines of SRP Section 9.1.5 and the Commission's regulations as set forth in GDC 2, 4, 5, and 61, and is acceptable.

9.2 Water Systems

For some systems listed in the SRP, the functions in the ABWR design will be performed by one or more different systems. For example, the functions of the closed cooling water system also will be performed by the RCW system, the heating, ventilation, and air conditioning normal cooling water (HNCW) system, the HVAC emergency cooling water (HECW) system, and the turbine building cooling water (TCW) system.

9.2.1 Station Service Water System

See Section 9.2.15, "Reactor Service Water."

9.2.2 Reactor Auxiliary Cooling Water System

See Sections 9.2.11 "Reactor Building Cooling Water System," and 9.2.12, "HVAC Normal Cooling Water System" of this report.

Auxiliary Systems

9.2.3 Demineralized Water Makeup System

See Sections 9.2.8, "Makeup Water System (Preparation)," 9.2.9, "Makeup Water System (Condensate)," and 9.2.10 "Makeup Water System (Purified)" of this report.

9.2.4 Potable and Sanitary Water System

The staff reviewed the design requirements for the potable and sanitary water (PSW) system in accordance with SRP Section 9.2.4. Staff's acceptance of the design is based on meeting the requirements of GDC 60 as it relates to preventing the release of liquid effluents containing radioactive material into the PSW system. Compliance is met if there are no interconnections between the PSW system and any potentially contaminated systems and if the PSW system is protected by an air gap, if necessary.

GE stated that only the portion of the PSW system located in the buildings of the certified standard plant is within the design scope, while the remainder of the system is not. The out-of-scope portion of the system will be designed by the applicant referencing the ABWR design. The DFSEER noted that the staff's review was in progress. This was DFSEER Open Item 9.2.4-1. SSAR Section 9.2.4 includes a conceptual design and interface requirements for the out-of-scope portion of the system. The staff reviewed the conceptual design and interface requirements and concluded that sufficient guidance has been provided to allow an applicant referencing the ABWR to design the out-of-scope portion to meet all applicable regulatory requirements. This is acceptable and resolved DFSEER Open Item 9.2.4-1.

The PSW system is a non-safety-related system designed to provide a minimum of approximately 45 m³/hr (200 gpm) of potable water during peak demand periods. The system is composed of a potable water subsystem, a sanitary drainage subsystem, and a sewage treatment subsystem.

Water is supplied to the potable water subsystem from the makeup water system (preparation) (MWP) to a potable water storage tank. The water is chemically treated, pressurized, heated, and distributed throughout the plant. According to SSAR Table 3.2-1, the system serves all areas of the plant within the ABWR scope except the primary containment and the MST.

Liquid wastes (including those from the nonradioactive drain system) are collected in the sanitary drainage subsystem and sent to the sewage treatment subsystem. The sewage treatment subsystem uses the activated sludge biological treatment process. The subsystem contains a comminutor, aeration tanks, aerobic digesters, air blowers,

clarifiers, a froth spray pump, a hypochlorite pump, and associated equipment. This subsystem can be operated in the extended aeration mode or the contact stabilization mode (used during high demand periods (e.g., refueling outages) when additional personnel are on site).

The PSW system does not include any connections to systems which may contain radioactive material. Where necessary, additional protection is provided through the use of air gaps.

The system contains adequate controls, instrumentation, and alarms to ensure adequate operation during normal conditions and to alert operators to abnormal conditions. Drainage piping will be hydrostatically tested.

In the DFSEER, the staff requested GE to submit adequate design description, ITAAC, and interface requirements for the PSW system. This was DFSEER Open Item 9.2.4-2. Subsequently, GE provided a revised set of design description, ITAAC, and interface requirements. The adequacy and acceptability of the design description, the ITAAC, and the interface requirements are evaluated in Section 14.3 of this report. On the basis of the above, DFSEER Open Item 9.2.4-2 is resolved.

The PSW system includes all components and piping from the supply connection to the municipal or other water source to all points of discharge to sewage facilities or other plant systems. Based on the review above, the staff determined that adequate design provisions have been made to prevent the inadvertent contamination of the system with radioactive material. The staff concludes that the design of the PSW system meets the guidelines of SRP Section 9.2.4, and since there are no interconnections between the PSW system and any contamination systems it also meets the requirements of GDC 60, and is acceptable.

9.2.5 Ultimate Heat Sink

The staff reviewed the design requirements for the ultimate heat sink (UHS) in accordance with SRP Section 9.2.5. Staff acceptance of a UHS design is based on meeting GDC 2 as it relates to structures housing the system and the system itself being capable of withstanding the effects of natural phenomena; GDC 5 as it relates to the capability of shared SSCs to perform required safety functions; GDC 44 as it relates to the capability to transfer heat loads from safety-related SSCs to the heat sink under normal and accident conditions, providing suitable redundancy of components to ensure adequate safety function given a single active component failure, and the capability to isolate parts of the system so that the safety function is not compromised; and GDC 45 and 46 as they relate to

inservice inspection and operational functional testing, respectively, of safety-related systems and components.

The SSAR states that the design of the UHS is outside the scope of the ABWR design. The SSAR includes a conceptual design and interface requirements, as required by 10 CFR Part 52, to allow an applicant referencing the ABWR design to provide a plant-specific UHS design that is capable of dissipating reactor decay heat and essential cooling loads after a normal reactor shutdown or a shutdown after an accident, including a loss-of-coolant accident (LOCA). The UHS must be designed to accept the heat loads of the RSW system (Section 9.2.15 of this report), which in turn accepts the heat loads of the RCW system (Section 9.2.11 of this report) under both normal and accident conditions.

The conceptual design for the UHS consists of a seismic Category I spray pond from which the RSW system will receive cooling water. The spray pond will be excavated below grade and contain adequate water volume to supply cooling for 30 days under design-basis conditions. Six spray networks, three functioning during normal operation, will cool the RSW return water. The spray nozzles may be bypassed during cold weather conditions allowing RSW return water to be returned directly to the pond. RSW pumps will be located in the spray pond pump structure, each pump in its own bay. The pond will also have a seismic Category I overflow weir to accommodate normal level fluctuations. The spray pond will receive makeup from a power cycle heat sink makeup line.

The structures and components of the UHS will be designed to seismic Category I requirements and will be designed to withstand the effects of natural phenomena such as floods, earthquakes, and tornados. The system design should ensure that the UHS can perform its safety function given the occurrence of any of the following: (1) the most severe natural phenomena appropriate with site conditions, (2) site-related events that have historically occurred, (3) reasonable combinations of natural phenomena and site-related events, and (4) a single failure of man-made structures. These interface requirements will allow an applicant to design a UHS that meets the requirements of GDC 2 regarding protection from natural phenomena and RG 1.27, "Ultimate Heat Sink for Nuclear Power Plants," Revision 2 and RG 1.29, "Seismic Design Classification," Revision 3.

The UHS design will ensure that safety-related portions of the system are protected from spraying, steam impingement, pipe whip, jet forces, missiles, fire, internal flooding, and the effects of failure of non seismic Category I equipment. Thus, the interface requirements meet the requirements of GDC 4 for protection of systems

from the environmental and dynamic effects associated with equipment failures.

The requirements of GDC 5 for the sharing of SSCs do not apply because the ABWR is designed as a single-unit facility. An application for a multi-unit facility will require review of the design for compliance with GDC 5.

The UHS must have the capability to transfer heat loads from safety-related structures and systems during normal and accident conditions, and suitable redundancy so that it will function given a single-failure coincident with a LOOP. It must also be capable of isolating portions of the system in such a way as to not interfere with the system's safety function. The UHS will provide cooling capability for 30 days.

The requirements of RG 1.72, "Spray Pond Piping Made from Fiberglass-Reinforced Thermosetting Resin," Revision 2, apply to the design of a spray pond as an UHS. The staff established Open Item 50 in the DSER because these requirements were not referenced in the SSAR submitted before the staff completed the DSER (SECY-91-235). SSAR Section 9.2.5.8 states that the COL applicant will submit information to show that all applicable requirements of RG 1.72 are met. This is acceptable and resolved Open Item 50.

The SSAR requires the COL applicant to prepare a preoperational test program in accordance with the requirements of Chapter 14 and shall perform inspections and tests during normal operations. Based on these commitments, the staff believes that the applicant referencing the ABWR design can perform inspections and tests which meet the requirements of GDC 45 and 46, respectively.

In the DFSE, the staff requested GE to include interface requirements for the UHS system in the Tier 1 information. This was DFSE Open Item 9.2.5-1. The adequacy and acceptability of the Tier 1 information is evaluated in Section 14.3 of this report. On the basis of the above, DFSE Open Item 9.2.5-1 is resolved.

The staff found the conceptual design and interface requirements for the UHS to be acceptable. They include adequate guidelines to ensure that the plant-specific design can meet the requirements of GDC 2, 5, and 44 with respect to protection against natural phenomena, sharing of SSCs and heat transfer, redundancy, isolation capabilities, and ability to transfer heat loads. The system must be designed by the COL applicants to allow periodic inspections and tests and must, therefore, meet the requirements of GDC 45 and 46 with respect to inspection and testing requirements for cooling water systems. The interface

Auxiliary Systems

requirement in the ABWR SSAR should enable an applicant referencing the ABWR to design an acceptable UHS.

9.2.6 Condensate Storage Facility

See Section 9.2.9, "Makeup Water System (Condensate)," of this report.

9.2.7 Chilled Water Systems

See Sections 9.2.12, "HVAC Normal Cooling Water System," and 9.2.13, "HVAC Emergency Cooling Water System," of this report.

9.2.8 Makeup Water System (Preparation)

The staff reviewed the requirements for the makeup water system (preparation) (MWP) in accordance with SRP Section 9.2.3. Staff acceptance of the MWP system design is based on meeting GDC 2 as it relates to safety-related portions of the system being capable of withstanding the effects of natural phenomena, GDC 5 as it relates to the capability of shared SSCs to perform required safety function, and position C.2 of RG 1.29, Revision 3 (September 1978) relative to protection of safety-related SSCs from failure of the system following an SSE.

The MWP system is not within the certified design scope of the ABWR. As required by 10 CFR Part 52, the SSAR provides a conceptual design and interface requirements for the MWP system. This design information and commitments provide sufficient detail to ensure that the system can be designed and built to meet the applicable regulatory requirements.

The MWP system is a non-safety-related system that supplies water for the makeup water system (purified) distribution system (MUWP) and the portable and sanitary water system (PSW). The MWP system design includes a requirement for two system divisions, each capable of producing approximately 45 m³/hr (200 gpm) of demineralized water and each with a storage capacity of at least 760 m³ (200,000 gallons).

The system will consist of both a permanently installed water treatment system and a mobile water treatment system. The permanently installed system will consist of a well, filters, reverse osmosis modules, demineralizers, storage tanks, and pumps to treat and store well water for use as demineralized water. The mobile water treatment system will be used before the permanent system is operable and if needed to supplement the permanent system. Therefore, the system does not require a seismic Category I makeup source.

The MWP system will be located in a building that does not contain any safety-related components, systems, or structures. Any failure of the system (including failures that could cause flooding) will not result in the failure of any safety-related SSCs and consequently will not adversely affect any safety-related function of the ABWR design. Therefore, the staff concludes that the conceptual design meets the requirements of GDC 2 with regard to protection of safety-related equipment from natural phenomena, and Position C.2 of RG 1.29, Revision 3 (September 1978).

The requirements of GDC 5 for the sharing of SSCs do not apply because the ABWR is designed as a single-unit facility. An application for a multi-unit facility will require review of the design for compliance with GDC 5.

In the DFSE, the staff requested GE to submit adequate design description, ITAAC, and interface requirements for the MWP system. This was DFSE Open Item 9.2.8-1. GE submitted a revised set of design description, ITAAC and interface requirements. The adequacy and acceptability of the design description, ITAAC, and interface requirements are evaluated in Section 14.3 of this report. On the basis of the above, DFSE Open Item 9.2.8-1 is resolved.

Based on the SSAR commitments discussed above, the staff concludes that GE has described an acceptable conceptual design and interface requirements sufficient to ensure that an applicant referencing the ABWR can design a MWP system that will meet the requirements of Position C.2 of RG 1.29, Revision 3 (September 1978); GDC 2 with regard to the ability of the non-safety-related portions of the system to withstand the effects of natural phenomena without affecting safety-related systems; and GDC 5 with regard to the sharing of SSCs important to safety. The interface requirements in the SSAR will enable an applicant referencing the ABWR to design an acceptable MWP system in compliance with the guidelines of SRP 9.2.3. The conceptual design is acceptable.

9.2.9 Makeup Water System (Condensate)

The staff reviewed the MUWC system (the condensate storage and transfer system) in accordance with SRP Section 9.2.6. Staff acceptance of the MUWC system design is based on meeting the requirements of GDC 2 as it relates to the system being able to withstand the effects of natural phenomena; GDC 5 as it relates to the capability of shared systems and components to perform required safety functions; GDC 44 as it relates to redundancy and isolability of components as well as the capability to provide makeup to safety-related systems; and GDC 45 and 46 as they relate to inservice inspection and testing,

respectively, of safety-related systems and components. The design of the entire MUWC system is within the scope of the ABWR certified design.

The MUWC system will supply condensate quality water and will include a piping distribution system from the source to the components that require this water during normal and emergency operations. MUWC system water will be stored in the condensate storage tank (CST) with a capacity of at least 2,100,000 L (555,000 gallons). The CST is located outdoors adjacent to the turbine building. The CST will reserve approximately 570,000 L (150,000 gallons) of this capacity to remove decay heat for up to 8 hours after a station blackout (SBO). Water will be supplied to the CST from the MUWP system. Level-sensing instrumentation and transmitters will automatically switch over the HPCF and reactor core isolation cooling (RCIC) pumps from the preferred CST to the safety-related suppression pool when the CST water level is low. The tank will also supply water for the control rod drive (CRD) supply pump (the preferred water source being the condensate treatment system) and the SPCU pump, which will be used for fuel pool makeup, when required. The MUWC system will normally supply water through three system transfer pumps for charging, flushing, pump sealing, surveillance testing, room decontamination, and makeup, as appropriate, for systems including RHR, HPCF, RCIC, fuel pool skimmer surge tanks, and the main condenser hotwell.

The MUWC system is not safety-related, except as noted below, because it will not affect the integrity of the reactor coolant system pressure boundary, prevent achieving and maintaining safe shutdown, or affect the capability to prevent or mitigate the consequences of accidents, which could result in unacceptable offsite radiological exposures. Therefore, GDC 44, 45, and 46 do not apply to the non-safety-related portions of the system.

The safety-related portions of the system discussed below meet the redundancy, inservice inspection, and inservice testing requirements for safety-related equipment and therefore meet the requirements of GDC 44, 45, and 46.

As stated in SSAR Table 3.2-1, for the MUWC, RCIC, and HPCF systems, certain parts of MUWC system piping, including supports and valves, will be designed to seismic Category I and QG B standards and will be located in seismic Category I, flood-protected and tornado-missile-protected structures. The safety-related portions include those forming part of the containment boundary and those system piping portions that interface with the safety-related RCIC and HPCF systems, up to and including the isolation/suction valves for the systems from the MUWC. The non-safety-related portions of the system

that could affect any SSCs important to safety if they fail during a seismic event are designed to ensure their integrity under seismic loading conditions resulting from an SSE. The level instrumentation in the MUWC system that facilitate the automatic switchover of the HPCF and RCIC pumps suction from the CST to the suppression pool and their power supplies are safety-related. The SPCU pumps will be switched over manually. The staff concludes that the safety-related and non-safety-related portions of this system meet Positions C.1 and C.2, respectively, of RG 1.29.

The staff stated in the DSER (SECY-91-235) that GE did not submit an analysis for flooding that could result from a possible failure of the non-safety-related portion of the MUWC system, including the CST, and how safety-related SSCs are protected from such flooding. This was DSER Open Item 51. SSAR Section 3.4.1.1.1 reports the results of a flood analysis and describes the safety features that will protect safety-related SSCs from external floods (including the failure of the CST) as discussed in Section 3.4.1 of this report. The instrumentation used to initiate the automatic switchover of HPCF and RCIC suction from the CST to the suppression pool is safety-related and is housed in a safety-grade standpipe in the reactor building, a seismic Category I structure designed to withstand tornadic winds and missiles, flooding, hurricanes, and an SSE. On the basis of this information, the staff concludes that the system meets the requirements of GDC 2 as it relates to the protection of safety-related portions of the system from the effects of natural phenomena. Therefore, DSER Open Item 51 is resolved.

The ABWR is designed as a single-unit facility. Therefore, the requirements of GDC 5 regarding the sharing of SSCs are not applicable. An application for a multi-unit facility will require review of the design for compliance with GDC 5.

Normal alignment for removal of decay heat is with the CST. Water for RCIC operation is taken from either the CST or the suppression pool as described in the emergency procedure guidelines of SSAR Appendix 18A. The volume of water in these two sources is sufficient to permit core cooling during SBO for a duration of 8 hours. The switchover from the CST to the suppression pool (or the reverse) is performed using station dc power and is not dependent upon either offsite ac power and systems or onsite emergency power systems. Therefore, the MUWC system complies with the applicable guidance of RG 1.155, "Station Blackout," Revision 0 (August 1988).

MUWC is demonstrated to be operable by normal system operation. Those portions of the system normally closed

Auxiliary Systems

to flow can be tested to ensure system operability and integrity.

Based on this information, the staff concludes that MUWC meets the applicable requirements of GDC 44, 45, and 46.

In the DFSER, the staff requested GE to provide adequate design description and the ITAAC relating to the MUWC system. This was DFSER Open Item 9.2.9-1. Subsequently, GE provided a revised set of design description and ITAAC. The adequacy and acceptability of the design description and the ITAAC are evaluated in Section 14.3 of this report. On the basis of the above, DFSER Open Item 9.2.9-1 is resolved.

The MUWC system includes all components and piping associated with the system to the points of connection with other systems. The staff determined that the SSAR design criteria and bases for the MUWC and the requirements for a sufficient water supply to safety-related systems during normal and emergency conditions are acceptable.

The staff concludes, as discussed above, that the MUWC system design complies with Positions C.1 and C.2 of RG 1.29, GDC 2, 5, 44, 45, 46, and, therefore, with the applicable acceptance criteria of SRP Section 9.2.6 and is acceptable.

9.2.10 Makeup Water System (Purified) Distribution System

The staff reviewed the MUWP system in accordance with SRP Section 9.2.3. The design is acceptable if it complies with the requirements of GDC 2 as it relates to protection of the safety-related portions of the system from natural phenomena, and GDC 5 as it relates to the capability of shared SSCs to perform required safety functions.

The design of the MUWP system is fully within the scope of the ABWR design. The staff evaluated the ABWR capability to prepare, store, and transport demineralized water to this system as described in Section 9.2.8 of this report. The MUWP system is not safety-related, except as noted below.

The MUWP system will supply demineralized makeup quality water and will include a piping distribution system from the source to the components that require this water. The MUWP system will receive makeup water from the MWP system and will normally supply demineralized water for flushing, sealing, surveillance testing, area decontamination, sampling, and makeup as appropriate. The MUWP system will supply makeup water to systems such as the MUWC system, the RCW system, the TCW system, the diesel generator cooling water (DGCW)

system, the liquid waste management system (LWMS), the standby liquid control system (SLCS), and other plant auxiliary systems. Protection from flooding for safety-related SSCs is discussed in Section 3.4.1 of this report.

The MUWP system is not safety-related, except as noted below, because it will not affect the capability of the reactor coolant system pressure boundary, the capability to achieve and maintain safe shutdown, or the capability to prevent or mitigate the consequences of accidents that could result in unacceptable offsite radiological exposures. The MUWP system enters the primary containment through one penetration. The system piping through this penetration has a locked-closed manual valve outside the containment and a check valve inside the containment. The portions of the system penetrating the containment (including these two valves) are designed to seismic Category I, QG B requirements in accordance with Position C.1 of RG 1.29.

Although it is not safety-related, this system is designed to prevent any radioactive contamination of the purified water. SSAR Table 9.2-2a presents chemistry requirements for the purified makeup water.

In the DSER (SECY-91-235), the staff stated, as part of Open Item 52, that it was not clear whether the non-safety-related portions of the system, which upon their failure during a seismic event can adversely impact SSCs important to safety, will be designed to ensure their integrity under seismic loading resulting from an SSE. SSAR Section 9.2.10.1 states that the portions of the MUWP whose failure can impact safety-related SSCs will be designed to ensure their integrity under seismic loading conditions resulting from an SSE. This is acceptable. Therefore, the design of the MUWP system complies with Positions C.1 and C.2 of RG 1.29 for the safety-related and non-safety-related portions of the system, and GDC 2 for protection against natural phenomena. This resolved the applicable part of DSER Open Item 52.

In the DSER (SECY-91-235), the staff also stated, as part of Open Item 52, that it was not clear whether the portions of the MUWP system in buildings other than the reactor building (e.g., turbine building) and the transport of the demineralized water to these buildings were within GE's scope or the scope of the applicant referencing the ABWR design. SSAR Section 9.2.10.1 states that the interfaces between the MUWP system and safety-related systems are located in the control building or reactor building which are seismic Category I, tornado-missile resistant and flood-protected structures. Therefore, the MUWP system complies with Positions C.1 and C.2 of RG 1.29 and with GDC 2 and, therefore, with the applicable acceptance

criteria of SRP Section 9.2.3. This is acceptable and the applicable part of DSER Open Item 52 is resolved.

In the DFSER, the staff stated that the applicant referencing the ABWR design should supply the following: testing capability for air-operated valves; adequate pump net positive suction head (NPSH), purified water storage tank overflow/drainage diversion to the radwaste system; material corrosion resistance; adequate distribution piping, valves, instrumentation, and controls; control room instrumentation that indicates the water level in the purified water storage tank; outdoor piping freeze protection; and adequate diking and other means to control spill and leakage from the demineralized water storage tank that will be located outdoors. This was DFSER COL Action Item 9.2.10-1. Upon further review, the staff has determined that the listing of this information as a COL action item is not needed in the SSAR because it will be provided as a normal part of the licensing process. Therefore, COL action item 9.2.10-1 is appropriately deleted from the SSAR.

The requirements of GDC 5 for the sharing of SSCs do not apply because the ABWR is designed as a single-unit facility. An application for a multi-unit facility will require review of the design for compliance with GDC 5.

In the DFSER, the staff requested GE to supply adequate design description and ITAAC for the MUWP system. This was incorrectly identified in the DFSER as Open Item 9.2.10-2. This designation was later corrected to DFSER Open Item 9.2.10-1. Subsequently, GE provided a revised set of design description and ITAAC. The adequacy and acceptability of the design description and the ITAAC are evaluated in Section 14.3 of this report. On the basis of the above, DFSER Open Item 9.2.10-1 is resolved.

Based on information in the SSAR and that discussed above, the staff concludes that the MUWP system complies with Positions C.1 and C.2 of RG 1.29, GDC 2, and GDC 5, and therefore, with the applicable acceptance criteria of SRP Section 9.2.3 and is acceptable.

9.2.11 Reactor Building Cooling Water System

The staff reviewed the RCW system in accordance with SRP Section 9.2.2. Staff acceptance of the design of the RCW system is based on meeting the requirements of GDC 2 as it relates to the system withstanding the effects of natural phenomena; GDC 4 as it relates to the system withstanding the effects of failed equipment and piping during both normal and accident conditions; GDC 5 as it relates to the capability of shared SSCs to perform required safety functions; GDC 44 as it relates to the

capability to transfer heat loads from safety-related SSCs to the heat sink under normal and accident conditions, providing suitable redundancy for components given a single active component failure, and the capability to isolate part of the system so that the safety function is not compromised; and GDC 45 and 46 as they relate to permitting inservice inspection and testing, respectively, for safety-related equipment.

The requirements of GDC 5 for the sharing of SSCs do not apply because the ABWR is designed as a single-unit facility. An application for a multi-unit facility will require review of the design for compliance with GDC 5.

The function of the RCW system is to remove heat from plant auxiliaries (some of which are required for safe shutdown) during normal operation and after a LOCA. The RCW system is required to operate, with and without preferred ac power available, at normal power, reactor shutdown, hot standby, and after a postulated LOCA has occurred. The RCW system is a closed cooling water system that provides cooling water to the following essential systems and components: RHR and fuel pool cooling heat exchangers; mechanical seals and motor bearings for RHR and HPCF pumps; air conditioning units (ACUs) for pump rooms (RHR, HPCF, FPC, and RCIC) and system rooms (SGTS, containment atmospheric monitoring system (CAMS), and flammability control system (FCS)); jacket water coolers and filtered water and lubricating oil coolers for diesel generators; and HECW system refrigerators. The RCW system supplies cooling water to the non-essential RIP pump motor coolers and motor generator sets, drywell coolers, reactor water cleanup (CUW) pump coolers, instrument air (IA) and service air (SA) system coolers, CUW non-regenerative heat exchangers, CRD pump oil coolers, and other nonessential auxiliary components in the reactor, turbine, and radwaste buildings (e.g., radwaste components, condenser offgas, and reactor building and turbine building sampling coolers).

The RCW system supplies cooling water, which absorbs heat from the plant auxiliaries it serves, and rejects the heat through the RCW system heat exchangers to the reactor service water system. The RSW system, in turn, rejects the heat to an UHS that will be designed by the COL applicants. The GE scope of the RCW system includes all the piping, valves, pumps, heat exchangers, instrumentation, and controls from the RCW system heat exchangers to their loads in the reactor, turbine, and radwaste buildings. GE specified the total heat removal rate, total flow rate, temperature drop, and pressure drop at the RCW system heat exchangers for all modes of operation identified above. These parameters provide the heat removal requirements for the referencing applicant to

Auxiliary Systems

design the plant-specific UHS system that would be connected to the RSW system.

The RCW system consists of three mechanically and electrically independent divisions, each consisting of its own separate piping (including supply and return headers), two pumps, three heat exchangers, valves, and instrumentation. Each division of the RCW system is powered by a different division of the engineered safety features (ESF) power system. Each division of the RCW system supplies cooling water to the auxiliaries of a separate emergency diesel generator, RHR heat exchanger, RHR pump room ACU, RHR pump motor and seal coolers and HECW refrigerators. Other safety loads and non-essential cooling loads are distributed among the three divisions; two divisions share the loads for systems with redundant components (e.g., HPCF and SGTS). Each division has one isolable train for nonessential loads.

Each division of the RCW system is equipped with a surge tank (shared with the HECW system) which the SSAR states is designed to accommodate 30 days of system design leakage without makeup water (GE response, dated March 7, 1989). Also, the system is designed to detect system leakage by associated level monitors, to provide adequate pressure for pump suction, and to allow for changes in system water volume without significant pressure variations. The system is initially filled with demineralized water from the MUWP system. Each division is further equipped with a chemical addition tank to add chemicals to the RCW system to protect it from corrosion or organic fouling.

The system is protected from water-hammer by high point vents in isolable portions of the system and operational procedures requiring filling and venting of any sections of the system before operation. During a LOCA, nonessential RCW system cooling loads are automatically isolated by the closure of valves except for system cooling loads to IA and SA system coolers, CRD pump oil coolers, and CUW pump coolers; these are isolated by the operator, if necessary. Level switches for the surge tank enable the automatic isolation of nonessential cooling loads in the event of significant system leakage results from piping failures in the non-safety-related portions of the system. One valve on each supply and discharge line, with suitable power and controls from applicable divisional sources, ensures isolation if a single active component fails. GE described the methods for determining whether the system leakage occurs in the nonessential portion of the system as indicated by a falling surge tank level. Radiation monitors located downstream of the RCW system pumps and heat exchangers indicate that radiation has leaked into that division. The ABWR design includes remote manual isolation capability for any division. The

two remaining operable divisions will be sufficient to meet the total essential cooling load.

The RCW system consists of safety-related and non-safety-related portions. Portions of the system piping (including valves forming part of the primary containment boundary and other safety-related portions of the system piping up to and including the isolation valves that isolate the system from its non-safety-related portions) are designed to the requirements of seismic Category I, QG B or C, and 10 CFR Part 50, Appendix B. The safety-related portions include the RCW system pumps, heat exchangers, surge tanks, and the division isolation valves. Instrumentation and controls performing safety-related functions (e.g., surge tank level switches) are located in the safety-related portions of the system. Electric modules (e.g., sensors, power supplies, signal processors) and cables performing safety-related functions are all designed to seismic Category I and QG B requirements. Non-safety-related portions of the system that can adversely impact safety-related SSCs if they fail during a seismic event, are designed to ensure their integrity under seismic loads resulting from an SSE. The safety-related portions are located in seismic Category I structures designed to protect against flood and tornado missiles. In the DSER (SECY-91-235), staff requested GE to submit confirmatory information that the safety-related electric modules and safety-related cables are located in seismic Category I, flood-protected, and tornado-protected structures. The SSAR incorporates this information. This is acceptable and resolved the unnumbered DSER (SECY-91-235) confirmatory item. Therefore, the staff concludes that the design of the RCW system complies with GDC 2 with respect to protection from natural phenomena, and meets Positions C.1 and C.2 of RG 1.29 with respect to its seismic requirements for the safety-related and non-safety-related portions.

The SSAR states that both the mechanical equipment and piping and electrical equipment, including instrumentation and controls of the redundant divisions of the RCW system, are sufficiently separated and protected to ensure availability of the needed equipment to shut the reactor down in the event of any of the following occurrences: pipe rupture or equipment-failure-induced flooding, spraying or steam release; pipe whip and jet forces from a postulated nearby high-energy pipeline break; missiles from equipment failure; fire; non seismic Category I equipment failure; or a single active component failure in the system. The amendments included the results of a failure analysis of the RCW system to demonstrate that a single active or applicable passive component failure will not compromise the ability of the RCW system to transfer heat loads from safety-related components to the RSW during all modes of operation. GE submitted design char-

acteristics for RCW system components (e.g., pump design flow rate; heat exchanger heat removal capacity) to show that the system will transfer the expected heat loads to the RSW system under all operating conditions.

The staff stated, as Open Item 53 in DSER (SECY-91-235), the following three concerns:

1. The heat removal design capacity of the RCW system heat exchanger may be inadequate for the heat load required to be removed during suppression pool cooling, when the pool temperature reaches 97 °C (207 °F) after a LOCA. The staff noted that a reactor shutdown at 4 hours after a blowdown to the main condenser may be the bounding case, which may require a greater heat removal rate and, consequently, a higher design capacity than that stated for the heat exchangers (see GE's response to RAI 440.73).

In the DFSE, the staff reclassified this concern as Open Item 9.2.11-1 because it had not yet been addressed by GE. The staff reviewed the SSAR and found that the RCW heat exchanger capacity design had been modified to ensure that the RCW system will be able to remove the worst anticipated heat loads as stated in SSAR Table 9.2-4. This is acceptable and resolved DFSE Open Item 9.2.11-1.

2. The staff stated in the DSER (SECY-91-235) that the projected heat loads and flow rates for hot standby conditions with a loss of ac power indicate that both RCW pumps and all three heat exchangers in a division are required. The staff also stated that this will also be the case for shutdown at 4 hours.

Data in SSAR Table 9.2-4 clearly indicate that this equipment is required for successful operation of each division for this mode of operation. This is acceptable and resolved the applicable part of DSER (SECY-91-235) Open Item 53.

3. The staff questioned in the DSER (SECY-91-235) whether the loss of an RCW system division during normal operation would result in plant shutdown or operation at reduced power.

In response, the SSAR states that the loss of one division of RCW to the drywell coolers will not affect plant operation and the loss of cooling to the RIP coolers will reduce plant power output but will not result in a reactor trip. This clarification is acceptable and resolved the applicable part of DSER Open Item 53.

All three divisions of the RCW system will have at least one RCW system pump operating. This configuration ensures the immediate availability of the RCW system for plant shutdown in the event of a LOCA. A LOOP concurrent with a LOCA will result in a temporary loss of pumping until the automatically sequenced restart of RCW system pumps from the emergency diesel generator loading sequence. A LOCA will result in the automatic isolation of most non-safety-related RCW system loads, the starting of the second RCW system pump, and the placing of the third heat exchanger in each division in service.

The staff concludes that the safety-related portions of the RCW system comply with the requirements of GDC 4, with respect to protection against the dynamic effects of postulated piping failures and internally- and externally-generated missiles, and with GDC 44 for the provisions of a system to transfer heat from SSCs important to safety to an UHS.

As stated in Section 9.2.5 of this report, GE submitted interface requirements for an applicant to design the UHS. The RCW system water quality requirements are established by the MUWP because this system supplies makeup water for the RCW system as discussed in Section 9.2.10 of this report. All three divisions of the RCW system are designed to allow periodic inservice inspection of all the system components. The inspections consist of structural and leak-tightness visual inspection, inspection of the entire system for operability, and inspection of the system components for operability and performance. Testing will be conducted to simulate as closely as possible the entire operational sequence of the RCW system for reactor shutdown and LOCA. The system design incorporates provisions for accessibility to permit inservice inspection as required. The staff finds that the system complies with GDC 45 and 46 for inspection and testing requirements for cooling water systems.

In the DFSE, the staff requested GE to submit adequate design description and the ITAAC for the RCW system. This was DFSE Open Item 9.2.11-2. Subsequently, GE provided a revised set of design description and ITAAC. The adequacy and acceptability of the design description and the ITAAC are evaluated in Section 14.3 of this report. On the basis of the above, DFSE Open Item 9.2.11-2 is resolved.

The RCW system includes pumps, heat exchangers, valves and piping, surge tanks, makeup piping, and the points of connection or interfaces with other systems. Portions of the RCW system that are necessary for safe shutdown, accident prevention, or accident mitigation are designed to seismic Category I and QG B or C requirements. The staff reviewed the SSAR design criteria, design bases, and

Auxiliary Systems

safety classification for the RCW system against the requirements for supplying adequate cooling water for the safety-related ECCS components and reactor auxiliary equipment for all conditions of plant operation.

Based on the above the staff concludes that the design of the RCW system is acceptable and meets the requirements of GDC 2, 4, 5, 44, 45, and 46 and the guidelines of SRP Section 9.2.2.

9.2.12 HVAC Normal Cooling Water System

The staff reviewed the HNCW system in accordance with SRP Section 9.2.2. Staff acceptance of the HNCW design is based on meeting GDC 2 as it relates to the system withstanding the effects of natural phenomena; GDC 4 as it relates to the system withstanding the effects of failed equipment and piping during both normal and accident conditions; GDC 5 as it relates to the capability of shared SSCs to perform required safety functions; GDC 44 as it relates to the capability to transfer heat loads from safety-related SSCs to the heat sink under normal and accident conditions, providing suitable redundancy for components given a single active component failure, and the capability to isolate part of the system so that the safety function is not compromised; and GDC 45 and 46 as they relate to permitting inservice inspection and testing, respectively, for safety-related equipment.

The entire HNCW system is within the scope of the ABWR. The HNCW system is not safety related except for portions of the system that penetrate the primary containment, the portions of the system that are part of the secondary containment boundary, and the associated isolation valves.

The major components of the HNCW system are five 25-percent-capacity chillers (one standby), each with an HNCW pump (one standby), a surge tank (shared with the corresponding division of the TCW system), and the associated piping, valves, and instrumentation. Cooling water to the chiller-condenser is supplied by the TCW system.

The requirements of GDC 5 for the sharing of SSCs do not apply because the ABWR is designed as a single-unit facility. An application for a multi-unit facility will require review of the design for compliance with GDC 5.

The function of the HNCW system is to provide chilled water to the drywell cooler cooling coils and cooling coils of other non-safety-related air conditioners, primarily in the reactor, control, radwaste, and service buildings. The HNCW system is not safety-related because it is not

required to ensure (1) integrity of the RCS pressure boundary, (2) capability to achieve and maintain safe shutdown, or (3) the ability to prevent or mitigate offsite radiological exposures during accidents. Therefore, GDC 44, 45, and 46, identified as acceptance criteria in SRP Section 9.2.2 do not apply to the non-safety-related portion of the HNCW system. The HNCW system joins the primary containment through two penetrations: one for the supply line and the other for the return line. The supply line penetration has one motor-operated isolation valve outside the containment and a check (isolation) valve inside the containment. The return line penetration has two motor-operated isolation valves, one inside and one outside the containment. Isolation valves and piping for the primary containment penetrations are safety-related and are designed to seismic Category I, QG B, and 10 CFR Part 50, Appendix B, standards. Piping for penetrations for secondary containment is designed to seismic Category I and 10 CFR Part 50, Appendix B, standards. Based on this information, the staff concludes that the safety-related portions of the HNCW system meet the requirements of GDC 44 regarding the provision for reliable systems for transferring heat loads to a heat sink.

The rest of the HNCW system is not safety-related, as stated above, and is designed to non seismic Category I standards. However, the non-safety-related portions of the system whose failure during a seismic event could affect any structure, system, or component important to safety, are designed to ensure their integrity under seismic loads resulting from an SSE. In the DSER (SECY-91-325), the staff stated that, subject to GE's confirmation that the safety-related portions include the isolation valves for the primary containment penetrations, the design of the HNCW system meets Positions C.1 and C.2 of RG 1.29, as addressed by the SRP Section 9.2.2 acceptance criterion with respect to the seismic requirements for the safety-related and non-safety-related portions of the system. This was DSER Confirmatory Item 6 and DFSER Confirmatory Item 9.2.12-1. SSAR Section 9.2.12.3 clarifies the safety classification of the containment penetrations, which is acceptable. This resolved DSER Confirmatory Item 6 and DFSER Confirmatory Item 9.2.12-1.

By virtue of their location in seismic Category I, tornado-missile-protected and flood-protected structures, the safety-related portions of the system are protected against damage from adverse natural phenomena. Further, as concluded in Section 3.4.1 of this report, all safety-related systems are protected against flooding that may result from system failure. Therefore, the system design complies with GDC 2 with respect to protection of its safety-related portions against natural phenomena and protection of other safety-related systems against the conse-

quences of failure of the non seismic portions of the system, as required by the SRP Section 9.2.2 acceptance criterion.

The major components of the HNCW system are located in the turbine building. Therefore, failure of any of these components will not adversely effect safety-related SSCs. Safety-related portions of the system are located such that the adverse consequences of a pipe or other component failure will not prevent the safety-related portions of the system from performing their safety function. Thus the system design meets the requirements of GDC 4 as it relates to the system's capability to withstand the effects of adverse environmental and dynamic effects.

This system is designed to allow periodic testing and inspection of major components. Appropriate American Society of Heating, Refrigeration, and Air Conditioning (ASHRAE), American Society of Mechanical Engineers (ASME), Tank Equipment Manufacturers Association (TEMA), and Hydraulic Institute (HI) standards are used for all tests. Based on this inspection and test information, the staff concludes that the HNCW systems meets the inspection and testing requirements of GDC 45 and 46, respectively.

Makeup water to the system is supplied by the TCW system surge tank which, in turn, receives water from the MUWP system. The MUWP system and the TCW systems are evaluated in Sections 9.2.10 and 9.2.14 of this report, respectively. The SSAR states design characteristics for the system (e.g., cooling capacity of the chillers, pump design flow rate, chilled water supply temperature) and the heat loads required to be removed from the components served by the system. These characteristics indicate that the system is capable of meeting the cooling water needs of the components it serves during normal plant operation and refueling shutdown. The chiller units are controlled individually by remote manual switches. The containment isolation valves for the system close automatically on a LOCA signal. These valves can also be operated manually by remote means. In the DSER (SECY-91-235), the staff stated, as Open Item 54, a discrepancy regarding the number of HNCW pumps and chillers. The SSAR piping and instrumentation diagram (P&ID) correctly shows five HNCW pumps and associated chillers. This resolved DSER Open Item 54.

In the DFSER, the staff requested GE to provide adequate design description and the ITAAC relating to the HNCW system. This was DFSER Open Item 9.2.12-1. Subsequently, GE provided a revised set of design description and ITAAC. The adequacy and acceptability of the design description and the ITAAC are evaluated in

Section 14.3 of this report. On the basis of the above, DFSER Open Item 9.2.12-1 is resolved.

The HNCW system includes pumps, chillers, valves and piping, surge tanks, makeup piping, and the points of connection or interfaces with other systems. The staff reviewed the SSAR design criteria, design bases, and safety classification for the HNCW system against the requirements for supplying adequate cooling water for auxiliary equipment for all conditions of plant operation. The staff concludes, as discussed above, that the design of the HNCW system is acceptable and meets the applicable requirements of GDC 2, 4, 5, 45, and 46, and the guidelines of SRP Section 9.2.2.

9.2.13 HVAC Emergency Cooling Water System

The staff reviewed the HECW system in accordance with SRP Section 9.2.2. Staff acceptance of the HECW design is based on meeting GDC 2 as it relates to the system withstanding the effects of natural phenomena; GDC 4 as it relates to the system withstanding the effects of failed equipment and piping during both normal and accident conditions; GDC 5 as it relates to the capability of shared SSCs to perform required safety functions; GDC 44 as it relates to the capability to transfer heat loads from safety-related SSCs to the heat sink under normal and accident conditions, providing suitable redundancy for components given a single active component failure, and the capability to isolate part of the system so that the safety function is not compromised; and GDC 45 and 46 as they relate to permitting inservice inspection and testing, respectively, for safety-related equipment.

The requirements of GDC 5 regarding the sharing of SSCs do not apply because the ABWR is designed as a single-unit facility. An application for a multi-unit facility will require review of the design for compliance with GDC 5.

The HECW is a closed cooling water system whose function is to provide cooling water to the main control room (MCR) air conditioners, reactor building essential electrical equipment room (diesel generator zone) coolers, and control building essential electrical equipment room coolers. The HECW system is required to operate at normal power, reactor shutdown, and after any postulated abnormal reactor conditions, including a LOCA. The HECW system has no primary or secondary containment penetrations. GE states that the entire HECW system is safety-related and is within the scope of the ABWR design. The GE scope of the HECW system includes all piping, valves, pumps, chillers, instrumentation, and controls from the HECW system chillers to their cooling loads.

Auxiliary Systems

The HECW system consists of three mechanically and electrically independent and completely redundant divisions. HECW Division "A" serves Division "A" diesel generator (DG) zone coolers and control building essential electrical equipment room coolers. Division "A" consists of one refrigerator and pump, a surge tank (shared with the corresponding division of RCW), piping, valves, and instrumentation. Divisions "B" and "C" serve their respective DG zone coolers, control building essential electrical equipment room coolers, and MCR air conditioners. Each division consists of two refrigerators and two pumps, a surge tank, and separate piping, valves, and instrumentation. Each refrigeration unit includes a condenser, an evaporator, a centrifugal compressor, refrigerant, piping, and package chiller controls. Cooling water is supplied to the condensers by the corresponding RCW system divisions. Each HECW system division is powered by a different division of the ESF power system. The system also has a chemical feed tank to add chemicals to each division to protect the system components from fouling.

The HECW system and the cooling water lines from the RCW system are designed to the requirements of seismic Category I, 10 CFR Part 50, Appendix B, and QG C. Thus, the system meets Position C.1 of RG 1.29 for seismic classification for safety-related systems. The system is located in the control building, a seismic Category I structure that protects against flood and tornado-missiles. Therefore, the system complies with GDC 2 with regard to protection of safety-related systems against adverse natural phenomena.

Each HECW system division is equipped with a surge tank that GE states is designed to accommodate more than 100 days' system leakage without makeup water during an emergency. The surge tank is connected to the MUWP system, which supplies normal makeup water. The tank includes level switches to detect system leakage and to allow makeup water to be supplied to the tank when required. These switches actuate the makeup water supply valves (open or closed on low or high tank water level, respectively) and annunciate control room alarms for high-high or low-low tank water levels.

The design of the HECW system includes sufficient separation and independence for both mechanical and electrical components of the redundant trains and protection for the system to perform its function under all reactor conditions, including a LOCA, loss of normal ac power, or a single active component failure in the system, or any combination of the above. GE performed a failure analysis of the HECW system and presented the results in the SSAR to demonstrate that failure of a single active component, failure of all power to a single Class 1E power

system bus, or a failure-of-refrigerator signal will not compromise the ability of the system to perform its function. With the system controls set for automatic operation, the system is automatically initiated whenever the HVAC systems in the control building or diesel generator areas are started. The system can also be manually started from the control room. Interlocks for the chillers automatically start the redundant division whenever the operating division fails (e.g., high temperature of the returned cooling water or inadequate chilled water flow). The system flow switches prevent the chiller from operating unless sufficient water is flowing through both the evaporator and the condenser. The chiller units can be controlled individually from the control room by remote-manual switches. The system includes instrumentation and controls for monitoring and controlling system parameters, such as chilled water flow and temperature, condenser water flow, and evaporator discharge flow and temperature. Since the system is not expected to contain any significant level of radioactivity, it has no radiation monitors. GE has provided the design characteristics for the system components (e.g., capacity of the HECW system refrigeration units, chilled water pump flow rate, chilled water and condenser water supply temperatures). GE also submitted the heat removal and flow requirements for the individual system components. This information indicates that any single division of the system is by itself capable of rejecting the total heat from the components the system serves via the refrigerant to the RCW cooling water under all reactor conditions.

From its initial review, the staff stated in the DSER (SECY-91-235) the following three concerns of DSER Section 9.2.12 as Open Item 55:

1. SSAR Table 9.2-9 did not indicate that a single HECW system pump by itself could deliver the required total chilled water flow rate. The staff noted that the statement in SSAR Section 9.2.13.2 that each division contained two 100-percent capacity pumps was contradictory.

In response, GE committed to provide information to reflect that Divisions "B" and "C" consist of two 50-percent capacity pumps and that Division "A" consists of one pump capable of meeting the hydraulic requirements of the division, which would be consistent with the information provided in revised SSAR Table 9.2-9. Incorporation of this information into the SSAR was Confirmatory Item 9.2.13-1 in the DFSE. The SSAR states that each HECW pump has a capacity of approximately 950 L/M (250 gpm). This capacity is sufficient to supply 100 percent of the needs of Division "A" and 50 percent of the needs for Divisions "B" and "C," which is acceptable. Therefore, the

applicable part of DSER Open Item 55 and DFSEF Confirmatory Item 9.2.13-1 are resolved.

2. It was not clear whether the single chemical feed tank provided for the system is safety-related. The staff also noted that GE had not indicated whether the associated isolation valve and its piping are safety-related, or whether any non-safety-related portions of the system are isolated from the safety-related portions of the system if isolation is warranted.

In response, GE indicated that the only non-safety-related portions of the HECW divisions are the chemical addition tank and the piping from the tank to the safety-related valves which isolate the tank from the safety-related portions of the system. This addressed the applicable part of DSER (SECY-91-235) Open Item 55. In the DFSEF, the staff requested that the piping and instrumentation diagrams (P&IDs) be appropriately updated and that the above information be incorporated into the system description. This was DFSEF Confirmatory Item 9.2.13-2. The staff reviewed Figure 9.2-3 of the SSAR and concluded that the surge tank isolation valves are included in the safety-related portion of the system. This is acceptable and resolved DFSEF Confirmatory Item 9.2.13-2 and the applicable part of DSER (SECY-91-235) Open Item 55.

3. The staff noted that it was not clear which division of the HECW supplies chilled water to the DG in Zone C. The staff also noted that SSAR Figure 9.4-4 and Section 9.2.13 were inconsistent in showing the number of divisions for the system.

SSAR Section 9.2.13 currently shows the three divisions of the HECW system. Divisions A, B, and C support diesel generator areas A, B, and C, respectively. This eliminates the discrepancy in the DSER (SECY-91-235) and is acceptable. Therefore, the applicable part of DSER Open Item 55 is resolved.

Based on the above information, the staff finds that the HECW system will comply with GDC 4 regarding protection for the system against dynamic effects resulting from postulated piping failures and internally- and externally-generated missiles, and with GDC 44 regarding system reliability.

The HECW system water quality requirements are established by the MUWP system, as this system is the source of the water for the HECW system surge tanks. (See Section 9.2.10 of this report.)

The design of the HECW system includes provisions to allow periodic inservice inspection of all the system components to ensure the integrity of the system and its capability to perform its intended function. Local display devices indicate vital parameters required in testing and inspections. For example, chilled water flow rate and temperature of the system can be checked by viewing the display of locally-mounted pressure and temperature gauges at the main control panel. The staff reviewed the SSAR and concluded that the system also includes provisions to permit periodic testing of system components as well as the system as a whole. The SSAR states that this testing capability includes structural and leak-tightness visual inspection, tests of entire system operability, and tests of system component operability and performance. This is acceptable. The staff finds that the design of the system complies with the requirements of GDC 45 and 46 for inspection and testing of safety-related cooling water systems.

As a result of further staff review, an additional concern was identified regarding the HECW system. Because of the properties of the refrigerant used in the HECW chiller units, unique problems may arise in recovering the units following a SBO. In the DFSEF, the staff requested GE to provide an analysis regarding the HECW system's ability to recover following a postulated SBO condition. This was DFSEF Open Item 9.2.13-1. As described below, the SSAR, Appendix 1C, provides information regarding the HECW system's response during an SBO. The system can be connected to an alternate ac (AAC) power source within 10 minutes after the initiation of an SBO. The CTG serves as the AAC source. During the SBO, little heat will be generated in the areas served by the HECW since only battery-powered equipment will be running. The temperature increase in the rooms over the 10-minute period will not exceed the qualification limits for the equipment in these areas. Once AAC is available, fans (which are normally powered from Class 1E sources) will be available to remove heat from the areas served by the HECW. As the chillers become available, normal area temperatures will be restored. Furthermore, the SSAR states that the applicant referencing the ABWR will provide the necessary means for restarting the system on an SBO after the AAC is available. The staff concludes that the design provisions in the SSAR give adequate assurance that, during an SBO condition, safety-related equipment served by the HECW system will not exceed its environmental operability limits. This is acceptable and resolved DFSEF Open Item 9.2.13-1.

In the DFSEF, the staff requested GE to provide adequate design description and the ITAAC relating to the HECW system. This was DFSEF Open Item 9.2.13-2. Subsequently, GE provided a revised set of design

Auxiliary Systems

description and ITAAC. The adequacy and acceptability of the design description and the ITAAC are evaluated in Section 14.3 of this report. On the basis of the above, DFSER Open Item 9.2.13-2 is resolved.

The HECW system includes pumps, chillers, valves, piping, surge tanks, makeup piping, and the points of connection with other systems. Portions of the HECW system that are necessary for safe shutdown, accident prevention, or accident mitigation are designed to seismic Category I and QG B requirements. The staff reviewed the SSAR design criteria, design bases, and safety classification for the HECW system against the requirements for supplying adequate cooling water for the safety-related ECCS components and reactor auxiliary equipment for all conditions of plant operation. The staff concludes, as discussed above, that the design of the HECW system is acceptable and meets the requirements of GDC 2, 4, 5, 44, 45, and 46, the applicable criteria of SRP Section 9.2.2.

9.2.14 Turbine Building Cooling Water System

The staff reviewed the non-safety-related TCW system in accordance with applicable portions of SRP Section 9.2.2. Staff acceptance of the TCW system design is based on meeting GDC 2 as it relates to the system withstanding the effects of natural phenomena; GDC 4 as it relates to the system withstanding the effects of failed equipment and piping during both normal and accident conditions; GDC 5 as it relates to the capability of shared SSCs to perform requires safety functions; GDC 44 as it relates to the capability to transfer heat loads from safety-related SSCs to the heat sink under normal and accident conditions, providing suitable redundancy for components given a single active component failure, and the capability to isolate part of the system so that the safety function is not compromised; and GDC 45 and 46 as they relate to permitting inservice inspection and testing, respectively, for safety-related equipment.

The TCW system is a non-safety-related system designed to remove heat for various turbine island auxiliary equipment. The TCW is a closed-loop system consisting of three 50-percent capacity pumps, three 50-percent capacity heat exchangers, a surge tank (shared with the corresponding division of the HNCW system), and associated piping, valves, and instrumentation. In Section 9.2.14 of the DSER (SECY-91-235), the staff noted, as part of Open Item 56, that the description of the TCW in SSAR Section 9.2.14 contained several inconsistencies. The component and system descriptions in SSAR Section 9.2.14.2.3 and in SSAR Figure 9.2-6a did not agree with the descriptions of SSAR Sections 9.2.14.2.1 and 9.2.14.2.2 and GE's responses to RAIs. The discrepancies (in particular, the number of heat exchangers and pump

capacities) required correction. GE submitted information and committed to update the SSAR to eliminate the discrepancies between the figures and the text. GE stated that the system will be a single-loop system consisting of three pumps and three heat exchangers, each of 50-percent-capacity as stated above. Incorporation of the corrected information into the SSAR was Confirmatory Item 9.2.14-1 in the DFSER. The staff reviewed the SSAR and concluded that the inconsistencies had been corrected. This is acceptable and resolved the applicable part of DSER Open Item 56 and DFSER Confirmatory Item 9.2.14-1.

The SSAR includes pump flow requirements, heat exchanger capacity, and turbine service water temperature limits for TCW operation. Demineralized water is added automatically based on surge tank level indications. During normal operation, two pumps are in operation and the third is in standby. The third pump automatically starts on low pump discharge pressure. There are no connections between the TCW and safety-related water systems, and the TCW system is designed in accordance with QG D standards.

The TCW system is located in and near the turbine building, away from safety-related systems. In response to RAIs 430.206 and 430.207, GE indicated that failure of any TCW components, including the atmospheric surge tank, would not cause any safety-related equipment to fail. In the DSER (SECY-91-235), the staff noted that, from equipment layout diagrams reviewed, this statement appears to be true for all equipment shown on the diagrams. The staff also noted that the atmospheric surge tank did not appear on these diagrams and the staff verification that failure of this component would not affect safety-related systems was not possible. The SSAR identifies the surge tank as being located above the TCW pumps in the turbine building. This location would place the tank in an area away from safety-related components, and failure of the tank would not affect any safety-related components, which is acceptable. This description meets Position C.2 of RG 1.29, "Seismic Design Classification," Revision 3, and resolved the remaining part of DSER Open Item 56.

Based on this information, the staff concludes that the TCW system design meets the guidelines of Position C.2 of RG 1.29, pertaining to seismic requirements for non-safety-related systems and components. Therefore, the design meets the requirements of GDC 2 in accordance with SRP 9.2.2.

The requirements of GDC 5 regarding the sharing of SSCs do not apply because the ABWR is designed as a single-

unit facility. An application for a multi-unit facility will require review of the design for compliance with GDC 5.

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

In the DFSE, the staff requested GE to provide adequate design description and the ITAAC relating to the TCW system. This was DFSE Open Item 9.2.14-1. Subsequently, GE provided a revised set of design description and ITAAC. The adequacy and acceptability of the design description and the ITAAC are evaluated in Section 14.3 of this report. On the basis of the above, DFSE Open Item 9.2.14-1 is resolved.

The TCW system includes pumps, chillers, valves, piping, surge tanks, makeup piping, and the points of connection with other systems. The staff reviewed the applicant's proposed design criteria, design bases, and safety classification for the TCW system against the requirements for supplying adequate cooling water for the auxiliary equipment for all conditions of plant operation. The staff concludes, as discussed above, that the design of the TCW system is acceptable and meets the applicable requirements of GDC 2, 4, 5, 44, 45, and 46 and the guidelines of SRP Section 9.2.2.

9.2.15 Reactor Service Water

The staff reviewed the reactor service water (RSW) system in accordance with SRP Section 9.2.1. Staff acceptance of the RSW system design is based on meeting the requirements of GDC 2 as it relates to protecting SSCs important to safety from the effects of natural phenomena; GDC 4 as it relates to protecting SSCs important to safety from the effects of piping and equipment failures during both normal and accident conditions; GDC 5 as it relates to the capability of shared SSCs to perform required safety functions; GDC 44 as it relates to the capability to transfer heat loads from safety-related SSCs to the heat sink under normal and accident conditions, providing suitable redundancy for components given a single active component failure, and the capability to isolate part of the system so that the safety function is not compromised; and GDC 45 and 46 as they relate to inservice inspection and testing of safety-related systems and components.

The portion of the RSW system within the ABWR design scope includes all the piping, valves, instrumentation, and controls within the control building. All other equipment outside the control building, including the RSW pumps, are

outside the ABWR design scope and are the responsibility of the COL applicant.

The SSAR includes a conceptual design and interface requirements for that portion of the RSW system outside the scope of the ABWR design as required by 10 CFR Part 52.

The function of the RSW system is to provide cooling water to the RCW system (reviewed in Section 9.2.11 of this report) for distribution to several safety-related and non-safety-related loads. The RSW is required to operate at normal power, reactor shutdown, hot standby, and after a postulated LOCA. Under each of these conditions, the RSW is required to function both with and without preferred ac power available and with a single active failure.

The RSW system is an open-cycle system that provides cooling water to the RCW heat exchangers. The RSW system supports no other heat loads. The RSW system picks up heat from the RCW heat exchangers and rejects the heat to the UHS, which is to be designed by COL applicants referencing the ABWR design as discussed in Section 9.2.5 of this report. Although earlier SSAR amendments discussed the total heat rate, total flow rate, temperature drop, and pressure drop at the RCW heat exchangers for all identified modes of operation for the RCW system, the staff completed the DSE (SECY-91-235) before receiving in an amendment similar parameters for the RSW system (including identification of sufficient NPSH at pump suction locations for low water levels). This was identified in Section 1.8 of the DSE (SECY-91-235) as Open Item 57. GE submitted information and committed to update the SSAR, identifying these parameters as actions by the COL applicant. This was DFSE COL Action Item 9.2.15-1. SSAR Section 9.2.15.2 requires the COL applicant referencing the ABWR design will submit sufficient information to allow the staff to perform a plant-specific safety evaluation on that portion of the RSW system outside of the ABWR design scope. This is acceptable. On the basis of the above, the applicable part of DSE Open Item 57 is resolved.

The RSW system is composed of three mechanically and electrically independent divisions. Each division consists of its own separate piping from intake to discharge, two pumps, two strainers, valves, and instrumentation. Each RSW division supplies cooling water to one division of the RCW system.

The SSAR states that the RSW system will be able to function during abnormally low or high water levels and that steps are taken to prevent organic fouling that may degrade

Auxiliary Systems

system performance. These steps include installing trash racks, biocide treatment (or non-biocide treatment where biocide treatment is not allowed), and thermal backwash capabilities. In the DFSER, the staff stated that selection of appropriate measures is site-specific and, therefore, the responsibility of the COL applicant. This was DFSER COL Action Item 9.2.15-2. SSAR Section 9.2.15.2.2 states that the COL applicant will design the out-of-scope portion of the RSW system to prevent excessive organic fouling, erosion, and corrosion of the RSW piping. This is acceptable and resolved the applicable portion of DSER Open Item 57.

In the DFSER, staff review of the interface requirements in the SSAR and the certified design material (CDM) was in progress. This was DFSER Open Item 9.2.15-1. Subsequently, GE submitted the design description, the ITAAC, and the interface requirements relating to the RSW system. The adequacy and acceptability of the design description, the ITAAC, and the interface requirements are evaluated in Section 14.3 of this report. On the basis of the above, DFSER Open Item 9.2.15-1 is resolved.

System protection from water-hammer is achieved through the use of high point vents and operational procedures requiring filling and venting of any sections of the system before operation. In the DFSER, the staff stated that the COL applicant should supply these procedures. This was DFSER COL Action Item 9.2.15-3. SSAR Section 9.2.15.2.1(6) states that the COL applicant will submit the above information. This is acceptable.

All portions of the RSW system are designed to seismic Category I, QG C, requirements. SSAR Table 3.2-1 states that the RSW pumps are located in the structures associated with the UHS; therefore, all portions of the system will be located in seismic Category I, flood- and missile-protected structures. The design of the RSW system complies with the requirements of GDC 2 with respect to its protection from natural phenomena, and meets Position C.1 of RG 1.29 with respect to its seismic requirements.

The SSAR states that both the mechanical equipment and piping and electrical equipment, including instrumentation and controls, of the redundant divisions of the RSW system are sufficiently separated and protected to ensure availability of the needed equipment to shut down the reactor in the event of any of the following occurrences: flooding or spraying steam release induced by pipe rupture or equipment failure, pipe whip and jet forces from a postulated nearby high-energy line break, missiles from equipment failure; fire; non seismic Category I equipment failure, or a single active component failure in the system.

In the earlier SSAR amendments, insufficient detail was provided to ensure that this design criteria can be met. Specifically, location and design features for the RSW pump and associated equipment were not specified prior to DSER (SECY-91-235) completion. This was identified in Section 1.8 of the DSER (SECY-91-235) as part of Open Item 57. The SSAR provides this information as an interface requirement and establishes interface criteria to assure an appropriate design. The staff reviewed this information and found it acceptable. This resolved the applicable part of DSER Open Item 57.

A portion of each division of the RSW system is located in separate divisional areas in the control building basement. An RSW pipe break in this area would expose the safety-related RCW heat exchangers to flood water from the UHS through the RSW system. GE incorporated a flood protection feature to isolate the affected division of the RSW system on a high water level signal in the divisional space within the control building. In this way, only the division experiencing the break and subsequent flooding will be affected. The redundant divisions of RCW and RSW will still be available to perform their safety functions. Based on this information, the staff concludes that the ABWR RSW design and related commitments are adequate to ensure compliance with GDC 4 as it relates to protection of safety-related equipment from the dynamic effects resulting from postulated pipe failures, floods, and internally- and externally-generated missiles. This resolved the second part of DSER Open Item 57. Incorporation of this information into the SSAR was identified as Confirmatory Item 9.2.15-1 in the DFSER. SSAR Section 9.2.15.2.1 states that the COL applicant referencing the ABWR will design the out-of-scope portion of the RSW system to withstand the effects of piping and equipment failures. This is acceptable and resolved DFSER Confirmatory Item 9.2.15-1.

The staff notes that, in addition to the interface criteria identified above, the portion of the RSW system within the scope of the COL applicant's action must meet the following:

- the system will be sized to remove the heat associated with the worst-case condition listed in Table 9.2-4 in the SSAR.
- RSW piping length will not exceed 2000 m (~6600 ft) and redundant system isolation capability will be designed to ensure that flooding in the control building resulting from the failure of RSW piping is within the bounds of the flood analysis. This is discussed in Section 3.4.1 of this report.

The requirements of GDC 5 for the sharing of SSCs do not apply because the ABWR is designed as a single-unit facility. An application for a multi-unit facility will require review of the design for compliance with GDC 5.

At least one RSW pump in each of the three divisions of the RSW system will be operating. This configuration ensures immediate availability of the RSW system for plant shutdown in the event of a LOCA. A LOPP concurrent with a LOCA will result in a temporary loss of pumping until the automatically sequenced restart of RSW pumps from the emergency diesel generator loading sequence. Upon the occurrence of a LOCA, the second RSW pump starts and the third heat exchanger in each division is placed in service. The design criteria for the RSW include the requirement that a single active or applicable passive component failure will not compromise the ability of the RSW system to transfer heat loads from the RCW system to the UHS.

GE discussed protection from adverse environmental conditions, such as freezing, icing, and biofouling in its response to Unresolved Safety Issues B-29 and B-32 and Generic Issue (GI) 51. The staff discusses the adequacy of the ABWR design provisions and interface requirements with regard to these issues in Sections 20.1 and 20.2 of this report.

The staff reviewed design information and interface criteria supplied by GE and concludes in view of the above that the RSW system design ensures that the heat removal requirements of GDC 44 will be met and are acceptable.

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

In the DFSER, the staff requested GE to provide adequate design description, the ITAAC, and interface requirements relating to the RSW system. This was DFSER Open Item 9.2.15-2. Subsequently, GE provided a revised set of design description and ITAAC. The adequacy and acceptability of the design description, the ITAAC, and the interface requirements are evaluated in Section 14.3 of this

report. On the basis of the above, DFSER Open Item 9.2.15-2 is resolved.

The design and related commitments of the RSW system comply with the applicable requirements of GDC 2, 4, 5, 44, 45, and 46, and the guidelines of SRP Section 9.2.1 discussed above, and are acceptable.

9.2.16 Turbine Service Water System

The staff reviewed the non-safety-related turbine service water (TSW) system in accordance with applicable portions of SRP Section 9.2.1. The TSW system design is acceptable if it meets the requirements of GDC 2 as it relates to protecting SSCs important to safety from the effects of natural phenomena; GDC 4 as it relates to protecting SSCs important to safety from the effects of piping and equipment failures during normal and accident conditions; GDC 5 as it relates to the capability of shared SSCs important to perform safety functions; GDC 44 as it relates to the capability to transfer heat loads from safety-related SSCs to the heat sink under normal and accident conditions, providing suitable redundancy for components if a single active component fails, and the capability to isolate part of the system so that the safety function is not compromised; and GDC 45 and 46 as they relate to inservice inspection and testing of safety-related systems and components.

The portion of the TSW system design within the ABWR design scope includes all the piping, valves, instrumentation, and controls within the turbine building. All other equipment outside the turbine building, including the TSW pumps, are outside the ABWR design scope and are the responsibility of the COL applicant.

The SSAR includes a conceptual design and interface requirements for that portion of the TSW system outside the scope of the ABWR design as required by 10 CFR Part 52.

The TSW system is a non-safety-related system designed to transfer heat from the TCW system heat exchangers to the power cycle heat sink. The TSW includes three 50-percent capacity pumps, three duplex strainers, and associated piping, valves, and instrumentation. In the DSER (SECY-91-235), the staff stated that GE did not provide system parameters, (pump flow requirements, system design pressure) but submitted a requirement that water supplied to the TCW heat exchangers be at a temperature not to exceed approximately 40 °C (100 °F). In response, GE committed to update the SSAR concerning the design of the TSW system. SSAR Table 9.2-16 provides the specific system parameters described above.

Auxiliary Systems

This is acceptable and resolved DFSER Confirmatory Item 9.2.16-1.

In the DFSER, the staff noted contradictory information concerning the number of pumps in the system. The text of SSAR Section 9.2.16 referred to two pumps and two duplex strainers. However, SSAR Figure 9.2-8 and Table 9.2.17 both showed three 50-percent capacity pumps. Resolution of this discrepancy was identified as Open Item 9.2.16-1 in the DFSER. The SSAR clarifies that the system consists of a single loop containing three vertical wet pit pumps. During normal operation, two pumps are operating and the third is in standby. The standby pump starts automatically if an operating pump trips or if the pump discharge pressure drops below a preselected limit. This clarification is acceptable and resolved DFSER Open Item 9.2.16-1.

The TSW system is located in the intake structure (the power cycle heat sink pump house) and the turbine building. The system does not have any connections with safety-related systems. The applicant must demonstrate that all safety-related components, systems, and structures are protected from flooding in the event of a pipeline break in the TSW system in order to meet Position C.2 of RG 1.29, and thus, comply with GDC 2. The staff recognized the site-specific location of some TSW components and stated in the DSER (SECY-91-235) that this requirement may need to be specified as an interface requirement. This was identified in Section 1.8 of the DSER as Open Item 58. Incorporation of this requirement in the SSAR was identified as Confirmatory Item 9.2.16-2 in the DFSER. GE stated that the COL applicant should submit an interface requirement for flood protection of safety-related SSCs in case of TSW component failures. The SSAR includes sufficient information to ensure that an applicant referencing the ABWR can design the out-of-scope portion of the TSW system to prevent flood damage to safety-related equipment as a result of a break in the TSW piping. The staff notes that in the SSAR, Section 3.4.1 includes a flood analysis which indicated that any flooding resulting from a break in the TSW system will be prevented from affecting safety-related equipment by several means. First, a break in the TSW line will result in a high water level alarm in the condenser pit. The operator can then isolate the system. If the operator is unsuccessful in isolating the system, flood waters would rise to plant grade where it would flow out of the truck door and onto the ground at this elevation. Second, the below-grade tunnel which connects the turbine, radwaste, and reactor buildings is sealed at all ends to prevent water from entering any of the buildings. On the basis of this information, the staff concludes that flooding as a result of a break in the TSW line will not adversely affect any safety-related equipment. Therefore, the

guidelines of Position C.2 of RG 1.29 and the requirements of GDC 2 can be met, and DFSER Confirmatory Item 9.2.16-2 is resolved. The CDM aspects are addressed in the discussion of DFSER Open Item 9.2.16-2 below.

The requirements of GDC 5 for the sharing of SSCs do not apply because the ABWR is designed as a single-unit facility. An application for a multi-unit facility will require review of the design for compliance with GDC 5.

The remaining requirements of SRP Section 9.2.1 (GDC 4, 44, 45, and 46) do not apply because the TSW system is non-safety-related and has no connections to safety-related systems.

In the DFSER, the staff requested GE to provide adequate design description, interface requirements, and the ITAAC relating to the TSW system. This was DFSER Open Item 9.2.16-2. Subsequently, GE provided a revised set of design description and ITAAC. The adequacy and acceptability of the design description and the ITAAC are evaluated in Section 14.3 of this report. On the basis of the above, DFSER Open Item 9.2.16-2 is resolved.

The design and related commitments of the TSW system comply with the requirements of GDC 2 as discussed above. The requirements of GDC 4, 5, 44, 45, and 46 do not apply. In view of the above the staff concludes that the system design and related commitments meet the applicable acceptance criteria of SRP Section 9.2.1 and are acceptable.

9.3 Process Auxiliaries

9.3.1 Compressed Air Systems

The design of the compressed air (CA) systems are discussed in SSAR Sections 6.2.5 (Atmospheric Control System), 6.7 (High Pressure Nitrogen Gas Supply), 9.3.6 (Instrument Air), and 9.3.7 (Service Air) and were reviewed in accordance with SRP Section 9.3.1. The review of the CA systems involved a review of information in the SSAR and GE's responses to staff RAIs. The acceptance criteria for the safety-related portions of the CA systems is provided in SRP Section 9.3.1 and includes compliance with GDC 1 as it relates to systems and components important to safety being designed, fabricated, and tested to quality standards in accordance with the importance of the safety functions to be performed; GDC 2 as it relates to safety-related CA systems being capable of withstanding the effects of natural phenomena; and GDC 5 as it relates to the capability of shared systems, and components to perform required safety functions. The

staff reviewed the CA systems for compliance with GDC 1 as discussed in Section 3.2 of this report.

The requirements of GDC 5 for the sharing of SSCs do not apply because the ABWR is designed as a single-unit facility. An application for a multi-unit facility will require review of the design for compliance with GDC 5.

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

The ACS system establishes and maintains an inert atmosphere within the primary containment during all plant operating modes except (1) during shutdown for refueling or equipment maintenance, and (2) during limited periods of time to permit access for inspection at low reactor power (15 percent). The ACS is non-safety-related except as necessary to ensure primary containment integrity (e.g., penetrations and isolation valves). The ACS includes nitrogen storage tanks, vaporizers, valves and piping carrying nitrogen to the containment, valves and piping from the containment to the SGTS and reactor building HVAC exhaust line, non-safety oxygen-monitoring, and all related instrumentation and controls. The ACS provides nitrogen from the nitrogen evaporator to the HPIN system during normal operation.

The HPIN system consists of both safety-related and non-safety-related portions. A single non-safety-related line provides a continuous nitrogen supply to all pneumatically operated components in the primary containment during normal operation. During normal operation, the HPIN system is supplied from the nitrogen gas evaporator/storage tank through the makeup line from the ACS.

The safety-related portion of the system consists of two independent divisions; each division contains a safety-related nitrogen supply capable of supplying 100-percent of the requirements of the division being serviced. High-pressure nitrogen gas storage bottles supply nitrogen gas for the safety-related portion of the system. Tielines connect the non-safety-related portion of the system to each division of the safety-related portion of the system. Each tieline has a motor-operated isolation valve.

The IA system provides dry, oil-free, CA for valve actuators, non-safety-related instrument control functions, and general instrumentation and valve services outside the containment. (All instrumentation and control systems inside the containment are supplied with nitrogen gas during normal plant operation.) The primary containment penetrations of the IA system are of seismic Category I, QG B, design and are equipped with sufficient isolation valves to satisfy single-failure criteria. In GE's response to RAI 430.215, the staff noted in the DSER (SECY-91-355), as an unnumbered confirmatory issue, that the reference to "... containment penetrations and drywell penetrations of the instrument air system ..." in SSAR Section 9.3.6.1.1 should be revised to reference primary containment penetrations only. The correction of the SSAR text was identified as DFSEER Confirmatory Item 9.3.1-1. GE subsequently made this correction in Amendment 24 of the SSAR. DFSEER Confirmatory Item 9.3.1-1 is resolved.

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

As noted earlier, only the HPIN system provides compressed gas to safety-related components. However, the ACS and the IA system directly interface with the HPIN system and could affect the reliability of safety-related components relied upon to mitigate the consequences of design-basis events. Therefore, the staff reviewed these three systems in assessing the adequacy of the CA systems in accordance with acceptance criteria in SRP Section 9.3.1.

The staff reviewed the SSAR to determine the safety-related portions of the CA system and stated in the DSER (SECY-91-355) that the text and figures in SSAR Section 6.7 did not clearly state which portions of the HPIN system are safety-related. However, the response to an RAI, and the valve and instrument numbers on a revision to SSAR Figure 6.7-1 specify which portions of the system are safety-related. These portions include the nitrogen storage bottles and their headers up to and including valves F002A through F002D, piping and valves from F002 (A through D) to the accumulators for the automatic depressurization system (ADS) valves, and piping from the cross-tie valves F012A and F012B to the piping leading to the accumulators (identified previously). Additional safety-

Auxiliary Systems

related piping in the IA system and ACS include piping and valves from F200 to F208, inclusive.

The staff noted an inconsistency in the designation of the inboard isolation valve. GE referred to this valve as F209 in all drawings and as F208 in the responses to RAIs. In DSER (SECY-91-355) Table 9.3.1-1, the staff identified (DSER Open Item 69) additional numbering inconsistencies, including valve numbers and valve operator types. The staff also stated [DSER (SECY-91-355) Open Item 70] that the SSAR text and figures did not include all information on the safety classification of components before the staff completed DSER (SECY-91-355). While the information in response to requests for information is acceptable, the staff stated in the DSER (SECY-91-355) that this information should be fully incorporated into the SSAR. The inconsistencies and lack of information noted in the DSER (SECY-91-235) were addressed in revised figures to be incorporated into the SSAR. This information resolved DSER Open Items 69 and 70; however, the staff identified incorporation of this information into the SSAR as Confirmatory Item 9.3.1-2 in the DFSER. GE submitted Amendment 20 of the SSAR which corrected the inconsistencies and included the additional information. This is acceptable, and resolved DFSER Confirmatory Item 9.3.1-2.

Contrary to the guidelines of the SRP, the SSAR does not indicate the failure mode for the valves in the HPIN system. Except as noted below, and assuming that the MOVs fail "as-is," the system configuration is acceptable. In the DSER (SECY-91-355), the staff stated as Open Item 71 that information related to the failure mode of components should be incorporated into the SSAR text and drawings. GE indicated in a letter dated March 11, 1992, that the motor-operated valves in the air systems fail as-is; air-operated valves fail open (unless otherwise indicated) in the HPIN, IA, and SA systems and fail closed in the ACS. This information resolved DSER Open Item 71; however, identification of the failure modes of the valves in the CA systems (i.e., confirmation of the failure states) in the SSAR was identified as DFSER Confirmatory Item 9.3.1-3. GE modified SSAR Figures 6.7-1, 6.2-39, 9.3-6, and 9.3-7 to include the failure modes of all pneumatically-operated valves in the CA systems. All motor-operated valves fail as-is. This is acceptable to the staff and resolved DFSER Confirmatory Item 9.3.1-3.

In the DSER (SECY-91-355), the staff stated that Valves AO F018A and B in SSAR Figure 6.7-1 were identified as "NC, FO" (i.e., normally closed and fail open) and that this was inconsistent with Section 19E.2.1.2.2.2(b), which states that upon a loss of power, the operator will have to manually open these valves. The staff noted that this is also inconsistent with a design which should protect

the storage bottles from inadvertent depressurization due to a postulated line break. Confirmation that these valves do not fail in the open position was identified as DSER (SECY-91-355) Open Item 72. In a letter dated March 11, 1992, GE changed these valves to motor-operated valves that are normally locked closed and as such they will fail as-is, closed. This resolved the DSER Open Item 72. Incorporation of this information into the SSAR was identified as DFSER Confirmatory Item 9.3.1-4. In resolving DFSER Confirmatory Item 9.3.1-2, GE changed the designation of valves F018A and B to F003A and B in SSAR Amendment 20 and changed these valves from pneumatically-operated valves to motor-operated valves which are normally closed. The staff found the modifications acceptable. DFSER Confirmatory Item 9.3.1-4 is resolved.

The vessels, piping, and fittings of the safety-related portions of the HPIN system (except penetrations) are designed to the requirements of seismic Category I, ASME Code Section III, Class 3, QG C, and Quality Assurance B. The SSAR states that the cross-tie valves (i.e., F012A and B) that connect the safety-related portions of the HPIN system to the non-safety-related portions are safety-related. While two isolation valves are not provided on the non-safety-related/safety-related interface within the HPIN system, the system is deemed adequate in accordance with information GE provided in the response to RAI 430.211. One isolation valve is provided between each division of the safety-related HPIN system and the non-safety-related portion of the system. Additionally, check valves are provided to prevent backflow of nitrogen from the accumulators through any possible break in the non-safety-related portion of the system. The inboard isolation valves are check valves and each safety-related accumulator is downstream of a separate check valve. The accumulators are sized to perform their function. The combination of isolation valves, check valves, and the sizing of the accumulators ensures that the system would fulfill its function in the event of a rupture in the non-safety-related portion of the piping.

The piping and valves for the containment and drywell penetrations for the HPIN and IA systems are designed to the requirements of seismic Category I, ASME Code Section III, Safety Class 2, QG B, and Quality Assurance B. The isolation provisions for the ACS primary containment penetrations include two isolation valves that are both located outside the primary containment, which is not strictly in conformance with GDC 56. However, the penetrations do not extend inside the containment, and an inboard isolation valve would not be practical (as described by GE in the response to RAI 430.209). An inboard isolation valve for the ACS would be exposed to a more severe environment and would not be easily accessible for

inspection, surveillance, and maintenance. The staff approved a similar design for the GESSAR II BWR/6 design. Therefore, the staff determined that the isolation design is acceptable. The containment isolation valve provisions are reviewed in detail in Section 6.2.6 of this report.

Based on this information, the staff concluded that the CA system complies with Positions C.1 and C.2 of RG 1.29 regarding the ability of the system to withstand the effects of earthquakes.

The safety-related portions of the HPIN system are located within the reactor building. The reactor building is designed to withstand and protect equipment from tornados, externally-generated missiles, floods, and other natural phenomena. In addition, the safety-related portions of the HPIN system will retain their function during a LOCA and seismic events in which non-safety-related portions may be damaged. SSAR Section 6.7.3 states that the space separating the pipe routing of Divisions I and II of nitrogen gas is sufficient to prevent a strike by a single high-energy whipping pipe, the jet force from a single broken pipe, or an internally-generated missile from preventing the other division from accomplishing its safety function. Thus, the system satisfies GDC 2 regarding protection of the system from natural phenomena.

The ADS valves will perform their safety-related functions using compressed gas that will be provided by the HPIN under most conditions. This nitrogen is supplied during normal operation from the ACS nitrogen storage tank. During design-basis events, nitrogen is supplied by accumulators charged by either the ACS, during normal operation, or by nitrogen bottles during periods when the ACS is unavailable. In addition, stored nitrogen can be used to replenish the accumulators or to supplement their operation. The ABWR design uses nitrogen containing particles up to 5 microns. As discussed in DSER Open Item 73 (SECY-91-355), this does not comply with the guidance of ANSI MC 11.1-1976, which recommends that nitrogen used by safety-related components contain particles no larger than 3 microns. In response to this issue, GE indicated that the filters in the HPIN system will be able to remove particles larger than 5 microns from the system, and the nitrogen supply subsystem will supply oil-free nitrogen with a moisture content of less than 2.5-ppm. The 5-micron capacity of the filters is not in compliance with the requirements of ANSI MC 11.1-1976, and GE had not provided sufficient justification for the staff to allow deviation from the 3-micron criterion. In the DFSE, the staff stated that GE should either commit to the 3-micron requirement of the standard or provide a commitment to use equipment demonstrated to be unaffected by the use of nitrogen containing particulates of

the larger size. DSER Open Item 73 (SECY-91-355) was identified as DFSE Open Item 9.3.1-1. Subsequently, GE modified the SSAR to clarify the design requirements for components using this nitrogen source. GE submitted Amendment 24 to the SSAR. The staff reviewed Section 6.7.2 of this amendment and concludes that GE had clarified that any equipment using this nitrogen will be capable of operating with nitrogen containing 5-micron particulates. The staff finds the modification acceptable. DFSE Open Item 9.3.1-1 is resolved.

IE Bulletin No. 80-01 concerns the operability of the ADS pneumatic supply. The bulletin states that the ADS pneumatic supply may not be operable for all possible events because of a combination of misapplication of check valves, a lack of testing of the accumulator system backing up each ADS valve operator, and questions about the continued operability of the pneumatic supply in a seismic event.

The bulletin requires licensees of GE BWR facilities which use a pneumatic operator for ADS function to:

- (1) Determine if the facility has installed hard-seat check valves to isolate the ADS accumulator system from the pneumatic supply system.
- (2) Determine if periodic leak tests have been performed in the ADS accumulator systems to assure emergency pneumatic supply for the FSAR-required number and duration of valve operations.
- (3) Review seismic qualifications of the ADS pneumatic supply system:
 - (a) from accumulator system isolation check valve to ADS valve operator,
 - (b) from isolation valve outside containment up to ADS accumulator check valve.
- (4) Based upon determination of Items 1, 2, and 3 above, evaluate the operability of the ADS for the conditions under which it is required to be operable, including a seismic event. If operability cannot be established, adhere to appropriate technical specification action statement.

Operational experience has shown that check valves with hard seats may lead to excessive valve leakage, thus undermining the ability of the accumulators to provide the required pneumatic fluid to ensure that the ADS valves actuate. The ABWR design may use check valves with a hard seat. GE states that accumulator operability will be

Auxiliary Systems

assured by ensuring that system leakage does not exceed 28 L/h (1 scfh) per valve. SSAR Sections 6.7.2 and 6.7.4 state that periodic leakage testing will be performed to ensure that the leakage rate does not exceed 28 L/h (1 scfh) for each valve. Section 6.7.3 states that the safety-related portions of the ADS system are seismic Category I. The section of system piping addressed in the bulletin is classified as safety-related for the HPIN system. Thus, these sections of the system will be able to withstand the conditions associated with a seismic event. Based on this information, the staff concludes that the HPIN design is sufficient to ensure that the ADS SRV accumulators will supply adequate nitrogen to the ADS SRVs to ensure proper valve actuation. The system design adequately addresses the issues in IE Bulletin No. 80-01.

The staff stated in the DSER (SECY-91-355) that SSAR Section 9.3.6.1.2 indicated that the non-safety-related IA system is also used as a backup to the nitrogen system when, during normal operation, the nitrogen gas supply pressure drops below a specified setpoint. The staff noted that this conflicted with GE's earlier response to RAI 430.218, dated March 11, 1992, which states: "Instrument air system does not serve as a backup to HPIN system during normal operation" The resolution of this discrepancy was identified as Open Item 74. Subsequently, GE committed to revise the response to RAI 430.218 to indicate that the IA system can be used as a backup to the HPIN system when nitrogen pressure drops below the system low pressure setpoint and recovery efforts have failed. IA would be used until repairs to the HPIN system are completed. This information resolved DSER Open Item 74, however, incorporation of this information into the SSAR was identified as DFSER Confirmatory Item 9.3.1-5. GE supplied the required information in Amendment 22 which is acceptable to the staff. DFSER Confirmatory Item 9.3.1-5 is resolved.

In evaluating the IA system as a potential backup to the HPIN system, the staff stated in the DSER (SECY-91-355) that it had found that the system complies with all aspects of the ANSI MC 11.1-1976 criteria except for particulate size. The ABWR design proposes a 5-micron criterion for particulate size that is contrary to the 3-micron criterion of the ANSI standard. This was identified as DSER Open Item 75. GE had not justified this aspect of the CA system design, therefore, the IA system's compliance with the requirements of GDC 1 remained an outstanding issue. The staff reclassified DSER Open Item 75 as Open Item 9.3.1-2 in the DFSER. The staff reviewed Section 9.3.6.2 of Amendment 24 of the SSAR and concludes that GE has clarified that any equipment using this air will be capable of operating with air containing 5-micron particulates. The staff finds the modification acceptable. DFSER Open Item 9.3.1-2 is resolved.

The staff reviewed the preoperational testing of the CA systems and compliance with RG 1.68.3 as addressed in Section 14 of this report.

GE submitted the design description and the ITAAC relating to the CA systems. This was identified as DFSER Open Item 9.3.1-3. The adequacy and acceptability of the design description and the ITAAC are evaluated in Section 14.3 of this report. On the basis of this evaluation, DFSER Open Item 9.3.1-3 is resolved.

The CA systems include all components and piping and the points of connection with other systems. The safety-related HPIN system provides a continuous nitrogen supply to safety-related components and is classified as seismic Category I and QG C. The staff reviewed the applicant's design and design criteria for the safety-related CA systems to verify that they conform to the Commission's regulations in the GDC, and to applicable regulatory guides and industry standards. As discussed in detail above, the staff concludes that the design of the CA systems is acceptable and conforms to the requirements of GDC 1, 2, and 5 for quality standards, protection from natural phenomena, and sharing of systems and components, and meets the guidelines of SRP Section 9.3.1.

9.3.2 Process and Post-Accident Sampling Systems

9.3.2.1 Process Sampling System

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

In DSER (SECY-91-355) Open Item 98, the NRC staff determined that SSAR Section 9.3.2.1 contained insufficient information for the staff to evaluate conformance with SRP Section 9.3.2 in the following areas:

1. Under SRP Section 9.3.2, the PSS should include the capability to obtain samples from at least the following points: main condenser evacuation system off gas, SLCS tank, and sumps inside containment and other

locations given in SRP Section 11.5 and those specified in the SSAR.

2. The guidelines in Position C.2 of RG 1.21, "Measuring, Evaluating and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluent from Light-Water-Cooled Nuclear Power Plants" and positions of RG 1.56, "Maintenance of Water Purity in Boiling Water Reactors," Revision 1, should be used to meet the requirements of GDC 13, 14, 26, 63, and 64 unless otherwise acceptable alternatives were proposed. The SSAR did not indicate conformance with these guidelines.
3. In accordance with ANSI N13.1-1969 provisions should be made to ensure that representative samples are taken from gaseous process streams and tanks. This needed to be addressed in the design.
4. To meet 10 CFR 20.1003 in keeping radiation exposures ALARA, and the requirement of GDC 60 to control the release of radioactive materials to the environment, passive flow restrictions to limit reactor coolant loss from a rupture of the sample line should be provided.
5. To meet the requirements of GDC 1 and 2, the seismic design and quality group classification of sampling lines for the PSS should conform to the classification of the system to which each sampling line and component is connected.

GE responded in its March 11, 1992, letter entitled "GE Responses to the Resolution of Issues Related to ABWR DSER Chapters 1, 2, 3, 4, 5, 6, 9, 10, 12, 13, 14, and 15 (SECY-91-355)." In an attachment to this letter (Response to Open Item 98), GE stated that a capability is provided to obtain samples from the main condenser evacuation off gas, SLCS tank, sumps inside containment, liquid radwaste system process lines, and liquid radwaste system collection and sampling tanks. They said that the guidelines of RGs 1.21 and 1.56, and ANSI N13.1-1969 will be used, except that passive flow restrictors are not provided in reactor sampling lines to control the release of radioactive materials from a ruptured sample line. These devices become crud traps during normal operation and overly restrict sampling flow rate during shutdowns when the reactor is at low pressure. They stated that each reactor water sampling line is provided with two remotely operable isolation valves to limit reactor water loss from a sample line rupture. GE also indicated that the seismic design and group classification of sampling lines and their components will conform to the classification of the system into which they are connected. The staff finds this response accept-

able and, therefore, DSER Open Item 98 was resolved. The staff concludes that the system meets the cited requirements and guidance and is, therefore, acceptable.

GE has submitted ITAAC and design description in Section 2.11.20 of the CDM for the PSS. This was DSER Open Item 9.3.2.1-1. GE provided a revised set of design description and ITAAC. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Section 14.3 of this report. On the basis of the above, this item is resolved.

9.3.2.2 Post-Accident Sampling System

After the accident at Three Mile Island Unit 2, the staff recognized the need for an improved post-accident sampling system (PASS) to determine the extent of core degradation following a severe reactor accident. Criteria for an acceptable sampling and analysis system are specified in 10 CFR 50.34(f)(2)(viii) and Item II.B.3 of NUREG-0737. According to these documents, the PASS should have the capability to obtain and quantitatively analyze reactor coolant and containment atmosphere samples without exposing any individual to radiation exceeding 5-rem to the whole body or 75-rem to the extremities (GDC 19) during and following an accident in which there is core degradation. Materials to be analyzed and quantified include certain radionuclides that are indicators of the severity of core damage (e.g., noble gases, isotopes of iodine and cesium, and nonvolatile isotopes), hydrogen in the containment atmosphere, and total dissolved gases, boron, and chloride in reactor coolant samples.

In DSER (SECY-91-355) Open Item 99, the NRC staff determined that the PASS design as described in SSAR Section 9.3.2 was not adequate. The staff stated that GE needed to address the Item II.B.3 of NUREG-0737 and indicate the PASS provisions required to satisfy each of the 11 specified criteria. The staff said that the upper limit for activity levels in liquid samples of 1 Ci/cm³ in GE's PASS design (SSAR Section 9.3.2.3.1) was not justified. Item II.B.3 of NUREG-0737, Criterion 9, and RG 1.97, "Instrumentation for Light Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," Revision 3, specify reactor coolant and sump gross activity sampling capability in the 1 µCi/ml to 10 Ci/ml range. Delaying sampling until the PASS sample radioactivity decays to 1 Ci/ml is unacceptable because inordinate and unjustified delay in obtaining sample radioactivity results. All PASS systems in operating reactor plants are presently designed with the capability to sample liquids with up to 10 Ci/ml radioactivity.

Auxiliary Systems

GE responded (March 11, 1992, letter) explaining that the ABWR PASS has been designed to meet the intent of the 11 requirements listed in NUREG-0737. GE also identified several exceptions to some of the requirements of NUREG-0737. The NRC staff reviewed this information and found it acceptable, subject to revising the SSAR to include the following design features:

1. SSAR Section 9.3.2 indicates that PASS is designed for upper limit for activity samples of 1 Ci/cm^3 . All PASS systems in operating reactor plants are designed with capability of sampling liquids up to 10 Ci/ml radioactivity. GE needs to design PASS with the capability of sampling liquids up to 10 Ci/ml . This is DFSE Open Item 9.3.2.2-1.
2. The ABWR PASS does not have capability to obtain pressurized or unpressurized reactor water samples for dissolved gas analysis. GE needs to design PASS to be capable of obtaining reactor water samples 24 hours after the end of power generation in order to evaluate concentrations of dissolved gases and chlorides in the reactor coolant. The information on the amounts of dissolved hydrogen, oxygen and chlorides in the reactor coolant is an important factor in evaluating post-accident conditions existing in the reactor vessel. This is DFSE Open Item 9.3.2.2-2.

The staff concluded in the DFSE that, subject to the resolution of DFSE Open Items 9.3.2.2-1 and 9.3.2.2-2 described above, GE will have adequately described the ABWR PASS.

GE addressed DFSE Open Item 9.3.2.2-1 in its SSAR, Section 9.3.2.3.1, which states that the area radiation is considered safe when the sample radioactivity is about 1 Ci/cm^3 . When the sample radioactivity level is higher than 1 Ci/cm^3 , abnormal or emergency conditions will be used to assess the situation. This is acceptable.

GE responded to DFSE Open Item 9.3.2.2-2 in its letter of January 28, 1993. The staff also discussed this issue in a telephone conference with GE on January 28, 1993. GE explained that whenever core uncovering is suspected, the reactor vessel is rapidly depressurized, and thus pressurized reactor water samples are not necessary. In a letter dated April 26, 1993, GE revised SSAR Section 9.3.2.3.1 to document this explanation. GE has also included this information in the SSAR. On the basis of the actions above, DFSE Open Item 9.3.2.2-2 is resolved.

In SECY-93-087, the staff recommended that the Commission approve its position that for evolutionary and passive ALWRs of boiling water reactor design, there

would be no need for the PASS to analyze dissolved gases in accordance with the requirements of 10 CFR 50.34(f)(2)(viii) and Item III.B.3 of NUREG-0737. In its April 2, 1993, SRM, the Commission approved the staff position to exempt the PASS for the evolutionary and passive ALWRs of boiling water reactor design from analyzing dissolved gases in accordance with the requirements of 10 CFR 50.34(f)(2)(viii) and Item III.B.3 of NUREG-0737.

In SECY-93-087, the staff also recommended that the Commission approve the deviation from the requirements of Item II.B.3 of NUREG-0737 with regard to the requirements for sampling reactor coolant for boron concentration and activity measurements using the PASS in evolutionary and passive ALWRs. The modified requirement would require the capability to take boron concentration samples and activity measurements 8 hours and 24 hours, respectively, following the accident. In its April 2, 1993, SRM, the Commission approved the staff position to require the capability to take boron concentration samples and activities measurements 8 hours and 24 hours, respectively, following the accident.

The ABWR design will have PASS which meets the requirements of 10 CFR 50.34(f)(2)(viii) and Item II.B.3 of NUREG-0737 with the modifications described in SECY-93-087. The system will have the capability to sample and analyze for activity in the reactor coolant and containment atmosphere 24 hours following the accident. This information is needed for evaluating the conditions of the core and will be provided during the accident management phase by the containment high-range area monitor, the containment hydrogen monitor and the reactor vessel water level indicator. The need for PASS activity measurements will arise only during the accident recovery phase and 24 hours sampling time is, therefore, adequate. PASS will also be able to determine boron concentration in the reactor coolant. It will be capable to make this determination within 8 hours following the accident. The concentration of boron is required for providing insights for accident mitigation measures. Immediately after the accident this information will be obtained by the neutron flux monitoring instrumentation which is designed to comply with the criteria of RG 1.97, and which has fully qualified redundant channels capable to monitor over the full power range. Boron concentration measurements will not be, therefore, required for the first 8 hours after the accident.

In order to approve an exemption from the requirements of 10 CFR 50.34(f)(2)(viii) concerning the elimination of analyses for dissolved gases and chlorides in the reactor coolant, special circumstances must exist. For the ABWR, whenever core uncovering is suspected, the reactor vessel

is depressurized to approximately the pressure within the wetwell and the drywell which results in partial release of the dissolved gases. Under these conditions, pressurized samples would not be meaningful. Therefore, application of the regulation in this particular circumstance would not serve the underlying purpose of the rule. During accidents when the reactor vessel has not been depressurized (such as when a small amount of cladding damage has occurred), reactor coolant samples could be obtained by the process sampling system.

With regards to the need for chloride analysis, determination of chloride concentrations is of a secondary importance because it is needed only for determining the likelihood of accelerated primary system corrosion which is a slow-occurring phenomenon. Chloride analyses could be performed on the samples taken by the process sampling system. In this case, therefore, the intended purpose of the rule could be achieved without the need for the PASS to have chloride sampling capabilities.

Accordingly, special circumstances required by 10 CFR 50.12(2)(ii) exist for the ABWR in that the regulation would not serve the underlying purpose of the rule in one circumstance and is not necessary in the other circumstance to achieve the underlying purpose of the rule because the intent of rule could be met with alternate design requirements proposed by the applicant. On this basis, the staff concludes that the exemption from analyzing dissolved gases and chlorides in the reactor coolant sample is justified.

GE has submitted ITAAC and design description of the post-accident sampling system as a part of the process sampling system in ITAAC Section 2.11.20. This was DFSE Open Item 9.3.2.2-3. GE provided a revised set of design description and ITAAC. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.1 of this report. On the basis of the above, DFSE Open Item 9.3.2.2-3 is resolved.

9.3.3 Non-Radioactive Drain System

The staff reviewed the non-radioactive drain (NRD) system in accordance with SRP Section 9.3.3. The NRD system is acceptable if it meets the requirements of GDC 2 as it relates to safety-related portions of the system being capable of withstanding the effects of natural phenomena; GDC 4 as it relates to the capability of the system to withstand the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents; and GDC 60 as it relates to providing a means to control the release of radioactive materials in liquid effluent, including anticipated operational occurrences.

SSAR Section 9.3.3 discusses the NRD, which is a non-safety-related system designed to transfer effluents that are not radioactive or potentially radioactive. Radioactive effluents are transferred by the radioactive drain transfer system (DTS), which is discussed in SSAR Section 9.3.8 and reviewed in Section 9.3.8 of this report.

Initially the NRD was not considered part of the ABWR design in the DFSE. Subsequently, GE brought portions of this system within the ABWR design scope. The portions of the system within standard plant buildings are within the ABWR design scope, while those portions outside these buildings are outside of the ABWR design scope. The portions of the system outside the design scope, as well as the design details of the in-scope portion of the system, will be provided by the COL applicant referencing the ABWR design. GE has provided a conceptual design and interface requirements for the out-of-scope portion of the system as required by 10 CFR Part 52.

The NRD system consists of sump pumps, valves, and associated piping and instrumentation to direct waste liquids, valve and pump leakoffs, and component drains and vents to the sanitary drainage portion of the PSW system. SSAR Table 3.2-1 states that the system serves all areas of the plant within the ABWR scope except for the primary containment and the MST. The areas of secondary containment housing the EDGs contain no drains. The system drains by gravity and contains no active components from the point of drainage to the sumps. Valves that are relied upon to prevent backflow can be inspected and tested and are designed to withstand the effects of a SSE. Flooding is prevented by designing the system with sufficient capacity to accommodate expected flooding as well as placing safety-related equipment on raised pads or gratings (see Section 3.4.1 of this report for the staff's review of flood protection). Based on this information, the staff concludes that the system can withstand the effects of natural phenomena and, therefore, meets the requirements of GDC 2.

The NRD system is arranged with separate piping in each quadrant. Flooding or backflow in one quadrant cannot affect other quadrants. This system has no connections to the radioactive drains transfer system, and drains are designed to withstand the adverse effects (including high pressure) associated with pipe and equipment failures in building compartments. Open drainage lines that are needed to maintain an air pressure differential are provided with a water seal. Based on this information, the staff concludes that the NRD system meets GDC 4 as it relates to protection of the system from environmental and dynamic effects associated with piping and components failures.

Auxiliary Systems

The NRD system design ensures that radioactive material cannot be discharged. However, effluent is sampled for radiation prior to discharge. In addition, level switches are provided in each sump to monitor leakage (see Section 5.2.5 of this report for the evaluation of leakage detection methods). GE included a COL action item in Amendment 29 of the SSAR requiring COL applicants referencing the ABWR design to develop a sampling and analysis program to ensure that radioactive liquids are not inadvertently being discharged from the nonradioactive drain system. This was identified as DFSEER COL Action Item 9.3.3-1. Based on these design and operational provisions, the staff concludes that the system contains adequate means to detect and control the release of radioactive materials in the system effluent and meets the requirements of GDC 60 and is acceptable.

SSAR Amendment 29 includes a conceptual design and interface requirements for the part of the NRD system outside of the ABWR design scope. The COL applicant referencing the ABWR design shall provide the details of the system design from the standard plant buildings to the site discharge structure. The system collects wastewater from plant buildings, precipitation, and other surface runoff and directs this water to dual settling basins where suspended solids are settled and oil is collected. Tests and analyses are performed to meet the plants discharge permit. Failure of this portion of the system will not adversely affect any safety-related equipment. Based on this information, the staff concludes that the out-of-scope portion of the nonradioactive drain system will also meet the requirements of GDC 2, 4, and 60.

The staff concludes that the design criteria, conceptual design, and interface requirements provided in the SSAR will allow the applicant to design a nonradioactive drainage system which will comply with the GDC.

GE submitted the design description, ITAAC, and the relevant interface requirements relating to the NRD system. The adequacy and acceptability of the design description, the ITAAC, and the interface requirements are evaluated in Section 14.3 of this report.

The basis for the staff review has been conformance of the COL applicant's system design and design criteria to the Commission's regulations as set forth in the GDC. The staff concludes that the nonradioactive equipment and floor drain system provides adequate design criteria to ensure compliance with GDC 2, 4, and 60 with respect to protection from natural phenomena, seismic design, environmental conditions, and control of potentially radioactive material, respectively. Therefore, the staff concludes that the design will comply with the guidelines of SRP 9.3.3 and is acceptable.

9.3.4 Chemical and Volume Control System

The ABWR does not include this system.

9.3.5 Standby Liquid Control System

The SLCS was reviewed in accordance with SRP Section 9.3.5 (NUREG-0800). An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed according to the guidelines provided in the "Review Procedures" portion of the SRP. Conformance with the acceptance criteria formed the basis for our evaluation of the SLCS with respect to the applicable regulations of 10 CFR Part 50.

The SLCS is a reactivity control system, which will inject sodium pentaborate solution into the reactor coolant to provide an independent means for shutting down the reactor. The SLCS can bring the reactor from rated power to cold shutdown any time during core life if the normal reactivity control system becomes inoperable. Thus, together with the control rod system, it satisfies the requirements of GDC 26 for reactivity control systems redundancy and capability. (Refer to Section 4.6 of this report for a discussion of reactivity control.)

The system consists of a storage tank, a test tank, and two positive displacement pumps with a motor-operated injection valve at each pump discharge, a motor-operated valve at each pump suction, piping, and controls.

All of the SLCS is located within the secondary containment. The maximum temperature at which the solid material would precipitate from solution is 15 °C (59 °F); the room in which the equipment containing the borated solution is located is kept at a temperature of about 38 °C (100 °F). An electrical resistance heating system maintains the solution between 24 °C and 29 °C (75 °F and 85 °F) to prevent precipitation of the sodium pentaborate solution during storage. Both the high and low liquid level and temperature in the tank are alarmed in the control room.

The two pumps in parallel trains take suction on the storage tank through separate suction lines and discharge it into the reactor vessel through a common injection line.

The liquid is piped into the reactor through the HPCF line downstream of the HPCF inboard check valve. The discharge from each pump is provided with a check valve (to prevent backflow) and a crossover line. Similarly, the piping at the pump suction is also connected by a crossover line. To meet the 10 CFR 50.62 requirements for poison injection, the system is designed for both pumps to start

simultaneously and inject 379 L/m (100 gpm) pentaborate solution.

The ABWR SLCS design includes motor-operated ac valves instead of the squib-activated (explosive) injection valves used in current BWRs. On earlier BWR plants, the SLCS piping was not completely isolated from the SLCS storage tank and it was possible for boron to be present in the SLCS piping. Consequently, GE decided to provide leakproof explosive valves so that boron would not leak into the reactor during SLCS testing. In the ABWR SLCS design, the boron storage tank is provided with normally closed isolation valves and a suction pipe keep fill system to prevent the boron solution from entering the SLCS piping. Because of this design change, GE concluded that the leak-tight explosive valves are not required in the ABWR pump discharge piping.

Each pump and its associated valves are powered from a redundant emergency power supply. They are arranged so that failure of a single pump or valve will not prevent adequate amounts of sodium pentaborate solution from entering the reactor vessel to effect shutdown.

The SLCS is automatically initiated after receiving an anticipated transient without scram (ATWS) signal or can be manually actuated by either of two keylocked spring-return switches in the control room. The SLCS system meets the ATWS rule 10 CFR 50.62 because the SLCS pumps are started automatically (see Section 15.5 of this report). Originally, in the DSER SLCS was started manually, which was identified as an open item (DSER Open Item 15). With the SLCS now modified to be automatically initiated, this open item is resolved.

The ATWS initiation signals for SLCS automatic start are high RPV pressure or low RPV water level 2, and startup range neutron monitor (SRNM) ATWS permissive for 3 minutes. The time delay of 3 minutes is provided to allow completion of electric scram which will take about 2 minutes. When the SLCS is automatically initiated to inject the boron into the reactor, the two injection valves and two storage tank discharge valves are opened, the two injection pumps are started, and both pumps run simultaneously. The CUW isolation valves are also closed automatically to prevent a loss of the sodium pentaborate from the vessel.

Turning either key-locked switch in the control panel switch to the "run" position starts an injection pump, opens one motor operated injection valve on the pump discharge, opens a pump suction valve (which is also the tank outlet valve), and closes the reactor cleanup system isolation

valves to prevent loss or dilution of boron. If the instrumentation provided indicate that the solution is not entering the reactor vessel, the operator can turn the other key-operated switch to the "run" position to actuate the alternate train.

The SLCS is located in a compartment outside the drywell and below the refueling floor in the seismic Category I secondary containment, which protects the SLCS from floods and tornadoes. All portions of the SLCS necessary to inject sodium pentaborate solution into the reactor are seismic Category I, QG B (or QG A if they are part of the reactor coolant pressure boundary). Thus, the SLCS meets the requirements of GDC 2, and the guidelines of RG 1.29, "Seismic Design Classification," Position C.1.

The secondary containment protects the SLCS against externally or internally generated missiles. The SLCS is separated from non seismic system components and from the effects of breaks in other high and moderate-energy piping systems (see Sections 3.5.1.2 and 3.6.1 of this report). Thus, the SLCS meets the requirements of GDC 4.

To ensure the availability of the SLCS, the system includes two parallel sets of components required to actuate the system (pumps and injection valves). The injection portion of the system can be functionally tested by injecting demineralized water from a test tank into the reactor. To ensure the storage tank discharge valves are reliable, the staff requires COL to confirm that the valves will have adequate reliability requirements and that the valves be incorporated into the operational reliability assurance program (ORAP). This issue is a COL Action Item 9.3.5-1. This was previously identified as Confirmatory Item 9.3.5-1 and the Confirmatory item is resolved.

GE submitted the design description and the inspections, tests, analysis, and acceptance criteria (ITAAC) for the SLCS. This was Open Item 9.3.5-1. GE has provided a revised set of descriptions and ITAAC. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Section 14.3 of this SER. On the basis of this evaluation, this item is resolved.

The SLCS meets the acceptance criteria of SRP Section 9.3.5 and concludes that the SLCS meets the requirements of 10 CFR 50.62 and GDC 2, 4, 26, and 27 as they relate to protected against natural phenomena, system function and redundancy, and testability, and the guidelines of Position C.1. of RG 1.29, as related to seismic classification of the system, and is, therefore, acceptable.

Auxiliary Systems

9.3.6 Instrument Air System

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

9.3.7 Service Air System

Because the SA system is one of the four systems that performs functions addressed in SRP Section 9.3.1, the staff reviewed this system as part of an integrated review of the ABWR CA systems. The results of this review are presented in Section 9.3.1 of this report.

9.3.8 Radioactive Drain Transfer System

The staff reviewed the DTS in accordance with SRP Section 9.3.3. Staff acceptance is based on meeting the requirements of GDC 2 as it relates to safety-related portions of the system being capable of withstanding the effects of natural phenomena; GDC 4 as it relates to the capability of the system to withstand the effects of the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents and to be compatible with these conditions; and GDC 60 as it relates to providing a means to control the release of radioactive materials in liquid effluent, including anticipated operational occurrences.

The DTS is designated as non-safety-related (except for containment penetrations and piping in the drywell) and designed to collect radioactive or potentially radioactive effluents in equipment or floor sumps and then transfer the effluents to the LWMS for processing. The DTS system is within the scope of the ABWR design certification. The system includes floor drains, drain lines from the equipment to the sumps, the sumps, the two sump pumps for each sump (each 100-percent capacity), sump instrumentation, and piping and valves from the sumps to the radwaste system.

The drains, piping, pumps, instrumentation, and valves of the DTS system are classified as non-safety-related, except for the containment (drywell) penetrations and containment isolation valves, which are safety Class 2, and designed in accordance with seismic Category I and QG B criteria, and the reactor building penetrations that meet ASME Code III, Section 3, requirements. In DSER (SECY-91-355) Open Item 76, the staff noted a discrepancy in the identification of these containment isolation valves between Figure 11-2 and Table 6.2-7. Figure 11-2 identified these valves as air-operated valves, while Table 6.2-7 showed these valves as motor-operated valves. In SSAR Amendment 20,

Figure 11-2 was modified to indicate that these valves are motor-operated valves. This modification resolved DSER (SECY-91-355) Open Item 76.

All system piping is designed to remain intact following a seismic event. The drain system is not the only method of leak detection available for any of the areas served by the system and this method is not considered in the facility flood analysis. However, the staff indicated in DSER (SECY-91-355) Open Item 77 that the check valves that provide backflow protection for sumps in the ECCS equipment rooms should be classified as safety Class 3 and designed to seismic Category I and QG C criteria. GE committed to revise SSAR Table 3.2-1 and Figure 11-2 to include the designation of these valves as being non-safety designed to seismic Category I and QG C criteria. This information resolved DSER Open Item 77, however, incorporation of the information into the SSAR was identified as DFSER Confirmatory Item 9.3.8-1. The staff reviewed Amendment 26 of the SSAR and concludes that GE modified Section K11 of Table 3.2-1 to require that the check valves be non-safety class and meet seismic Category I and QG C requirements. GE modified Section 9.3.8.1.1 to require that the check valves be inspectable, testable, and able to withstand an SSE. The staff finds that these modifications ensure that the check valves will provide adequate protection from backflow into areas containing safety-related equipment. DFSER Confirmatory Item 9.3.8-1 is resolved.

All drywell sumps are automatically isolated on a LOCA signal to prevent the uncontrolled release of primary coolant outside primary containment.

In the DSER (SECY-91-355), the staff discussed two additional items in its review of the systems and components shown in SSAR Figures 11.2-1 and 11.2-2. First, Figure 11.2-1 showed the shower facility discharging into the high conductivity waste collector tank. In all other figures, and in the text, the shower facility discharges to the hot shower drain (HSD) receiver tank. Second, GE did not identify the points at which changes in component qualification requirements occur (e.g., for the containment isolation valves described in the preceding paragraphs). The staff designated resolution of the discrepancy in Figure 11.2-1 and the addition of component qualification requirements to the figures as Open Item 78.

SSAR Amendment 20 revised the figure to show the shower facility discharging into the HSD receiver tank. In addition, the staff reviewed Amendment 27 of the SSAR and concludes that GE clarified the description of the containment (drywell) sump isolation valves. The valves meet seismic Category I, safety Class 2, and QG B, requirements.

The staff noted in the DSER (SECY-91-355) that a GE interface requirement in the SSAR prevents connections between radioactive and nonradioactive systems. Upon further evaluation, the staff determined that this requirement can be accomplished by identifying a COL Action Item requiring this information. This was identified as DFSER COL Action Item 9.3.8-1. GE has included this information in the SSAR. This is acceptable.

IE Bulletin No. 80-10 identified an issue concerning the potential contamination of nonradioactive systems which could result in unmonitored, uncontrolled releases of radioactivity to the environment. The ABWR design employs several methods to prevent such an occurrence. The methods include ensuring that no cross-connections exist between nonradioactive and potentially radioactive systems (such as with the DTS system), by providing sampling and monitoring of both radioactive and nonradioactive effluents before discharge, and by separating nonradioactive systems from potentially radioactive systems by barriers, along with radiation monitors to detect leakage across the barrier (such as with the RCW system). These methods are discussed in detail in various SSAR sections (e.g., 9.2.11 and 11.5) and are reviewed in various sections of this report (e.g., 9.2.11 and 11.5). These methods ensure that all discharges to the environment are monitored and controlled. Based on this information, the staff concludes that the ABWR design has provided sufficient design features and design requirements to ensure that the issues discussed in IE Bulletin No. 80-10 are resolved.

In the DSER (SECY-91-355), the staff discussed a GE interface requirement for monitoring the effluent from nonradioactive systems before discharge to ensure that no unacceptable (radioactive) effluents are discharged from the nonradioactive drain systems. This requirement for monitoring nonradioactive effluents will allow the COL applicant to design a NRD system that will satisfy the requirements of GDC 60 (the nonradioactive drain system is evaluated in Section 9.3.3 of this report). Upon further evaluation, the staff has determined that this requirement can be accomplished by establishing a COL Action Item requiring this information. This was identified as DFSER COL Action Item 9.3.8-2. GE included this information in the SSAR, which is acceptable.

The staff stated in the DSER (SECY-91-355) that SSAR Section 9.3.8.2 inaccurately referred to SSAR Section 9.3.9.1, when the interface requirements discussed in SSAR Section 9.3.12 would be the appropriate reference. The staff stated that GE should revise the first design basis discussed in SSAR Section 9.3.8.1 to clearly indicate, consistent with the staff's dialogue with GE, that only portions of the drain system are considered safety-

related. GE corrected the SSAR reference and revised SSAR Section 9.3.8.1, which resolved DSER Open Item 79. Incorporation of the revisions in the SSAR was identified as DFSER Confirmatory Item 9.3.8-2. The staff reviewed Amendment 29 of the SSAR and determined that the required corrections were made. The staff determined that the classifications in Section K1 of Table 3.2-1 clarify which parts of the system are safety-related and which are not. The staff has determined that these classifications will ensure that the check valves will provide adequate backflow protection. Therefore, the staff finds these modifications acceptable. Therefore, DFSER Confirmatory Item 9.3.8-2 is resolved.

GE submitted the design description and the ITAAC relating to the DTS system. This was identified in the DFSER as Open Item 9.3.8-1. GE provided a revised set of design description and ITAAC. The adequacy and acceptability of the design description and the ITAAC are evaluated in Section 14.3 of this report. On the basis of this evaluation, DFSER Open Item 9.3.8-1 is resolved.

The staff concludes that the design of the RD system is acceptable and conforms to the requirements of GDC 2, 4, and 60 for protection against natural phenomena, environmental conditions, missiles, and the release of radioactivity to the environment. Therefore, the staff finds that the design complies with guidelines of SRP Section 9.3.3, and is acceptable.

9.3.9 Hydrogen Water Chemistry System

Hydrogen water chemistry (HWC) reduces intergranular stress corrosion cracking (IGSCC) by using feedwater additions of hydrogen to decrease the oxidizing power of water and reduce its aggressiveness toward plant material. To suppress IGSCC, reactor coolant conductivity must be maintained below 0.3 micro-Siemens per centimeter and sufficient hydrogen must be added to the feedwater to reduce the electro-chemical potential below -0.23 volts (Standard Hydrogen Electrode). These conditions are specified in EPRI Report NP-4947-SR, "BWR Hydrogen Water Chemistry Guidelines: 1987 Revision," October 1988. SSAR Section 9.3.9 references this report and commits to use its guidelines for design and operation of the HWC system.

Section 9.3.9 of the SSAR also addresses the means of storing and handling of hydrogen. These operations will be performed in accordance with the recommendations of EPRI Report NP-5283-SR-A, "Guidelines for Permanent BWR Hydrogen Water Chemistry Installations."

The staff finds that the EPRI guidelines presented in these reports describe a satisfactory means for storing and

Auxiliary Systems

handling hydrogen for the ABWR, and the GE commitments in the SSAR to the EPRI guidelines are thus acceptable. The staff concludes that DSER (SECY-91-355) Open Item 100 is resolved.

Certified Design Material

GE has proposed ITAAC and included Tier 1 design description of the hydrogen water chemistry system in Section 2.11.17 of the CDM. This was identified as DFSER Open Item 9.3.9-1. The adequacy and acceptability of GE's design descriptions and ITAAC are evaluated in Section 14.3 of this report. On the basis of this evaluation, DFSER Open Item 9.3.9-1 is resolved.

9.3.10 Oxygen Injection System

The oxygen injection system is designed to add sufficient oxygen (20 to 50 ppb) to suppress erosion/corrosion, general corrosion, and the release of corrosion products in the condensate and feedwater systems. The requirements for design, operation, maintenance, surveillance, and testing of the oxygen storage facility are specified in EPRI Report NP-5283-SR-A. The staff finds that the EPRI guidelines describe a satisfactory means for storing and handling oxygen for the ABWR, and are thus acceptable. Section 9.3.10 of the SSAR references this report and commits to use these guidelines. The staff concludes that DSER (SECY-91-355) Open Item 101 is resolved.

Certified Design Material

GE had not proposed ITAAC nor a Tier 1 design description of the oxygen injection system in Section 2.11.7 of the CDM. This was DFSER Open Item 9.3.10-1.

The adequacy and acceptability of GE's design descriptions and ITAAC are evaluated in Section 14.3 of this report. On the basis of this evaluation, DFSER Open Item 9.3.10-1 is resolved.

9.3.11 Zinc Injection System

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

Laboratory tests confirmed that Co-60 deposition is greatly reduced in both normal and hydrogen water chemistry with the presence of soluble zinc. Zinc injection into the feedwater system to provide reactor water concentrations of 10 to 15 ppb zinc during initial conditioning and 5 to 10 ppb over the fuel cycle will help keep radiation levels as

low as possible; thereby, reducing personnel exposure especially during outages.

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

Certified Design Material

At the time of the DFSER, GE had not proposed ITAAC nor a Tier 1 design description of the zinc injection system in Section 2.11.18 of the CDM. This was DFSER Open Item 9.3.11-1. GE has provided a final version of the CDM and it has been evaluated in Chapter 14.3 of this report. Based on this evaluation, DFSER Open Item 9.3.11-1 is resolved.

9.4 Heating, Ventilation, and Air Conditioning Systems

Certified Design Material

GE submitted in Sections 2.15.5 of the CDM Tier 1 design description and the ITAAC for HVAC systems which were under staff review. The results of the staff's review were to be provided in the FSER. This was DFSER Open Item 9.4-1. GE provided a revised set of design description and ITAAC on August 31, 1993. The adequacy and acceptability of the design description and the ITAAC are evaluated in Section 14.3 of this report. On the basis of this evaluation, this open item is resolved.

9.4.1 Control Building Heating, Ventilating, and Air Conditioning System

The staff reviewed the control building HVAC systems in accordance with SRP Section 9.4.1. The system consists of two separate HVAC systems: one for the MCR habitability area and another for the control building safety-related equipment area. The staff evaluated these systems as described in Sections 9.4.1.1 and 9.4.1.2, respectively, of this report, and are evaluated for compliance with the SRP acceptance criteria as discussed below.

9.4.1.1 Control Room Habitability Area Heating, Ventilating, and Air Conditioning System (CRHA HVACS)

The CRHA HVAC system serves the main control area envelope (MCAE) containing the control room proper (including the critical document file), computer room, control equipment room, upper and lower corridors, office and chart room, instrument repair room, and sleeping area, as described in SSAR Section 6.4.2.1. The MCAE is maintained at a minimum positive pressure of 3.2 mm

(1/8 in.) of water gauge relative to the surrounding spaces with outside makeup air of not more than 360 m³/hr (212 cfm) @ 760 mm Hg (30 in.), 0 °C (32 °F). The kitchen and lunch rooms, men's lavatory, and women's lavatory and lounge facilities are in the Service Building.

The system consists of two fully redundant trains of equipment, including ductwork, air handling units (AHUs), control dampers, fire zone dampers, and supply and exhaust fans. Each AHU consists of a bag-type filter, a chilled-water cooling coil, an electric heating coil, and a humidifier as described in SSAR Section 9.4.1.1.3. Each train also incorporates an emergency filtration unit (EFU) which is an ESF. The EFU is comprised of an electric heating coil, pre-filter, pre-high efficiency particulate air (HEPA) filter, charcoal adsorber (50 mm (2 in.) deep, as a minimum), post-HEPA filter, and two circulating fans. Independent and separate discharge to and return from the MCAE is provided to each EFU. Chilled water to the cooling coils is supplied by the HVAC emergency cooling water (HECW) system. The CRHA HVAC system is designed to maintain a controlled temperature environment under normal and accident conditions, and provides for detection and removal of smoke and filtration of radioactive material. The range of design conditions for the MCAE is 21 °C through 26 °C (70 °F - 79 °F) and 10 percent to 60 percent relative humidity. SSAR Section 6.4.4.2 discusses the protection from exterior smoke, toxic chemicals, and chlorine releases.

Normally, one AHU, one supply CRHA HVAC system fan, and one exhaust fan are in operation. During normal operation, the CRHA HVAC system performs HVAC functions and pressurizes the MCAE using a combination of filtered outdoor air and recirculated indoor air. The combined air stream is passed through an AHU. Two parallel, 100-percent capacity supply fans (one standby) draw air from the instrument panel areas, corridors, MCR, computer room, office areas, and the switch and tag room, and return it to the AHU. The exhaust fan starts automatically when the supply fan starts. Two parallel, 100-percent exhaust fans (one standby), controlled by a pressure controller, draw air from the areas and exhaust it to the environment to maintain positive pressure of 3.2 mm (1/8 in.) of water gauge relative to the surrounding spaces. The pressure controller is located in the instrument panel area of the MCR. The supply and return ducts have manual balancing dampers that are locked in place after the system is balanced. The supply and return ducts also have modulating dampers to maintain the required positive pressure. Sufficient air is provided to pressurize the control room equipment HVAC envelope. As described in SSAR Section 6.4.2.1, the control room area envelope is maintained at a positive pressure of 3.2 mm through 6.4 mm (1/8 in. through 1/4 in.) water gauge with respect

to the surrounding spaces at all times. The CRHA HVAC system flow diagram is shown in SSAR Figure 9.4-1, Sheets 1 and 2, flow rates are given in SSAR Table 9.4-3, and component descriptions are given in SSAR Tables 9.4-4 and 9.4-4a through 9.4-4c.

GE submitted a draft of a revised SSAR Section 9.4.1.1.4 which addresses IE Bulletin 80-03 compliance by stating that the charcoal tray and screen will be of all welded construction to preclude the potential loss of charcoal from adsorber cells per this bulletin. Therefore, the emergency air filtration system of the CRHA HVAC system precludes the potential loss of charcoal from adsorber cells as reflected in SSAR Amendment 34.

The CRHA HVAC system and components are located in a seismic Category I control building that is protected against tornado missiles and floods and are operable during LOOP. All essential control room HVAC equipment, including ductwork, is of seismic Category I design. SSAR Table 3.2-1 states that those nonessential portions of the system which, by their failure during a seismic event can affect safety-related portions, of the system are designed in accordance with Position C.2 of RG 1.29. Outside air intake valves are protected against freezing and other environmental conditions, and tornado missile barriers protect the intake vents. The CRHA HVAC system meets the requirements of GDC 2 by complying with the provisions of Position C.1 of RG 1.29 for the essential portions, and Position C.2 for the applicable nonessential portions.

SSAR Figures 1.2-15 and 1.2-21 show that the redundant CRHA HVAC system trains are located in separate rooms. In SSAR Amendment 7, GE stated that the walls, floor, and ceiling of the room that houses each safety-related system act as missile barriers or shields from missiles generated outside the room. In SSAR Amendment 17, GE stated that non-safety-related components are arranged in such a way that any missile-generating component is in a separate room away from safety-related components. For these reasons, the staff concludes that the redundant CRHA HVAC system trains are protected against internally-generated missiles. The CRHA HVAC system complies with GDC 4 with respect to protection of the system against environmental and dynamic effects, as discussed in Section 3.6.1 of this report regarding the effect of postulated piping failures outside the containment on the system. Therefore, the first part of DSER Open Item 60 which deals with staff's concern relating to compliance of this system with the requirements of GDC 4, is resolved.

Smoke detectors in the MCAE will actuate an alarm on detection of smoke when smoke is detected in a division of the CRHA HVAC system, MCR operators will manually

Auxiliary Systems

switch the appropriate divisional HVAC system to a smoke removal mode, in which the exhaust fan is stopped, the recirculation damper is closed, and the exhaust bypass damper is opened. The MCR operators can exhaust 100 percent of the conditioned air to the atmosphere by manually activating a switch, which closes return dampers and opens exhaust dampers. In SSAR Amendment 32, GE stated that the CRHA HVAC system fire dampers are equipped with fusible links and are capable of closing under anticipated air flow conditions after the fusible link melts. Tests will be performed to verify these conditions at a test facility.

In the DSER (SECY-91-235), the staff stated the system design meets the control room habitability requirements of TMI Action Item III.D.3.4 of NUREG-0737 with regard to smoke removal, contingent upon GE submitting revised P&IDs showing smoke detection capability (second part of DSER Open Item 60). The staff stated the above contingency, since the earlier P&IDs for the control room and the control equipment room did not include smoke detectors at the air intakes. However, GE committed to place smoke detectors at the air intakes which will cause the control room HVAC system to shift to the recirculation mode, if smoke is detected in the air supply. Since this commitment resolved the second part of DSER Open Item 60, the staff classified the SSAR commitment as DFSEER Confirmatory Item 9.4.1.1-1. SSAR amendments through Amendment 32 included revised P&IDs which show the smoke detectors at the air intakes. Therefore, this confirmatory item is resolved.

The evaluation of requirements for protection against hazardous chemical releases will be site specific as stated in Section 6.4 of this report. Therefore, the staff stated in the DFSEER that COL applicants should provide information regarding compliance with RG 1.78, "Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release," and RG 1.95, "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release," if the potential for hazardous chemical, including chlorine, release exists at their sites. The staff further stated in the DFSEER that COL applicants should identify plant design features that will automatically isolate the control room from the outside when necessary, and thus protect the control room operators from toxic substances in the plant vicinity. The need for providing the above information was identified as DFSEER COL Action Item 9.4.1.1-1. By SSAR amendments up to and including Amendment 32, GE identified COL license information (Section 6.4.7.3) which states that the design acceptance is based upon meeting the guidelines of the subject guides.

A four-channel radiation monitoring system is provided in the outside air intake ducts. A radiation monitor is provided in the MCR to monitor MCR area radiation levels. These monitors alarm in the MCR upon detection of high radiation levels. On receiving a high radiation signal, the redundant normal outside air intake dampers and exhaust dampers close, the emergency outside makeup air intake dampers open, and only the corresponding EFU of the operating division starts automatically. Also, the emergency recirculation mode can be started manually. In the emergency recirculation mode, the makeup air and part of the return air pass through the ESF-grade filter train, which filters the radioactive particles and iodine to keep the control room operator exposures below the GDC 19 limit of 5 rem whole body or its equivalent for the duration of a design-basis accident (DBA). The control room equipment HVAC system thus meets the requirements of GDC 19 regarding protection of the control room. For further information on system compliance with GDC 19 limits, see Section 6.4 of this report. Radioactivity monitors at the inlets allow operator selection of inlet vent manually. Connections are provided for testing the HEPA and charcoal filters pressure drops and radioiodine removal efficiency. A differential pressure switch across the emergency filtration system alarms on high pressure. A flow switch in the emergency filtration fan discharge duct initiates an alarm if the fan fails and automatically starts the standby system.

After the May 5, 1992 meeting between the NRC and GE, in a telephone conversation on May 12, 1992, GE committed to provide SSAR appendices identifying the compliance of (1) the ESF-grade filter train in the system with RG 1.52 guidelines and (2) control room filter train instrumentation with the minimum instrumentation requirements listed in SRP Table 6.5.1-1. These appendices were to be similar to those provided for the SGTS. GE provided the information in the SSAR, Appendices 9C and 9D. The staff reviewed GE's justifications for compliance with RG 1.52 and NUREG-0800, Standard Review Plan, Table 6.5.1-1, "Minimum Instrumentation, Readout, Recording and Alarm, Provisions for ESF Atmosphere Cleanup Systems," for the CRHA HVAC system and found that the proposed RG 1.52 conformance and SRP Table 6.5.1-1 instrumentation and controls in the MCR, as well as selected local panels, meet the intent of the above guidance. While the staff would prefer that GE provide all of the local instrumentation and controls described in the above guidance for operational efficiency and convenience, the staff finds that the CRHA HVAC system instrumentations and controls meet the intent of the above guidance based on the following:

1. The instrumentation and controls (i.e., readout, recordings, and alarms) provisions for the related

parameters (i.e., pressure drop, temperature, humidity, flow) are available and would be made accessible in the MCR.

2. The COL applicant will prepare plant-specific procedures to respond to the instrumentation and controls.
3. The filtration train fully complies with the regulatory position of RG 1.52.
4. The acceptance, pre-operational, and surveillance testing meet the intent of ASME Codes N509 and N510.
5. GE revised SSAR Sections 9.4.1.1.4 and 9.1.1.1.5, stating that the unfiltered in-leakage is controlled by the use of welded ducts, except galvanized steel is used for outdoor air intake and exhaust ducts, and unfiltered in-leakage testing will be performed periodically on all system ductwork outside MCAE, in accordance with ASME Code N510.
6. CRHA HVAC system dose analysis is bounded by the minimum outside air intake flow, and any higher flow will be filtered through the EFUs.

Based on the above, the staff has determined that the CRHA HVAC system's EFUs will have a creditable removal efficiency of 95 percent for all forms of radio-iodine.

In DFSER Open Item 9.4.1.1-1, the staff stated that the SSAR did not specify heaters to maintain proper humidity (less than 70 percent) to ensure proper operation of the charcoal adsorbers. GE revised SSAR Section 9.4.1.1 and Table 9.4-4 to include electric heaters in the ESF filter trains. Therefore, this item is resolved.

Based on the above, the staff concludes that the control room habitability area HVAC system complies with GDC 2, 4, 19, and 60 as they relate to protection against natural phenomena, maintaining proper environmental limits for equipment operation, and protecting those who access the control room under accidental radioactive releases. This system also complies with the guidelines of RGs 1.29 and 1.52 relating to the system seismic classification and system capability to clean the control room atmosphere during emergency operation, and thereby meets the applicable acceptance criteria of SRP Section 9.4.1. Therefore, the staff concludes that the system is acceptable.

9.4.1.2 Control Building Safety-Related Equipment Area (CBSREA) HVAC System

The ABWR design includes three similar, but independent, HVAC systems to service the safety-related electrical, HVAC, and mechanical equipment (including the refrigerator, pump, and heat exchanger equipment) of Divisions A, B, and C of the CBSREA HVAC system. Safety-related areas serviced by each system include a safety-related battery room(s), HECW refrigerator room, reactor building cooling water pump and heat exchanger room, HVAC equipment room, and safety-related electrical equipment room. The systems differ with regard to safety-related areas serviced, in that the system serving Division B includes two safety-related battery rooms, while the systems serving Divisions A and C include one safety-related battery room each. Each divisional system serves non-safety-related passages. In addition, Division A serves a non-essential battery room, and a non-essential electrical equipment room, and Division C serves the non-safety-related MG sets. Each divisional system draws outside air through a normal air supply unit, and system exhaust fans discharge to the atmosphere. Chilled water to the cooling coil is supplied by the emergency cooling water (HECW) system. Each supply unit consists of a bag filter and cooling coil. Each divisional system contains an air supply unit with two 100-percent capacity supply fans and exhaust fans. The supply and exhaust fans of each system (Divisions A, B, and C) are powered from the corresponding Class 1E Electrical Divisions I, II, or III. One of the two supply fans is started manually for normal operation. When the ACU supply fan is started, one of the two exhaust fans start automatically. The exhaust fans can be started manually to exhaust the air to the atmosphere as needed such as to remove hydrogen and smoke.

In normal operating mode, one ACU, one supply fan, and one exhaust fan in each division operate. The HVAC system includes fire dampers for the supply and exhaust duct penetrations of firewalls for Division IV safety-related electrical equipment room and battery room, served by Division B of CBSREA HVAC system. The smoke removal mode is manually initiated by closing the recirculation damper, closing the exhaust fan, and opening the exhaust fan bypass damper to allow outside air purging of the affected area. The smoke removal mode of the HVAC systems is discussed in CBSREA Section 9.5.1 of this report. SSAR Amendment 32 stated that the CBSREA HVAC system fire dampers separating safety-related electrical equipment and battery Divisions II and IV rooms use fusible links in HVAC ductwork which will close under anticipated air flow conditions after the fusible link melts, and that tests will be performed to verify these conditions at a test facility.

Auxiliary Systems

The HVAC subsystems and components are located in a seismic Category I structure that is tornado missile-protected and flood-protected. The essential components of each system are of seismic Category I design. The low-pressure safety-grade ESF ductwork is designed to withstand the maximum positive pressure to which it can be subjected under normal or abnormal conditions. The outdoor air intake and exhaust are conducted of galvanized steel, and all other ducts are of welded black steel construction. Tornado missile barriers protect the air intake and exhaust vents. Applicable nonessential portions of the system which, by their failure during a seismic event can effect safety-related portions, are designed in accordance with Position C.2 of RG 1.29, and as stated in the SSAR Table 3.2-1. The systems meet the requirements of GDC 2 regarding protection from natural phenomena by complying with the provisions of Positions C.1 of RG 1.29 for the essential portions and Position C.2 for the applicable nonessential portions.

Each system is designed to maintain hydrogen concentration below 2 percent by volume in the battery rooms, and temperature controllers regulate heating and cooling to maintain conditions within pre-set limits. SSAR Figures 1.2-15 and 1.2-21 show that the divisional CBSREA HVAC systems are located in separate rooms. SSAR Amendment 7 stated that the walls, floor, and ceiling of the room that houses each safety-related system act as missile barriers or shields from missiles generated outside the room. SSAR Amendment 17 stated that non-safety-related components are arranged in such a way that any missile-generating component is in a separate room away from safety-related components. For these reasons, the staff concludes that the redundant essential electrical equipment HVAC systems are protected against internally-generated missiles. The CBSREA HVAC systems comply with GDC 4 with respect to protection of the system against environmental and dynamic effects (see Section 3.6.1 of this report for an evaluation of the effect of postulated piping failures outside the containment on the system).

In DSER (SECY-91-235) Open Item 61, the staff stated that the RCW pump rooms are located in the secondary containment and that the design of the HVAC system as constituted did not account for the potential for radioactivity contamination following an accident. The staff re-examined this issue and finds that SSAR Figure 1.2-15 shows that the subject rooms serviced by the HVAC system are in the control building and are unlikely to be contaminated to the extent that the exhaust from the subject areas has to be filtered. The staff further discusses this in Section 11.5 of this report and concludes that no monitoring of this exhaust is required. Therefore, this resolved DSER Open Item 61.

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

The staff concludes that the system complies with GDC 2 and 4 as they relate to protection against natural phenomena and maintaining proper environmental limits for equipment operation, and with the guidelines of RG 1.29 relating to the system seismic classification, and thereby meets the acceptance criteria of SRP Section 9.4.1. Therefore, the staff concludes that the CBSREA HVAC system is design acceptable.

9.4.2 Spent Fuel Pool Area Ventilation System

The spent fuel pool area ventilation system is part of the reactor building ventilation system and is evaluated in Section 9.4.5 of this report.

9.4.3 Auxiliary Area Ventilation System

The auxiliary area ventilation system is part of the reactor building ventilation system and is evaluated in Section 9.4.5 of this report.

9.4.4 Turbine Island HVAC System

The staff reviewed the turbine island ventilation system in accordance with SRP Section 9.4.4. The system consists of the turbine building (TB) HVAC system and the electrical building (EB) HVAC system. The TB HVAC and EB HVAC systems supply filtered and tempered air to all turbine island spaces during normal, plant startup, and shutdown operations. The TB HVAC and EB HVAC systems do not serve or support any safety function and have no safety design basis. The system description, equipment list, and figures are given in SSAR Section 9.4.4, Tables 9.4-5 and 9.4-5a through 9.4-5c, and Figures 9.4-2a through 2c. Failure of the system does not compromise the operation of essential systems, the ability to shut down the reactor, or result in unacceptable releases of radioactivity. Both the TB HVAC and EB HVAC systems comply with GDC 2 because the applicable nonessential portions of the systems are designed in accordance with Position C.2 of RG 1.29 as stated in SSAR Table 3.2-1. The requirements of GDC 60 for control of releases of radioactive materials to the environment do not apply because the ventilation exhausts of these systems do not include HEPA filters or charcoal adsorbers. The TBVS exhausts, including turbine building compartment exhaust and lube oil areas exhaust (SSAR Figures 9.4-2a through and 2c), will be monitored at the plant vent. The EB HVAC system will exhaust directly to the atmosphere (for additional information on

effluent monitoring, see Section 11.5 of this report). The staff concludes that the turbine island HVAC system design complies with the applicable GDC referenced in SRP Section 9.4.4 and consequently with the subject SRP criteria and is, therefore, acceptable.

9.4.5 Reactor Building Ventilation System

The reactor building ventilation system (RBVS) consists of eight subsystems: (1) reactor building (RB) secondary containment HVAC system, (2) RB safety-related equipment HVAC system, (3) RB non-safety-related equipment HVAC system, (4) RB safety-related electrical equipment HVAC system, (5) RB safety-related diesel generator HVAC system, (6) RB primary containment supply exhaust system, (7) RB main steam tunnel (MST) HVAC system, and (8) RB reactor internal pump (RIP) power supply panel adjustable speed drive (ASD) HVAC system.

The RBVS cools all areas in the reactor building serviced by these subsystems except the area serviced by the RB primary containment supply/exhaust system (cooling for this area is evaluated in Section 9.4.9 of this report). Certain subsystems including the RB secondary containment HVAC system, RB safety-related electrical equipment HVAC system, RB safety-related diesel generator HVAC system, and the primary containment supply/exhaust system have outside air intakes and exhaust air from the areas serviced by these subsystems. The HVAC systems are reviewed in accordance with SRP Section 9.4.5, as applicable, for ESF ventilation systems. The staff reviewed the RB safety-related equipment HVAC system in accordance with SRP Section 9.4.2, the subject HVAC system also services the spent fuel pool area.

9.4.5.1 RB Secondary Containment HVAC System

In the DSER (SECY-91-235), the staff included only a partial evaluation of the system, stating that the evaluation would be completed on receipt of the system P&IDs and interface descriptions. The staff designated this lack of information as Open Item 62 in the DSER (SECY-91-235). SSAR amendments through Amendment 34 included the above information in revised SSAR Section 9.4.5.1, Figure 9.4-3, and Tables 9.4-3, 9.4-4f through 9.4-4h, 11.5-1, and 11.5-2. Therefore, the DSER Open Item 62 is resolved based on the staff's evaluation given below. SSAR Section 9.5.1.1.6 describes the smoke control mode of the secondary containment HVAC system. In Section 9.5.1.4.4 of this report, the staff evaluated the HVAC system design for smoke control in the secondary containment.

The RB secondary containment HVAC system provides the HVAC needs of the secondary containment area of the reactor building. The system provides the HVAC needs of the safety-related equipment areas in the secondary containment, during normal operation. The system also provides the HVAC needs of the non-safety-related equipment areas in the secondary containment in conjunction with the RB safety-related equipment HVAC system during normal operation. The system is a once-through type. Outdoor air is filtered (bag-type filter), tempered, and delivered to the secondary containment areas. The exhaust air from all the areas serviced by the system is filtered (bag-type) and discharged to the monitored plant vent. The supply system consists of a bag-type filter, a cooling/heating coil, and three 50-percent supply fans, two of which operate normally (one is standby). The exhaust system also has three fans (one standby), each of which is provided with a bag-type filter. The system includes radiation monitors for air exhaust through the refueling floor area ducting, and the common secondary containment area ducting which provide input signals to process radiation monitoring system. Isolation dampers automatically isolate the system and the SGTS is initiated on (1) secondary containment exhaust high radiation signal, (2) refueling floor high radiation signal, (3) LOCA signal, or (4) loss of secondary containment HVAC supply/exhaust fans signal (for SGTS evaluation, see Section 6.5.1 of this report). The secondary containment is maintained at a negative pressure with respect to the surrounding spaces during normal operation by the system and during accident conditions by the safety-related SGTS. The supply fans will operate only when the exhaust fans are operating. Fire dampers with fusible links in the HVAC duct work will close under anticipated air flow conditions after the fusible links melt. The system is started manually.

The secondary containment HVAC contains no equipment needed to maintain the integrity of the reactor coolant pressure boundary or to ensure the safe shutdown of the reactor. The interface of the secondary containment HVAC system with the SGTS, which provides radioactivity filtering capability for containment releases, does include components (e.g., isolation devices and monitors) whose function is required to maintain releases below acceptable levels.

The system is located in the seismic Category I reactor building, which protects against flood and tornado missiles. Although the system is non-safety-related, SSAR Table 3.2-1 states that the radiation monitors and the inboard and outboard dampers for isolating the system under situations identified above are designed to seismic Category I requirements and that the applicable nonessential portions are designed in accordance with

Auxiliary Systems

Position C.2 of RG 1.29. Thus, the system complies with GDC 2 by meeting Position C.1 of RG 1.29 for the essential portions and Position C.2 for the nonessential portions. The inboard and outboard isolation dampers are pneumatically operated and they fail to the closed position in the event of loss of pneumatic pressure or loss of power to the actuating solenoids. The safety-related components are protected against damage from internally-generated missiles by separation of redundant equipment and are also protected against piping failures. Thus, the essential components of the system comply with GDC 4 regarding protection from the environment and dynamic effects associated with equipment failures. GDC 17 is not applicable to the system because the system is not safety related. However, because the secondary containment HVAC system air intake is filtered, the system meets GDC 17 requirements as it relates to protection of the electrical components against accumulation of dust and particulates.

The system complies with GDC 60 regarding the control of releases of radioactive materials to the environment because under certain situations, the system is automatically isolated to prevent significant radioactivity release to the environment. When isolated, the system discharges to the SGTS, which will filter any release to the environment.

Based on the above, the staff concludes that the system design complies with applicable GDC referenced in SRP Section 9.4.5 and consequently with the subject SRP acceptance criteria, and, therefore, is acceptable.

9.4.5.2 RB Safety-Related Equipment HVAC System

The system cools the following safety-related equipment areas in the reactor building during accident conditions: the RHR pump rooms, HPCF pump rooms, RCIC pump room, FCS equipment rooms, SGTS equipment rooms, and the CAMS equipment rooms. As stated in Section 9.4.5.1 of this report, the RB secondary equipment HVAC system cools the above areas under normal conditions. Increased and reliable cooling of these rooms will be required during accident conditions when the equipment in the rooms operate. The system is designed to provide a controlled temperature environment in the rooms in accordance with equipment temperature qualification. Separate safety-related fan coil units (FCUs), one for each room serve the 12 rooms that house the redundant components of the five systems mentioned above and the RCIC pump room. The pump room FCUs are automatically initiated when RHR pumps, HPCF pumps, and RCIC turbines are started. The safety-related FCUs in other rooms are automatically initiated upon isolation of the RB secondary containment

HVAC system. The FCS room FCUs are initiated upon receiving a manual FCS start signal.

All components of the RB safety-related equipment HVAC system are designed to ESF requirements and are safety-related. The units are open-ended and continuously recirculate cooling air within the space served. Divisional RCW is the cooling medium for the associated cooling coils. Space temperatures in the pump rooms are maintained less than 66 °C (150 °F) during pump operation. Space temperatures in all rooms, including the pump rooms are normally maintained less than 40 °C (104 °F). The units will receive power from the same divisional power source as that for the equipment being served. The design of the FCUs includes both remote and local manual override of the system controls. The system description, figure, and tables are listed in SSAR Section 9.4.5.2, Figure 9.4-3, and Table 9.4-4e, respectively.

All components of the RB safety-related equipment HVAC system (including associated cables) serving safety-related functions are located in seismic Category I structures that protect against flood and tornado missiles. The components of this safety-related system are designed to seismic Category I requirements as follows from SSAR Table 3.2-1. Thus, the system meets Position C.1 of RG 1.29, and consequently complies with GDC 2 regarding protection from natural phenomena.

In the DSER (SECY-91-235), the staff included only a partial evaluation of the system for compliance with GDC 4 as it relates to protection of the system against internally-generated missiles and piping failures. The staff stated that it would complete the evaluation upon receiving system P&IDs and clarifying information on system compliance with GDC 4. The DSER (SECY-91-235) designated the above lack of information as Open Item 63. SSAR amendments through Amendment 34 included the required information, which the staff reviewed. The staff finds that the design satisfies the requirements of GDC 4 for protection against the effects of piping failures and internally-generated missiles by compartmentalization and separation. (For additional information, see Section 3.6.1 of this report, SSAR Section 3.6.4, and SSAR Table 3.6-2.) This resolved DSER Open Item 63.

This internally recirculating cooling system has no supply air from the outside or exhaust air from the equipment areas to the outside when the system operates. The system operates only after the secondary containment HVAC system is isolated. The normal RB secondary containment HVAC system air intake is filtered. Therefore, the system meets GDC 17 as it relates to protection of the electrical components of the system against accumulation of dust and particulates. The system does not include any HEPA

filters or charcoal adsorbers. Therefore, GDC 60, which pertains to the control of radioactive material released to the outside, does not apply to the system. Position C.4 of RG 1.13, (an acceptance criterion for SRP Section 9.4.2), which deals with the ventilation and filtration basis for the release of radioactive materials to the environment from the spent fuel storage facility, does not apply to that part of the essential equipment HVAC system that services the fuel pool cooling equipment rooms. The SGTS that serves the fuel pool area during a DBA, including the fuel-handling accident, meets Position C.4 of RG 1.13 and, therefore, complies with GDC 61. For further information, see Section 6.5.1 of this report.

The staff concludes that the system complies with applicable GDC referenced in SRP Section 9.4.5 and consequently with the subject SRP acceptance criteria. The part of the system that services the spent fuel pool area complies with the applicable GDC (GDC 2 and 4) referenced in SRP Section 9.4.2, and consequently with the subject SRP acceptance criteria. Therefore, the staff concludes that the RB safety-related equipment HVAC system design is acceptable.

9.4.5.3 RB Non-Safety-Related Equipment HVAC System, RB Mainstream Tunnel HVAC System, and RB RIP Adjustable Speed Drive (ASD) HVAC System

DSER (SECY-91-235) Sections 9.4.5.3, 9.4.5.7 and 9.4.5.8 are consolidated in this section as part of this report.

The RB non-safety-related equipment HVAC system consists of six FCUs and four AHUs, each consisting of a cooling coil and a fan. Each fan coil unit cools the associated room during normal conditions in conjunction with the RB secondary containment ventilation system. The 10 individual rooms are identified in SSAR Section 9.4.5.3.2. The units are open-ended and continuously recirculate cooling air within the space served. This system is manually started. The system description, figure, and component description are given in ABWR SSAR Section 9.4.5.3, Figure 9.4-3, and Tables 9.4-4i, respectively.

The RB main steam tunnel HVAC system consists of two closed-loop fan-coil recirculation units that remove heat from the tunnel area. Each fan-coil unit consists of a cooling coil and two fans, one of which operates while the other is in standby. A flow switch in the operating fan discharge ductwork automatically starts the standby fan on

receiving indication of operating fan failure. The system description, figure, and component description are given in SSAR Section 9.4.5.7, Figure 9.4-3, and Tables 9.4-3, 9.4-4f, 9.4-4g, and 9.4-4i, respectively.

The Divisions I and II for the RB RIP ASD HVAC system are identical. Each division consists of two supply fans (one standby), one cooling coil for each fan, and a closed-loop system that cools the associated power panel. The divisions are started manually from the control room, and an air flow failure sensed by the flow switch automatically starts the standby fan and activates an alarm in the control room to indicate the fan failure. The system description, figure, and component description are given in SSAR Section 9.4.5.8, Figure 9.4-5, and Tables 9.4-3 and 9.4-4g, respectively.

The three separate HVAC systems are not safety-related and are not required to maintain reactor coolant pressure boundary or to achieve and maintain safe-shutdown conditions. These HVAC systems include features that provide reliability over the full range of normal plant operation such as redundant components for the RB main steam tunnel HVAC system and the RB RIP power supply panel room HVAC system. HVAC normal cooling water is the cooling medium for these systems. These three systems comply with GDC 2 regarding protection from natural phenomena because the applicable nonessential portions of the system are designed in accordance with Position C.2 of RG 1.29 as stated in SSAR Table 3.2-1. The requirements of GDC 60 regarding control of releases of radioactive materials to the environment are not applicable because these systems are recirculation type systems and do not contain any HEPA filters or charcoal adsorbers. Further, because the systems do not perform any safety function and are not used during an accident, GDC 4 and 17 are not applicable.

GE provided a set of design description and ITAAC relating to the RB non-safety-related equipment HVAC system, RB main steam tunnel HVAC system, and RB RIP ASD room HVAC system. The adequacy and acceptability of the design description and the ITAAC are evaluated in Section 14.3 of this report. On the basis of this evaluation, Open Item 9.4-1 as it relates to the above systems, is resolved.

The staff concludes that the RB non-safety-related equipment HVAC system, RB MST HVAC system, and RB RIP ASD HVAC system comply with applicable GDC referenced in SRP Section 9.4.5 and consequently with the subject SRP acceptance criteria and are, therefore, acceptable.

Auxiliary Systems

9.4.5.4 RB Safety-Related Electrical Equipment HVAC System

The system provides air and heat removal (cooling function) to maintain controlled temperature environments to ensure the continued operation of safety-related components (in the areas serviced by the system) under accident conditions. The system consists of three divisions, each serving five areas: the day tank, diesel generator engine, diesel generator MCC area, electrical equipment, and HVAC equipment rooms. Divisions A and B also serve remote shutdown panel rooms A and B, respectively, and Divisions B and C also serve RIP ASD control panel rooms. Division B of the HVAC system serves Division II electrical equipment room and the Division IV electrical equipment room. The staff evaluated these systems in accordance with SRP Section 9.4.5.

Each of the three RB safety-related electrical equipment HVAC system divisions consists of an ACU and redundant, 100-percent capacity supply and exhaust fans. Each ACU unit includes a medium-grade bag-type filter and a chilled water cooling coil for each supply fan and an electric heater. A safety-related divisional HECW system supplies chilled water. Each room is maintained at positive pressure with respect to the surrounding spaces under normal and accident conditions. Each divisional system is started manually from the control room. The system includes temperature control features. The system and the RB safety-related DG HVAC system supply fan maintain the DG room temperature below 45 °C (113 °F) when the DG is operating. The system maintains other areas served by it below 40 °C (104 °F). The system also has smoke removal capability. Upon receiving smoke alarm in a division of the system, operators will manually switch the system to smoke removal mode by closing the recirculation damper, stopping the exhaust fan, and opening the exhaust fan bypass damper to allow outside air to purge the affected area. No other division is affected by this action. Fire dampers separating electrical Divisions II and IV rooms that use fusible links in HVAC ductwork close under air flow conditions after the fusible links melt.

The system and components are located in the seismic Category I, reactor building, which protects against tornado missiles and flooding. All components are rated seismic Category I, and air intake and exhaust structures are protected against tornado missiles. Thus, the system meets the Position C.1 of RG 1.29 and, thereby, complies with the requirements of GDC 2 as it relates to protection from natural phenomena.

The power supply to the RB safety-related electrical equipment HVAC divisions during a LOOP. Medium-

grade bag-type filters in each of the three divisions of the HVAC system will remove dust and particulate matter from air intakes (supply air). The intake louvers are located at 15.2 m (50 ft) above grade, thereby meeting the guidance of Item 2, Section A, of NUREG/CR-0660. Therefore, the essential electrical equipment HVAC system meets the pertinent requirements of GDC 17 relating to the protection of essential electrical components from failure as a result of accumulation of dust and particulate material.

In the DSER (SECY-91-235), the staff stated that it would evaluate the system for compliance with GDC 4 for protection from the environmental and dynamic effects associated with equipment failures upon receiving system P&IDs and clarifying information on compliance with the GDC. GE submitted this information in revised SSAR Section 9.4.5.4, SSAR Figure 9.4-4, and component description Tables 9.4-3, 9.4-4 9.4-4a and b. The staff reviewed this information, the reactor building arrangement plan drawings, Section 3.6.1 of this report, and SSAR Section 3.6.4 and Table 3.6-2, and finds that the system components are protected against the effects of piping failures and internally-generated missiles by compartmentalization and separation, thereby satisfying the requirements of GDC 4. This resolved the unnumbered open item in DSER Section 9.4.5.4 not listed in Section 1.8 of the DSER.

In DSER (SECY-91-235) Interface Information Item 32, the staff stated that it would evaluate the monitoring of the exhaust from the areas served by the HVAC system and the electrical controls (i.e., interfaces) upon receiving additional system description. SSAR amendments through Amendment 34, GE included additional system description. The staff reviewed the information and the revised SSAR Section 11.5.2.2.4, and finds that the areas serviced by the RB safety-related electrical HVAC system contain no radioactive systems. These areas are at positive pressure with respect to the surrounding spaces (supply fan flow rate exceeds exhaust fan flow rate) and potentially contaminated adjoining areas. Because radioactive releases through exhausts from these areas to the environment are only from what would have to be brought first into these areas by their own supply fans, the exhaust from these areas are not monitored. Therefore, the DSER Interface Information Item 32 is resolved. For the reasons stated above, no HEPA filters or charcoal adsorbers are provided to control releases from these areas to the environment. Therefore, GDC 60 is not applicable to the system.

The staff concludes that the RB safety-related electrical equipment HVAC system meets applicable GDC referenced in SRP Section 9.4.5 consequently with the subject SRP acceptance criteria and, therefore, is acceptable.

9.4.5.5 RB Safety-Related Diesel Generator HVAC System

The RB safety-related DG HVAC system consists of three independent and identical divisions, each of which will service one of the three DG rooms. The divisions supply fresh air to ensure the continued operation of the safety-related diesels under accident conditions. The system and the RB safety-related electrical equipment HVAC system maintain the DG room below 45 °C (113 °F) when the DG is operating. The staff reviewed the system in accordance with SRP Section 9.4.5. Accordingly, the following criteria were considered: GDC 2 as related to the system being capable of withstanding the effects of earthquakes (meeting the guidance of RG 1.29, Position C.1, GDC 4, with respect to maintaining environmental conditions in essential areas compatible with the design limits of the essential equipment located therein during normal, transient, and accident conditions, GDC 5, as related to shared systems and components important to safety, GDC 17, as related to assuring proper functioning of the essential electric power system (protection from the effects of dust and particulate materials), and GDC 60, as related to the systems' capability to suitably control the release of radioactive effluent to the environment.

Each division consists of two intake louvers, a medium-grade inlet air filter, two supply fans, ductwork, and an exhaust louver. Both fans of each system draw air from outside, filter it, and distribute it to the respective diesel generator room. The exhaust air is forced out through the exhaust louvers. Each system is interlocked with the associated diesel generator starting system, but remote manual override capability is provided for the system fans. The systems are designed to facilitate inspection. The air intake louvers are located at 11.5 m (37.7 ft) and exhaust louvers at 8.5 m (28 ft) above grade, thereby meeting the guidelines of Item 2, Subsection A, of NUREG-CR-0660. Therefore, the systems meet the pertinent requirements of GDC 17 for protection of the essential electrical components of the systems against dust accumulation and particulate material. The power supply to each of the systems allows uninterrupted operation during LOOP.

From its review of SSAR Table 3.2-1, the staff concludes that each essential diesel generator HVAC system is composed of seismic Category I components. The air intake and exhaust structures are protected against tornado missiles. Each system, by virtue of its location in the seismic Category I flood-, tornado-, and missile-protected reactor building, is protected from the effects of earthquakes, tornadoes and floods. Consequently, the safety-related systems meet Position C.1 of RG 1.29, and, thereby, comply with the requirements of GDC 2.

In the DSER (SECY-91-235), the staff included a partial evaluation of system compliance with GDC, stating that it would complete a full evaluation upon receiving adequate information to demonstrate the system compliance with GDC 4. Therefore, the staff identified this issue of system compliance with GDC 4 as Open Item 64 in the DSER (SECY-91-235). After the staff issued the DSER (SECY-91-235), GE submitted a revised SSAR Section 9.4.5.5, SSAR Figure 9.4-4, and component description Tables 9.4-3, 9.4-4a and 9.4-4b. The staff reviewed this information, the reactor building arrangement drawings, and SSAR Figure 9.4-4, and finds that the system components are protected against the effects of piping failures and internally-generated missiles by being located in adequately separated and dedicated diesel generator rooms, thereby satisfying GDC 4 regarding protection from environmental and dynamic effects associated with equipment failures. This resolved DSER Open Item 64.

In the DSER (SECY-91-235), the staff stated that it would evaluate the monitoring of the exhaust from the areas served by the system and associated interfaces upon receiving P&IDs and additional system information from GE. The staff designated this lack of information as Interface Information Item 33 in the DSER (SECY-91-235). SSAR amendments through Amendment 34 included additional system description. The staff reviewed this information and the revised SSAR Section 11.5.2.2.4 and finds that the areas serviced by the system contain no radioactive systems. Because radioactivity releases from these areas to the environment are only from what would have to be brought first into these areas by their own supply fans, the system exhaust is not monitored. Therefore, the DSER Interface Information Item 33 is resolved. The system includes no HEPA filters or charcoal adsorbers to control releases from the system to the environment. Therefore, GDC 60 does not apply to the system. The cooling function of the system and smoke removal aspects are separately addressed in Sections 9.4.5.4 and 9.5.1.4.4 of this report.

The staff concludes that the RB safety-related diesel generator HVAC system complies with applicable GDC 2, 4, and 17 of SRP Section 9.4.5 as discussed above and, therefore, is acceptable.

9.4.5.6 RB Primary Containment Supply and Exhaust HVAC System

The system ventilates the primary containment, using supply from, and exhaust to, the secondary containment HVAC system. The staff reviewed operation of the system in accordance with SRP Section 9.4.5. The staff reviewed other aspects of the system in accordance with SRP

Auxiliary Systems

Section 6.2.4, BTP CSB 6-4 as discussed in Section 6.2.4 of this report.

The system has no safety-related function and is not required to maintain the reactor coolant pressure boundary or to achieve and maintain safe-shutdown conditions. The system consists of a purge supply fan, a HEPA filter, a purge exhaust fan, duct work, and controls. The system has the capability to purge the drywell and the wetwell, if required. When the system is in use and the air is not highly radioactive, the system discharges to the secondary containment HVAC system for filtering by a bag-type filter and exhausting to the plant vent stack. However, if the air is highly radioactive, it is discharged through the SGTS system. A high-radiation signal actuates an alarm, closes the isolation valves in the exhaust duct, and initiates the SGTS (for evaluation of SGTS, see Section 6.5.1 of this report). The system description, P&ID, and component description are in SSAR Section 9.4.5.6, Figures 6.2-39 and 9.4-3, and Tables 6.2-7, 9.4-4g, and 9.4-4h, respectively.

SSAR Section 6.2.5 describes the use of the primary containment purge system during the long-term post-accident cleanup operation and the deinerting operation (shutdown) during which the system provides air to the drywell and wetwell through the purge supply. Refer to Section 6.2.4.1 of this report for the above interfaces and compliance with BTP CSB 6-4.

The purge supply is filtered twice, first by a bag-type filter (secondary containment HVAC system) and then by a system HEPA filter. The purge air supply has an isolation damper downstream of the system HEPA, and the exhaust has redundant isolation dampers in the flow path through the secondary containment HVAC system and the safety-related SGTS.

The system is located in the seismic Category I reactor building, which protects against flood and tornado missiles. The system is non-safety-related. However, SSAR Table 3.2-1, the staff concludes that the radiation monitors and dampers for isolating the system are designed to seismic Category I requirements, and the applicable nonessential positions are designed in accordance with Position C.2 of RG 1.29. Thus, the system complies with GDC 2 regarding protection from natural phenomena by meeting Position C.1 of RG 1.29 for the essential portions, and Position C.2 for the nonessential portions.

In the DSER (SECY 91-235), the staff stated that it would review the essential portions of the system for compliance with GDC 4 upon receiving detailed system description including P&IDs. The staff identified the lack of information pertaining to system compliance with GDC 4 as Open

Item 65 in the DSER (SECY-91-235). SSAR amendments through Amendment 34 included the subject system information. The staff has reviewed the information and finds that the essential portions of the system are protected against damage from internally-generated missiles by separation of redundant equipment and are also protected against piping failures. Thus, the essential portions of the system comply with GDC 4 as they relate to protection from the environmental and dynamic effects of equipment failures. Therefore, DSER Open Item 65 is resolved.

GDC 17 does not apply to the system. The system complies with GDC 60 for control of releases of radioactive materials to the environment because under certain situations, the system is automatically isolated to prevent significant radioactivity release to the environment and facilitates the filtered release to the environment via the SGTS.

GE provided a set of design description and ITAAC relating to the primary containment (containment purge supply/exhaust) HVAC system. The adequacy and acceptability of the design description and the ITAAC are evaluated in Section 14.3 of this report. On the basis of this evaluation, DFSE Open Item 9.4-1 as it relates to the above systems, is resolved.

The staff concludes that the system design complies with applicable GDC referenced in SRP Section 9.4.5 and discussed above and consequently with the subject SRP acceptance criteria and, therefore, is acceptable.

9.4.5.7 RB Mainsteam Tunnel HVAC System

See Section 9.4.5.3 of this report.

9.4.5.8 RB Reactor Internal Pump Adjustable Speed Drive (ASD) HVAC System

See Section 9.4.5.3 of this report.

9.4.6 Radwaste Building HVAC System

The staff reviewed the radwaste building ventilation system in accordance with SRP Section 9.4.3. The system is located in the radwaste building. It is non-safety-related, and its failure does not compromise any safety-related system or component and does not prevent reactor safe shutdown. The system is designed to provide an environment with controlled temperature and air flow patterns to ensure both the comfort and safety of plant personnel and the integrity of equipment and components. The system services two zones: the radwaste building control room and the radwaste building process area.

Heating, cooling, and pressurization of the radwaste building control room are accomplished by an ACU which has two 100-percent supply fans (one standby). Outdoor air and recirculating air are mixed and drawn through a pre-filter, a high efficiency filter, a heating coil and a cooling coil by the associated fan. The supply sources for the heating and cooling coils are the hot water heating (HWH) system and the HNCW system, respectively. A differential pressure indicating controller modulates the inlet vanes in the supply fan air inlets to maintain the subject zone at positive pressure with respect to the surrounding spaces which includes the radwaste building process area. The zone has no exhaust fans; however, it has a 100-percent smoke removal fan to release smoke directly to the atmosphere, and this fan is operated manually. An area radiation monitor is provided in the radwaste control room and it will alarm on high radiation to alert personnel in the area.

The once-through HVAC system for the radwaste building process area consists of two 100-percent supply fans (one standby) each associated with a pre-filter, a high efficiency filter, a heating coil, and a cooling coil, and three 50-percent exhaust fans (one standby), each associated with a pre-filter and a high efficiency filter. Separate filters receive exhaust from the sorting table and compactor area. The exhaust air from the process area is monitored for radiation before it is released to the monitored plant vent. The overall air flow pattern in the zone is from the least potentially contaminated areas to the most contaminated areas. The design includes features to isolate manually affected areas on receipt of the area exhaust radiation alarm. The zone is at negative pressure with respect to surrounding spaces. The system description is given in SSAR Section 9.4.6.

The radwaste building ventilation system complies with GDC 2 regarding protection from natural phenomena because the applicable non-essential portions of the system are designed in accordance with Position C.2 of RG 1.29, as stated in SSAR Table 3.2-1. The design meets the requirements of GDC 60 (regarding control of releases of radioactive materials to the environment) for select areas in the radwaste building work areas by isolating the affected areas. Since the design of the system does not include HEPA's or charcoal adsorbers, the system is not required to meet RG 1.140 guidelines. In a letter dated August 22, 1990 (response to RAI 430.258), GE committed to submit the P&IDs, flow diagrams, and component description tables for the radwaste building HVAC system. However, the staff designated DFSER Open Item 9.4.6-1 because it did not receive this information until after issuing the DFSER. SSAR amendments through Amendment 32 specified COL License Information (Section 9.4.10, Item 9.4.10.2) which requires

the COL applicant to submit flow rates and equipment lists. GE also submitted SSAR Figure 9.4-10, the P&ID for the system. The staff finds GE's approach to resolve the subject open item acceptable. Therefore, this item is resolved.

Based on the above, the staff concludes that the radwaste building ventilation system design complies with the applicable GDC referenced in SRP Section 9.4.3 and consequently, with the subject SRP acceptance criteria and is, therefore, acceptable.

9.4.7 Diesel Generator Area Ventilation System

The diesel generator building ventilation system is part of the reactor building ventilation system, which is reviewed in Section 9.4.5.5 of this report.

9.4.8 Service Building Ventilation System

The staff reviewed the service building ventilation system in accordance with SRP Section 9.4.3. The system description is given in SSAR Section 9.4.8. The system is non-safety-related and is not required for accident mitigation, maintenance of reactor coolant pressure boundary integrity, or achievement and maintenance of safe shutdown. It is located in the service building and operates during all normal conditions manually and continuously and provides an environment with controlled temperature and air flow patterns to ensure both the comfort and safety of the plant personnel, and the integrity of equipment and components in the building. The system consists of two non-safety-related HVAC systems: (1) the clean area HVAC system, which serves the technical support center (TSC), the operational support center (OSC), and other normally clean areas of the service building (instrument repair, locker, men and women's change and lunch rooms and laundry), and (2) the service building controlled area HVAC system, which serves the balance of the service building.

The clean area HVAC system supplies filtered, heated or cooled air to both the clean and controlled areas through a central fan system consisting of an outside air intake; an air conditioning unit consisting of filters, heating coils, cooling coils, two 50-percent capacity supply air fans; and a supply air duct. Additionally, the system has two 50-percent capacity exhaust fans and an emergency filtration unit (EFU). An automatic damper in the supply system ductwork regulates the flow of air to maintain the TSC and other clean areas at a positive pressure with respect to the surrounding spaces. The exhaust fans discharge the ventilation air from the clean areas to the outside from the service building roof top. The system also functions during a high radiation mode. For this

Auxiliary Systems

purpose, radiation monitors are provided to detect radiation levels in the outside air intake. On receipt of a signal for high radiation in the normal outside air intake, the normal air intake damper closes, the minimum outside air intake opens, and the ventilation air for the clean areas is routed through the EFU. The EFU consists of a heater and demister, pre-filter, HEPA filter, 5.1-cm (2-in.) charcoal adsorber, a second HEPA filter, and two circulating fans. The EFU can remove at least 95 percent of all forms of iodine from the influent stream. If the reactor site is adjacent to a toxic gas source that could produce releases of significance to plant operating personnel in the TSC, the COL applicant will establish protection against the intrusion of toxic gas into the areas served by the system. GE has identified COL license information in SSAR Section 9.4.10 which calls for the COL applicant to locate required toxic gas monitors in the outside air intake of the clean area HVAC system, with capability to detect toxic gas concentrations at which personnel protective actions have to be initiated.

The controlled area HVAC exhaust system consists of two 50-percent exhaust fans. The controlled area exhaust is released to the environs through the monitored plant vent. The controlled area is maintained at a slightly negative pressure with respect to the surrounding spaces including the TSC and other clean areas.

The service building ventilation system complies with GDC 2 regarding protection from natural phenomena because the applicable non-safety-related portions of the system are designed in accordance with Position C.2 of RG 1.29, as stated in SSAR Table 3.2-1.

There is no need to monitor or control releases to the environment from the clean areas since these releases are from areas which do not contain radioactive sources and the only manner radioactive materials can get into these areas is due to its own HVAC system supply fans bringing outside air into the areas. The controlled area also does not contain radioactive sources; however, due to leakage from the secondary containment or turbine building, the radiation levels in the controlled area may become high. If this happens, the controlled area HVAC system can be manually isolated to prevent releases from the area to the environment. Thus, the service building HVAC system complies with GDC 60 with regard to control of releases of radioactive materials to the environment.

By letter dated August 22, 1990 (response to RAI 430.262), GE stated that the COL applicant will handle the details of the system, including P&IDs and component description tables, and compliance with applicable guidelines of RG 1.140 for the system filtration unit. The staff agreed with GE's approach. The staff designated this

as DFSEER COL Action Item 9.4.8-1. SSAR amendments through Amendment 34, discussed COL license information (Section 9.4.10) which requires the COL applicant to submit the above information. This is acceptable to the staff.

Based on the above, the staff concludes that the service building ventilation system design complies with GDC 2 and 60 as referenced in SRP Section 9.4.3 and consequently, with the subject SRP acceptance criteria and is, therefore, acceptable.

9.4.9 Drywell Cooling System

The staff reviewed the drywell cooling system in accordance with SRP Section 9.4.5. The system is not designed to ensure the reactor coolant pressure boundary or to achieve and maintain safe-shutdown conditions. However, the system provides features to include redundant components to provide reliability over the full range of normal plant operation. The system provides conditioned air and nitrogen to cool equipment and maintain temperature within limits, as specified in SSAR Section 3.11, during normal operation and in the drywell head area, upper and lower drywell, shield wall annulus, and the wet-well air space. The drywell cooling unit function is manually controlled from the control room.

Two of the three fan coil units operate under normal conditions. Each fan coil unit consists of two cooling coils arranged in series, a drain pan, and a centrifugal fan. The return air passes over the first coil, which is cooled by RCW. Part of the cooled air is subsequently cooled by the second coil, which is cooled by HNCW. The twice-cooled air is mixed with the air that bypasses the second cooling coil. During a LOOP (when no LOCA signal exists), the fan coil units start automatically if power is available from the diesel generators. During such a situation, only RCW coils will provide cooling. The system description, figure, and table are in ABWR SSAR Section 9.4.9, Figures 9.4-8 and 9.4-9, and Tables 9.4-1 and 9.4-2, respectively.

The drywell cooling system complies with GDC 2 for protection from natural phenomena because the applicable nonessential portions of the system are designed in accordance with Position C.2 of RG 1.29, as stated in SSAR Table 3.2-1. The requirements of GDC 60 regarding control of releases of radioactive materials to the environment do not apply because the system is only a recirculation system and does not contain any HEPA filters or charcoal adsorbers. GDC 14 and 17 do not apply because the system will not perform or support any safety function and is not used during an accident.

Based on the above, the staff concludes that the drywell cooling system meets the applicable acceptance criteria of SRP Section 9.4.5 and is, therefore, acceptable.

9.5 Other Auxiliary Systems

9.5.1 Fire Protection System

The Commission directed that special attention be given to measures for fire protection in addition to the staff's review of other aspects of the ABWR design in accordance with the requirements for current operating plants. For example, the Commission concluded that the ABWR design must incorporate the resolution of significant fire protection issues raised through operating experience and through the External Events Program.

The NRC established fire protection requirements for nuclear power plants in GDC 3, 10 CFR 50.48, and Appendix R to 10 CFR Part 50. The Commission considered Sections III.G., III.J, and III.O of Appendix R to be of particular importance. In July 1981, NRC revised BTP APCSB 9.5-1 (SRP Section 9.5.1) to include these provisions from Appendix R.

The staff has also issued supplemental guidance on fire protection in documents such as Generic Letter (GL) 81-12 (45 FR 76602, November 19, 1981), dated February 20, 1981, and GL 86-10, dated April 24, 1986. GL 81-12 presents information on safe-shutdown methodology, and GL 86-10 presents technical information on conformance with National Fire Protection Association codes and standards.

To minimize fire as a significant contributor to the likelihood of severe accidents for the ABWR, the staff concluded that current NRC guidance must be enhanced. As stated in SECY-90-016, "Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements," dated January 12, 1990 (and reiterated in SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs"). The proposed applicable regulation discussed below encompasses the enhanced, NRC Guidance.

The NRC expects any new reactor design to propose fire protection systems based on the best technology available, not on the methods allowed for plants already operating or in advanced stages of design and construction. Therefore, the staff evaluated the fire protection system of the ABWR against the criteria of SRP Section 9.5.1 (BTP CMEB 9.5-1 Rev. 2), which meets the requirements of GDC 3, and against the following proposed applicable regulation for fire protection:

The standard design must comply with 10 CFR Part 50 Appendix R, Section III.G.1.a and ensure that safe shutdown can be achieved assuming that all equipment in any one fire area will be rendered inoperable by fire and that re-entry into the fire area for repairs and operator actions is not possible. The design must ensure that smoke, hot gases, or fire suppressant will not migrate into other fire areas to the extent that could adversely affect safe-shutdown capabilities including operator actions. The control room is excluded because an alternative shutdown capability is provided which is physically and electrically independent of the control room. In the reactor containment, redundant shutdown systems must be provided with fire protection to ensure, to the extent practical, that one shutdown division be free of fire damage. Because of unique design layout, other areas may be accepted on an individual basis.

9.5.1.1 General Evaluation Fire Protection Program

GE generally followed the NRC's concept of defense in depth for fire protection as described in the SSAR. The three steps of defense in depth and GE's implementation of these steps follow.

- (1) To reduce the possibility of fire starting in the plant, GE used fire-resistant and fire-retardant materials in its design of the ABWR to minimize and isolate fire hazards. Low-voltage multiplexed circuits and fiber-optic circuits eliminate the need for cable-spreading rooms and thereby substantially reduce the amount of combustible cable insulation and higher voltage ignition sources in the control room.
- (2) To promptly detect and suppress a fire, GE has provided automatic detection and a suitable combination of automatic and manual fire suppression capabilities in the ABWR design.
- (3) GE designed the plant so that any fire that might occur will not prevent safe shutdown of the plant even if fire detection and suppression efforts fail.

SSAR Section 9.5.1 and Appendix 9A (Fire Hazards Analysis) describe a fire protection program intended to protect safe-shutdown capability, prevent release of radioactive materials, minimize property damage, and protect personnel from injury as a result of fire.

Auxiliary Systems

GE also considered such features of general plant arrangement as:

- access and egress routes
- equipment locations
- structural design features separating or isolating redundant safety-related systems
- floor drains
- ventilation
- construction materials

The SSAR reflects the use of applicable National Fire Protection Association codes and standards in design and layout of the ABWR. In the DSER (SECY-91-355), the staff identified as Open Item 102, the need for GE to describe any deviations from those codes and standards and to describe in the Fire Hazards Analysis the measures taken to ensure that equivalent protection is provided. Section 9.A.3 of SSAR Appendix 9A describes three deviations from the fire codes: (1) exceptions to penetration requirements (Section 9A.3.5), (2) wall deviations (Section 9A.3.6), and (3) door deviations (Section 9A.3.7). In each case, GE described each deviation, why it was required, and how the underlying fire protection requirements are still satisfied. The staff found the GE explanations acceptable. Therefore, this resolved DSER Open Item 102.

In the SSAR, Section 9.5.1, GE indicated that the ABWR design meets the commitments as specified in BTP CMEB 9.5-1 except in five cases:

1. Deviation from BTP CMEB 9.5-1, Section 7.j, "Diesel Fuel Storage Areas"

The BTP states that the diesel fuel oil tanks with a capacity greater than 4,164 L (1,100 gallons) should not be located inside buildings containing safety-related equipment. If located within 15 m (50 ft) of such buildings, the tanks should be housed in a separate building with a construction having a minimum fire resistance rating of 3 hours.

GE states that the capacity of each tank provides 8 hours supply, 12,113 L (3,200 gallons), for the emergency diesel generators. The diesel fuel oil day tanks are located in the reactor building but outside of secondary containment. Walls, ceiling, floors, and doors are all 3-hour fire barriers. SSAR Section 9.5.1 states that the sunken volume of the room will accommodate (hold) the entire contents of the tank and discharge from the automatic foam

sprinkler system for 30 minutes if an uncontrolled leak occurs.

The doors from each fuel tank room open into the respective DG equipment room which is in the same division, but in a different fire area. Therefore, should a fire occur in the fuel tank room, it will need to propagate and damage two 3-hour barriers before it can penetrate and threaten another division. Smoke removal in the fuel tank room is accomplished by the HVAC system.

The staff position is that the capacity of the fuel oil day tank exceeds the specified limits in the BTP; however, GE committed to add an automatic foam system with early detection and suppression capabilities. In addition, as required by the BTP, the SSAR indicates that the COL applicant will provide a fire brigade capable of extinguishing any oil-type fire that may occur onsite, including one in the diesel generator rooms.

Based on the above discussion, the staff finds GE's justification acceptable for having the diesel fuel oil day tanks inside the reactor building.

2. Deviation from BTP 9.5-1, Section C.7.b, "Control Room Complex"

The BTP 9.5-1 requires that automatic fire suppression should be provided for the control room complex subfloor if it contains cable runs unless cable is run in 4-inch or smaller steel conduit or the cables are in fully enclosed raceways internally protected by automatic fire suppression.

SSAR Section 9.5.1(2) states that the subfloor area will not contain a fire suppression system as recommended by BTP CMEB 9.5-1 Section C.7.b because the amount of cabling under the floor is substantially less than that used in current designs. The types of cables located in the raised floor area smolder for a long time and are self-extinguishing. Cables will be located within conduit. The control room is continuously staffed so that the presence of a fire will be quickly detected. There is also a fire detection system in the subflooring which will quickly detect a fire in that area. Finally, in the unlikely event that the control room were to require evacuation, the remote shutdown panels contain the necessary controls to bring the plant to cold shutdown.

The staff finds GE's justification for not installing a suppression system in the subfloor of the control room complex acceptable.

3. Deviation from BTP 9.5-1, Section C.7.b, "Control Room Complex"

The BTP 9.5-1 requires that the office space in the control room complex be provided with an automatic fire suppression system.

GE SSAR indicated in Section 9.5.1(3) that the office spaces contained in the control room complex do not have automatic fire suppression systems installed.

GE SSAR indicated in Section 9.5.1(3) that the control room complex is continuously manned so that any fire will be quickly detected and manual suppression commenced without delay. Papers are limited and stored in file cabinets, book cases or other storage locations except when in use. GE indicated that fire detection will be installed in the peripheral rooms including office spaces. (Refer to SSAR 9A.4.2.4.1.)

The staff finds GE's justification for not installing an automatic fire suppression system in the office spaces of the control room complex to be acceptable.

4. Deviation from BTP 9.5-1, Section C.7.b, "Control Room Complex"

The BTP 9.5-1 requires fire detectors for the consoles and cabinets in the control room complex.

SSAR Section 9.5.1(4) states that the control room complex is continuously staffed so that any fire will be quickly detected and manual suppression commenced without delay. The cabinets and consoles contain limited combustibles and are air cooled so that smoke from a cabinet fire will exhaust to the control room complex. A fire in any single cabinet or console will not disable the capability to safely shut down the plant.

The staff finds the above commitments acceptable since these consoles are not walk-in type and are small. Therefore, no detection need be installed within these consoles. If a fire was to occur, the control room detection system will alarm or a control room operator will see or smell the smoke. The automatic fire detectors and continuous presence of operators will likely result in the fire being discovered and suppressed early.

GE also indicated that fire detection is not provided in the cabinets or consoles because the control room ceiling and peripheral rooms are to be provided with early detection systems.

GE agreed to install a fire detection system in the sub-floor area of the control room.

The staff finds GE's justification for this deviation acceptable.

5. Deviation from BTP CMEB 9.5-1, Section C.7.i, "Diesel Generator Area"

The BTP indicates that an automatic fire suppression system should be installed to combat any diesel generator or lubricating oil fire; such systems should be designed to operate when a diesel generator is running without affecting the diesel.

GE states that the ABWR has an automatic foam sprinkler system for diesel generators and diesel day tanks. The generator is not a 100-percent sealed unit. Openings are provided for cooling purposes. When a diesel is operating and fire occurs, the automatic foam sprinkler suppression system will activate and there is a possibility that the foam could be siphoned into the generator openings, causing damage to the generator or a short circuit to occur.

The ABWR design assumes fire would result in the loss of function for one division. Thus, two more diesel generators are still available for duty. While the DG equipment is designed to continue operating, actuation of automatic foam system due to a fire or inadvertent actuation of the foam system should prompt an operator to shut down the affected diesel generator.

SSAR Section 9.5.1.3.7 states that the automatic foam sprinkler system is actuated by cross-zoned rate of temperature rise, rate-compensated heat detectors, and infrared detectors. The system will not discharge foam until a heat activated fusible link activates in the sprinkler head.

The applicant's automatic foam sprinkler system significantly reduces the probability of an inadvertent actuation of foam on the emergency diesel generators. The applicant's meets the design aspects of GDC 3 (BTP 9.5-1) and GI 57. Therefore, the staff finds GE's justification acceptable.

GE also indicated in the SSAR that excess water, foam, and diesel fuel oil spilled outside the reactor building will not have an adverse effect on any safety-related equipment.

9.5.1.2 Specific Features of Protection

9.5.1.2.1 Protection of Safe-shutdown Equipment

In general, the ABWR design relies on 3-hour-rated fire barriers to separate safe-shutdown equipment from the remainder of the plant and from redundant systems and components outside of primary containment. However, where safety related equipment is not divisionally separated by 3-hour fire-rated barriers, the safety-related equipment is electrically isolated from its respective division by fuses

Auxiliary Systems

and/or breakers for breaker coordination and multiple high impedance faults. This is documented in the SSAR, Section 9A.5. SSAR Section 9.5.13.12 states that the COL applicant will demonstrate that multiple high impedance faults of those circuits described in Table 9A.5-2, resulting from a fire within any one fire area, will not negatively affect other equipment fed from the same power. The use of 3-hour-rated fire barriers exclusively for separation of safe-shutdown equipment in the nuclear power block (NPB) areas outside containment is in accordance with the review criteria, and is acceptable. The use of fuses and/or breakers to address breaker coordination and high impedance fault concerns where safety-related equipment is not divisional, separated by 3-hour fire-barrier equipment as discussed in SSAR Section A.5, is in accordance with the review criteria and is acceptable. This design will ensure the capability for safe shutdown assuming all equipment in any one fire area will be rendered inoperable. In addition, this design provides the operator the ability to achieve safe shutdown without re-entry into the fire area for repairs or operator actions and meets the fire protection guidance of SECY-90-016.

The staff recognizes the need for open communication between compartments inside containment to relieve and equalize pressure following a high-energy line break. GE stated in the SSAR that the safety divisions will be widely separated around containment so that a single fire will not fail any combination of active components that could prevent safe shutdown. The staff finds that this meets the fire protection requirements of SECY-90-016, SECY-93-087, SRP Section 9.5.1, and 10 CFR Part 50, Appendix R, and is acceptable. The ABWR containment will be inerted with nitrogen during power operation, which will prevent propagation of any potential fire inside containment. In accordance with BTP CMEB 9.5-1, Section 7.a., the inerted containment significantly reduces oxygen so that combustion and/or fire will not be supported. This is an acceptable means of preventing a fire inside containment. The SSAR commits to an inerted containment. Therefore, the staff finds that this fire-suppression feature meets the guidelines of BTP CMEB 9.5-1, Section 7.a., and is acceptable. Therefore, the use of structural walls inside containment as fire barriers to separate safety-related systems (cabling, components, and equipment), even though such walls may not fully enclose the equipment requiring separation, is acceptable.

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

1. The MST is an exception to separation of redundant safety-related components outside containment.

SSAR Section 9A.4.1.4.26(9) states that all valves in the MST are designed to fail with acceptable consequences. For example, power-operated valves are backed up with air-operated valves not subject to damage from the same fire, or redundant valves are located in another fire area.

2. The MCR is an exception to separation of redundant safety-related components outside containment.

SSAR Section 9.5.1.1.2 describes alternative shutdown capability that is spatially remote from, and electrically isolated from, the MCR.

In the event of fire in the MCR, control room operators will have full safe-shutdown capability available at this alternative shutdown station. In the control room, fiber optics are used to operate power, instrument, and control circuits for safe-shutdown equipment.

In discussions with the staff, GE stated that all controls and instrumentation (except for scram function, HPCF, and MSIV) will utilize a touch screen and touch panel. The characteristics of the specific touch screen and touch panel used and the affects of a fire on this equipment will depend on the as-procured equipment. If a fire can cause inadvertent operation of the touch screen or panels, the COL applicant will need to demonstrate that under the worst-case scenario, the reactor can be safely shutdown in a controlled manner and that long-term core cooling can be established and maintained. In addition, at no time shall top of the active fuel be uncovered. This is a COL Action Item (GI 147, "Fire Induced Alternative Shutdown Control Room Interactions").

3. Inside containment is an exception to separation of redundant safety-related components.

The entire containment is one fire area. SSAR Section 9.5.1.1.2 states that the shutdown trains will enter containment widely spaced around the perimeter. This spacing ensures that no single fire will be able to damage any combination of active components that would prevent safe shutdown. In addition, the ABWR containment will be inerted with nitrogen during power operation, which ensures that any potential ignition/fire hazards inside containment will not propagate. In accordance with BTP CMEB 9.5-1, Section 7.a., the inerted containment significantly reduces oxygen so that combustion will not be supported. This is an

acceptable means of preventing fire within containment. GE committed to an inerted containment in the SSAR. Therefore, the staff finds that this fire suppression feature meets the guidelines of BTP CMEB 9.5-1, Section 7.a., and is acceptable.

9.5.1.2.2 Passive Fire Protection Features

Passive fire protection features for the ABWR design consist of building assemblies such as walls, partitions, floor-ceiling assemblies, columns, beams, fire dampers, fire penetrations seals, and fire doors. Penetrations through the building assemblies, such as doorways, hoistways, stairways, and cable trays and conduits are protected by appropriate fire-rated doors, dampers, plugs, and seals. SSAR Section 9.5.1.1.3 states that it intends to select passive fire-protection components of proven design, which have previously been tested and are listed by nationally recognized testing laboratories.

SECY-90-016 requires separation by 3-hour fire-barriers for redundant trains of all safety-related equipment, not just safe-shutdown equipment. Therefore, the staff reviewed the lack of 3-hour barriers between redundant trains of four separate systems: (1) SLCS, (2) SGTS, (3) FCS, and (4) spent fuel pool cooling system (FPC). The staff evaluations for these systems are provided below:

1. The SLCS is the alternative to the CRD system and is separated from it by 3-hour fire-rated barriers. This arrangement satisfies the SECY-90-016 requirements and is, therefore, acceptable.
2. During a meeting with the staff on May 5, 1992, GE committed to move one train of the SGTS to an available space on a mezzanine level directly above the location presently shown (SSAR Figure 9A.4-6). In a subsequent amendment to SSAR Section 6.5.1.3.1, GE states that the SGTS has independent, redundant active trains. The two SGTS trains are mechanically and electrically separated and are located in two side-by-side compartments separated by rated fire barriers and adjacent to the HVAC system exhaust. This arrangement satisfies the staff's concern because these two areas will be completely separated from each other by 3-hour fire-rated barriers.
3. GE changed SSAR Figure 9A.4-4 to show a 3-hour fire-rated barrier separating the redundant trains of the FCS. This arrangement satisfies the staff's concern and is, therefore, acceptable.
4. The FPC system is not a safety-related system. If the FPC is lost because of fire, the fuel pool will slowly heat up and eventually boil. GE calculated that the

pool would begin boiling in about 16 hours under the most adverse conditions of decay heat load from a recently off-loaded full-core with no emergency makeup water supplied to the pool. However, emergency makeup water can be supplied to the fuel pool within 30 minutes from (1) the RHR system, (2) the SPCU system, (3) MUWC system, or (4) the fire protection water system by means of hoses. The staff finds that the FPC is not required to prevent off-site releases and is not a safety-related system. Therefore, this staff concern is resolved.

9.5.1.3 Fire Protection System Description

GE submitted the design description and the ITAAC for the fire protection system. The staff designated this as DFSER Open Item 9.5.1.3-1. GE submitted a revised set of design description and ITAAC, which are evaluated in Section 14.3 of this report. Open Item 9.5.1.3-1 is resolved.

9.5.1.3.1 Fire Detection Systems

A previous amendment to the SSAR stated that the ABWR automatic fire detection systems are designed and installed in accordance with National Fire Protection Association (NFPA) Standards 72D and 72E, and will protect all safe-shutdown components from all significant hazards. In the DFSER, the staff stated that NFPA 72A, 72B, 72D, and 72E have been incorporated into NFPA 72, and are no longer separate standards. Therefore, all references to NFPA 72D and 72E should be changed to NFPA 72. The staff designated this as DFSER Open Item 9.5.1.3.1-1. Amendment 33 to the SSAR states that fire detection systems are designed according to NFPA 72 Class A and NFPA 70 Class 1 requirements. Therefore, DFSER Open Item 9.5.1.3.1-1 is resolved.

The ABWR will include detection capability for major cable concentrations, safe-shutdown-related major pumps, switchgear, motor-control centers, battery and inverter areas, relay rooms, fuel areas, and all other areas that may contain appreciable in-situ or transient combustibles. Detector devices will be selected on the basis of the type of anticipated fire and located on the basis of ventilation, ceiling height, ambient conditions, and burning characteristics of the involved materials. Detection systems will alarm and be annunciated in the control room and will give a distinctive audible and, if necessary, visual local alarm to aid the fire brigade in finding the fire.

Therefore, the staff concludes that the automatic fire detection capability provided for the ABWR meets the guidelines of Section C.6.a of BTP CMEB 9.5-1, as discussed above, and is acceptable.

Auxiliary Systems

9.5.1.3.2 Fire Protection Water Supply System

The dedicated fire protection water supply and distribution system is designed and will be installed in accordance with NFPA Standards 11, 13, 14, 15, 16, 16A, 20, and 24 to meet the anticipated needs for fixed water suppression systems and manual hose stations.

The sprinkler systems in the reactor building and the wet standpipe systems in the reactor and control buildings are designed in compliance with ANSI B31.1 and analyzed to remain functional following an SSE (seismically analyzed). Portions of the water supply system, including a tank, a pump, and part of the yard supply main, are also designed to these requirements. The remainder of the water systems are designed to appropriate fire protection standards. During normal operation, the seismically analyzed systems will be separated from those not seismically analyzed by normally closed valves and a check valve so that a break in the non seismically analyzed portion of the system cannot impair the operation of the seismically designed portion of the system.

The water supply system is required to be a fresh water system, filtered if necessary to remove silt and debris. The system has two sources, each with a minimum capacity of 1,136,000 liters (300,000 gal.). If the primary source is a volume-limited supply such as a tank, a minimum of 455,000 liters (120,000 gal.) must be passively reserved for use by the seismically-designed portion of the fire suppression system. This reserve will supply two manual hose reels for 2 hours. A diesel-driven pump in the train is designed to remain functional following the SSE. A jockey pump will maintain pressure on the system.

The turbine building will have modified Class III standpipes, hose reels, and ABC portable extinguishers throughout the building. The following fire suppression systems will have primary fire suppression capability for the following areas:

- Automatic closed-head sprinkler systems in the open grating area of the three floors under the turbine.
- Deluge foam-water sprinkler systems in the lube oil conditioning area and the lube oil reservoir area.
- A deluge sprinkler system in the hydrogen seal oil unit area.
- A preaction sprinkler system in the auxiliary boiler area.

The turbine building fire suppression systems receive water from the portion of the supply system that is not required to be seismically analyzed for SSE.

The main power, unit auxiliary, and reserve transformers will have deluge water spray suppression systems. The systems are automatically actuated by flame or temperature detectors. A dike is provided to collect oil and water beneath each transformer. Drains are provided for each pit to divert oil and water away from buildings and transformers. Shadow-type fire-barrier walls separate adjacent transformers.

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

The two fire pumps are located in separate fire areas, separated from each other and the plant by 3-hour-rated fire barriers.

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

On the basis that the fire water supply and distribution system conforms to the applicable NFPA standards mentioned above, the staff concludes that the system will meet the guidelines of Section C.6.b of BTP CMEB 9.5-1 and is acceptable

9.5.1.3.3 Water Fire Suppression Systems

The SSAR states that automatic water and foam-fire suppression systems will be installed over major fire hazards identified in the fire hazards analysis. Each system will be designed and installed in accordance with NFPA 11, 13, and 15.

Standpipe and hose stations will be installed throughout the plant as determined in the fire hazards analysis. The standpipe systems will be designed and installed in accordance with NFPA 14. Each hose station will be equipped with a maximum of 30 m (100 ft) of 3.8-cm (1.5-in.) hose and an adjustable on/off spray nozzle listed or approved by a nationally recognized testing laboratory.

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

Exterior hydrants and hose houses will meet needs described in the fire hazards analysis and will be designed and equipped in accordance with NFPA 24.

Control and sectionalizing valves in the fire water system will be electrically supervised and will be indicated in the MCR.

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

The fire protection water distribution and extinguishing systems will conform to the guidelines contained in Sections C.6.b and C.6.c of BTP CMEB 9.5-1 and is acceptable. This resolved DSER (SECY-91-355) Open Item 104.

9.5.1.3.4 Automatic Foam Fire Suppression Systems

GE specified an automatic foam sprinkler fire suppression system for protection of the EDGs. GDC 3 of Appendix A of 10 CFR Part 50 states that an inadvertent operation of the foam system should not adversely affect the diesel. Section C.7.i. of BTP 9.5-1 CMEB states that "Automatic fire suppression should be installed to combat any diesel generator or lubricating oil fire; such systems should be designed for operation when the diesel is running without affecting the diesel." GI 57 addresses the potential for safety-related equipment to become inoperable due to an inadvertent operation.

The applicant committed to meet the design aspects of GDC 3, BTP 9.5-1 and GI 57, therefore, the staff concludes that the automatic foam fire suppression systems are acceptable. This item was previously discussed in Section 9.5.1.1 of this report.

9.5.1.3.5 Fire Extinguishers

Portable fire extinguishers will be provided in areas with in-situ or potentially transient combustibles. They will be chosen on the basis of the anticipated type of fire in the area and the effect of the extinguishing agent on equipment in the area. The portable extinguishers will be selected, installed, and maintained in accordance with the requirements of NFPA 10.

The portable fire extinguishers will conform to the guidelines of Section C.6.f of CMEB 9.5-1 and will, therefore, be acceptable.

9.5.1.4 Fire Protection Support Systems

9.5.1.4.1 Emergency Communication and Lighting

The BTP 9.5-1 requires portable radios be provided for fire brigade and plant operations personnel to communicate during a fire incident. This communication system will have a distinct and separate frequency so that plant security force communications and actuation of protection relays will not be affected. The portable radio communication system will use fixed repeaters, as necessary, to ensure communications capability with any location in the station from the control room. The fixed repeaters will be arranged and protected so that exposure to fire damage will not disable the entire system. SSAR Section 9.5.2.6.2 states that "Design of fixed emergency communication and portable communication systems shall comply with BTP CMEB 9.5.1, Position C.5.g.(3) and (4)."

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

GE responded to staff questions about battery-powered emergency lights in harsh (extreme high or low temperature) environments by stating the following in SSAR Section 9.5.3.1.1(5):

- (f) Non-essential battery pack lamps shall be self-contained units suitable for the environment in which they are located.
- (g) The light fixtures for essential battery packs may be located remotely from the battery if the environment at the lamp is not within the qualified range of the battery. Alternatively, lamps powered from the station batteries may be provided.

Based on the applicant's commitments, the emergency communication and lighting design conforms to the guidelines of Section C.5.g of BTP CMEB 9.5-1 and is acceptable.

9.5.1.4.2 Emergency Breathing Air

The staff reviewed GE's commitments regarding the provision of emergency breathing air in accordance with BTP CMEB 9.5-1. Specifically, Position C.3(c) indicates that the fire protection program will include the provision

Auxiliary Systems

of fire brigade equipment including self-contained breathing apparatus for brigade members.

In accordance with BTP CMEB 9.5-1, GE has made the following commitments concerning the provision of emergency breathing air for the fire brigade staff. Emergency breathing air will be provided for fire brigade and control room personnel. The breathing air will be delivered by a self-contained apparatus or a storage reservoir. Full-face positive-pressure masks approved by the National Institute for Occupational Safety and Health will be used by all personnel required to use emergency breathing air.

A minimum of 10 self-contained breathing units will be provided for fire brigade use. Each unit will be provided with two extra air bottles located on site. The rated service life for the self-contained units will be a minimum of 1/2 hour. In addition to the two extra bottles for each self-contained unit, compressors will be provided so that exhausted air bottles may be quickly replenished. The compressors will operate in areas free of dust and contaminants and will be powered from a vital power bus so that breathing air is available if off-site power is lost.

The specific provisions for emergency breathing air conform to the guidelines contained in Section C.3(c) of BTP CMEB 9.5-1 and are acceptable.

9.5.1.4.3 Curbs and Drains

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

SSAR Section 9.5.1 states that the control room will not have floor drains. The SSAR describes the flow path that the water will travel if the subflooring was to overflow. The path traveled by the water will not jeopardize safety-related equipment because of floor drains and pedestals which elevate safety-related equipment. One division outside the control room will not be in the path of the discharged water.

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

The provisions for curbs and drains conform to the guidelines in Section A.2 of BTP CMEB 9.5-1 and are acceptable.

9.5.1.4.4 Smoke Control

In SECY-90-016, "Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements," dated January 12, 1990, the staff stated that the ABWR design must ensure that smoke, hot gases, or the fire suppressant will not migrate into other fire areas to the extent that they could adversely affect safe-shutdown capabilities, including operator actions.

SSAR Section 9.5.1.1.6 states that the HVAC systems will control smoke and meet the requirements of ASHRAE's, "Design of Smoke Control Systems for Buildings," and NFPA's "Recommendations Practice for Smoke Control Systems." GE describes the major features that will be incorporated into their design which include the following:

- Venting of the area of the fire to prevent the fire from causing pressure to increase.
- Pressure control across the fire barrier to ensure that any leakage is into the area experiencing the fire.
- Pressure control and purge air supply to prevent back flow of smoke and hot gases when fire-barrier doors are maintained open for access for manual fire-fighting activities.

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

The general arrangement of the ABWR design of safe-shutdown trains features a high degree of separation with no piping and minimal cabling interconnections between divisions. With such a physical arrangement, the ventilation system can become the most likely pathway for fire propagation and smoke dispersal. Outside primary and secondary containment, the ABWR employs separate, dedicated HVAC systems for each fire area containing safe-shutdown equipment.

This arrangement of the ventilation systems serving the areas containing safe-shutdown equipment facilitates the venting of smoke from one area containing safe-shutdown

equipment, and precludes the smoke from spreading to the redundant area containing safe-shutdown equipment. This arrangement also means that ventilation air supply, return, and exhaust for any fire area is independent of all other fire areas and that HVAC ducting does not penetrate the 3-hour-rated fire barriers separating fire areas. Therefore, the ABWR design will not include fire dampers outside the secondary containment. This simplifies not only design of the ABWR HVAC systems, but also installation and maintenance of the system throughout the life of the plant.

The HVAC system removes smoke in the event of fire. For all systems, except the secondary containment and diesel generator rooms, placing a HVAC system in the smoke-removal mode causes a damper in the return line to close and a much larger exhaust damper to atmosphere to open so that the system becomes a no-return, flow-through system with the large exhaust plenum vented directly to atmosphere. The pressure in the area experiencing the fire drops to atmospheric pressure plus the duct loss in the exhaust duct and the pressure in other divisional fire areas remains at their normal positive pressure of approximately 0.06 Kpa (1/4 of an inch of water). This assures that air leakage through any openings in the fire barriers surrounding the fires is towards the fire. GE has stated in SSAR Section 9.5.13.10, that during the detailed design phase the required differential pressure for each barrier will be calculated and the HVAC systems are to be designed to provide the required pressure. This is a COL Action Item and is not needed for design certification.

The smoke removal mode for the RB secondary containment HVAC system differs from that of the other systems, since common supply and exhaust systems are used for all three divisional areas within the secondary containment. The systems from each division are branched from the common system. Each supply and exhaust branch includes a dual purpose isolation/fire damper valve. Each exhaust branch also includes a two-position motor-operated volume damper. Upon detection of a fire, a normally non-operating exhaust fan is started to increase the negative pressure of the exhaust system. The motor-operated dampers in the exhaust ducts for the divisional/HVAC fire areas without the fire, reposition to their pre-determined fire settings to maintain normal negative pressure in their zones. The pressure in the HVAC and fire area will become more negative with the change in exhaust pressure. This establishes a differential pressure across the fire barriers surrounding the fire. As discussed above, the required differential pressure for each barrier will be calculated during the detailed design phase, and the HVAC systems will be designed to provide the required pressure.

GE also committed to install smoke detectors in the fresh air intakes for the ABWR secondary containment and the control building. If smoke enters the fresh air supply for these HVAC systems, they will isolate from the outside supply and start operating in the 100-percent recirculation mode.

Since the primary containment is inerted with nitrogen, the separation of HVAC systems inside primary containment is not an issue during plant operation.

To remove smoke from the diesel generator room, the once-through supply fans for the room start, and purge the room through the large exhaust opening [4 m² (about 43 ft²)]. The applicant has committed to design the smoke removal systems in accordance with ASHRAE's "Design of Smoke Control Systems for Buildings" and NFPA's "Recommendations Practice for Smoke Control Systems." The applicant states that a minimum of 0.06 pa (.25 in. H₂O) will be maintained across the fire barriers. The applicant states that sufficient flow and pressure will be available to prevent smoke migration into the non-fire area when doors are open to support any fire fighting operations.

Based on the above applicants commitments, the staff concludes that the design of the ABWR HVAC systems in the smoke-removal mode of operation is in accordance with the review criteria and is acceptable. This resolved DSER (SECY-91-355) Open Item 103. (See also Section 9.5.1.2.2 of this report.)

9.5.1.4.5 Access/Egress Routes

Fire exit routes will be clearly marked provided for each fire area. These routes will be designed to comply with applicable life safety codes and standards. The provisions for access and egress routes in all areas needed for operation of safe shutdown equipment will conform to the guidelines contained in Section III.J of Appendix R to 10 CFR Part 50, and are acceptable.

9.5.1.4.6 Construction Materials and Combustible Contents

GE has defined non-combustible materials used for interior wall and structural components, thermal insulation, radiation shielding, soundproofing, interior finishes, and suspended ceilings in Section 9A.2.3(4) of SSAR Appendix 9A. GE also states acceptance criteria for surface finishes in Section 9A.2.4(6) of SSAR Appendix 9A. Both the definition and the acceptance criteria are based on the technical requirements of American Society for Testing and Materials (ASTM)-E84/NFPA 255, and are acceptable to the staff.

Auxiliary Systems

BTP CMEB 9.5.1 Section 5.A.12 indicates that transformers installed inside fire areas containing safety-related systems should be of the dry type or insulated and cooled with noncombustible liquid. Transformers filled with combustible fluid located indoors should be enclosed in a transformer vault.

With regard to non-combustible liquid-insulated transformers, the staff identified in Section 9.5.1.4.6 of the DSER (SECY-91-355), an interface requirement involving consideration of the potentially unacceptable health hazards in the event of the release of material from these transformers.

As a result of staff review of interface requirements, GE clarified in a letter dated March 11, 1992, that transformers located within fire areas containing safety-related equipment will be of the dry-type only. For those areas utilizing liquid-insulated transformers, the COL applicant will include features to prevent the insulating liquid from becoming an unacceptable health hazard to employees if the material is released to the building environment. The staff will review these features on a plant-specific basis. This was DSER COL Action Item 9.5.1.4.6-1. GE later submitted SSAR Section 9.5.13.15, requiring the COL applicant to design protective features for liquid-insulated transformers. This is acceptable to the staff.

The use of non-combustible material and the use of dry-type transformers located within fire areas containing safety-related equipment conforms to BTP CMEB 9.5-1 and is acceptable.

9.5.1.4.7 Interaction with Other Systems

The aspects of the fire protection design to address interaction with other systems was reviewed in accordance with Section C.3.a of BTP 9.5.1 relative to the separation of safety-related systems from any potential fires in non-safety-related areas, and the existence of separate redundant divisions of safety-related systems not subject to damage from a single fire.

The vulnerability of safe-shutdown equipment to fire protection water is not an issue because the ABWR design includes three trains of safe-shutdown capability and assumes separate fire areas for each shutdown train, which can survive total loss of all equipment in any fire area. Safe-shutdown equipment in the ABWR design requires no special protection from the effects of failures of the fire-protection water-suppression systems. However, GE includes such protection and assumes no credit for it.

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

These provisions comply with the guidelines of BTP CMEB 9.5-1 Section C.5.a for building design with fire barriers and divisional separation of BTP CMEB 9.5-1 and are acceptable.

9.5.1.4.8 Preoperational Testing

All of the active components of the entire plant fire-protection systems are required to pass a preoperational acceptance test in accordance with the appropriate NFPA standard governing design and installation of the system. Components and systems that must pass the preoperational testing before being placed in service include the following:

- fire pumps - controls, flow volume and pressure
- water distribution - flush and hydrostatic
- control valves
- fire-detection and alarm systems, including electronic supervision for other fire-detection and fire-suppression systems
- water fire-suppression systems
- emergency radio communication systems
- emergency lights
- emergency breathing air systems and components

9.5.1.5 Administrative Controls

The staff will perform a detailed review of the administrative controls for various plant operations during the plant-specific licensing process of a COL application referencing the GE ABWR design. Items of interest under the administrative controls review will include:

- control of combustible materials such as combustible or flammable liquids and gases, fire-retardant-treated wood, plastic materials, and dry ion exchange resins
- transient combustible materials and general housekeeping, including health physics materials

- open-flame and hot-work permits, and cutting and welding operations
- quality assurance for fire-protection systems components, installation, maintenance, and operation
- qualification of fire-protection engineering personnel, fire brigade members, and maintenance and testing personnel for fire-protection systems
- instruction, training, and drills for fire brigade members

The staff designated this as COL Action Item 9.5.1.5-1 in the DFSER. GE added SSAR Section 9.5.13.18 to state that the COL applicant will submit the description of administrative controls outlined in SSAR Section 9.5.1.6.4. This is acceptable to the staff.

9.5.1.6 Summary

The staff finds the fire-protection design for the ABWR acceptable. As discussed above, the fire-protection features described in SSAR Section 9.5.1 and SSAR Appendix 9A (Fire Hazards Analysis) of the ABWR application conform to the applicable sections of CMEB BTP 9.5-1 and meet the applicable regulation discussed in Section 9.5.1 above.

9.5.2 Communications Systems

This topic is discussed in Section 7.7.15 of this report.

9.5.3 Lighting Systems

This topic is discussed in Section 8.3.5 of this report.

9.5.4 Diesel Generator Fuel Oil Storage and Transfer System

Section 9.5.4.1 of this report addresses compliance of all the diesel generator auxiliary systems with the requirements of GDC 2, 4, and 5. Section 9.5.4.2 of this report addresses issues specific to the diesel generator fuel oil storage and transfer system.

9.5.4.1 Diesel Generator Auxiliary Support Systems (General)

There are three standby (emergency) diesel generators (DGs) in the ABWR design. Each DG has the following auxiliary systems, which are addressed in detail in the sections of this report indicated: fuel oil storage and transfer (FOST) (Section 9.5.4.2), cooling water (DGCW) (Section 9.5.5), starting air (DGSA) (Section 9.5.6),

lubrication (DGLS) (Section 9.5.7), and combustion air intake and exhaust (DGCA) (Section 9.5.8).

Adequacy of the systems is based on compliance with the requirements of GDC 2 (protection against natural phenomena), GDC 4 (protection from environmental and dynamic effects of equipment failure), and GDC 5 (sharing of SSCs between units) as well as the recommendations of NUREG/CR-0660, "Enhancement of Onsite Emergency Diesel Generator Reliability." Compliance with the requirements of other GDC will be reviewed on a system-specific basis in later sections of this report.

The diesel engine vendor has not been selected, therefore, the interaction between the diesel engine and the auxiliary systems cannot be fully defined. The staff will evaluate the COL applicant's design of specific engine and support systems on a plant-specific basis.

Most components of the diesel generators and their auxiliary support systems are located in the seismic Category I reactor building structure, which will protect against the effects of tornados, missiles, and floods. Each diesel generator and its associated auxiliary systems is physically and electrically separated and is located in separate divisional areas of the reactor building. Thus, a failure of equipment in one division cannot adversely affect the safety function of more than one division of the diesel generator and its associated auxiliaries. The diesel generator exhaust silencer is located on top of the reactor building, well above the probable maximum flood level and is designed to function during design-basis events such as seismic vibrations, wind, hail, tornados, rain, and snow storms. Fuel oil storage tanks, pump motors, valves, and piping are located underground and are of seismic Category I construction. The only portions of the FOST system located above ground are the fill, sample, and vent lines. In the DSER (SECY-91-355), the staff concluded that GE should submit additional information on provisions to minimize the effect of tornado missiles for these exposed components. SSAR Section 9.5.4 states that the FOST system is protected from damage from flying debris carried by tornados. Subject to confirmation that the fill, vent, and sample connections were adequately protected from tornado missiles, the staff concluded in the DFSER that the system designs met the requirements of GDC 2 and 4, and RGs 1.115 and 1.117. This was DSER Open Item 80 and was also identified as DFSER Open Item 9.5.4.1-1. GE submitted Amendment 24 of the SSAR, which resolved this issue, by stating that the fill, vent, and sample connections are located outside of the seismic Category I buildings at or slightly above plant grade. It is highly unlikely that one tornado missile could damage more than one redundant division because (1) these connections are not part of the fuel path from the storage

Auxiliary Systems

tank to the diesel, (2) the fuel path is located entirely within the reactor building, and (3) the connections for each train are widely spaced around the outside of the reactor building. The staff finds the design features acceptable because the design complies with GDC 2 and 4 of the diesel generator auxiliary systems. Therefore, DFSEER Open Item 9.5.4.1-1 is resolved.

The requirements of GDC 5 regarding the sharing of SSCs do not apply because the ABWR is designed as a single-unit facility. An application for a multi-unit facility will require review of the design for compliance with GDC 5.

The staff evaluated the design of the diesel generator support (auxiliary) systems for the effects of postulated pipe failures as discussed in Section 3.6 of this report. The adequacy of the fire protection for diesel generators and the associated auxiliary support systems is addressed in Section 9.5.1 of this report.

The applicant referencing the ABWR design will provide information to ensure compliance with the recommendations of NUREG/CR-0660, "Enhancement of Onsite Emergency Diesel Generator Reliability." The staff will review this information on a plant-specific basis. This was identified as DFSEER COL Action Item 9.5.4.1-1.

The TS in SSAR Chapter 16 specify test intervals for the diesel generator auxiliary systems. Most auxiliary systems are tested during diesel generator start/run tests as required by RG 1.108. The TS prescribes additional tests for the fuel oil transfer system. The incorporation of these tests in the TS will resolve the open item in DSER Section 9.5.4.1 regarding the test frequency. This was TS Compliance Item 9.5.4.1-1. GE provided its set of TS in accordance with the standardized technical specifications for BWRs. Chapter 16 of this report discusses the evaluation of the TS. Based on the approval of the ABWR TS, this TS item is resolved.

Instrumentation is to be located in dust-tight steel cabinets with gasketed doors/openings and filtered louvers where ventilation is required. Ventilation is to be taken from a location high in the reactor building, approximately 11.5 m (38 ft) above grade. Construction-related activity will be required to use appropriate dust control techniques. Concrete flooring is to be painted with concrete or masonry paint.

GE submitted the design description and the ITAAC relating to the emergency diesel generator and its auxiliaries. This was identified as DFSEER Open Item 9.5.4.1-2. GE provided a revised set of design description and ITAAC. The adequacy and acceptability

of the design description and the ITAAC are evaluated in Section 14.3 of this report. On the basis of this evaluation, this item is resolved.

This section applies to the emergency diesel generator auxiliary systems, which include the diesel generator fuel oil storage and transfer system (FOST), diesel generator cooling water system (DGCW), diesel generator starting air (DGSA), diesel generator lubrication (DGL), and diesel generator combustion air intake and exhaust (DGCA). The staff reviewed these auxiliaries for conformance of the design criteria and bases to the Commission's regulations as set forth in the GDC of Appendix A to 10 CFR Part 50. The staff concludes that the plant design is acceptable and meets the requirements of GDC 2 as it relates to the ability of the auxiliaries and the structures housing them to withstand the effects of natural phenomena such as earthquakes, tornados, hurricanes, and floods; GDC 4 as it relates to structures housing the auxiliaries and the auxiliaries themselves being capable of withstanding the effects of externally and internally-generated missiles, pipe whip, and jet impingement forces associated with pipe breaks; and GDC 5 as it relates to the capability of shared systems and components important to safety to perform required safety functions. In addition, GE has provided adequate guidance for an applicant referencing the ABWR design to ensure compliance with the recommendations of NUREG/CR-0660.

The staff concludes that the design of the auxiliaries conforms to the above-mentioned GDC and the guidelines of SRP 9.5.4 and is, therefore, acceptable.

9.5.4.2 Diesel Generator Fuel Oil Storage and Transfer System

The staff reviewed the FOST system in accordance with SRP Section 9.5.4. The system should be designed to meet the requirements of GDC 2 (protection against natural phenomena), GDC 4 (protection from environmental and dynamic effects of equipment failure), GDC 5 (sharing of SSCs between units), and GDC 17 (availability of electric power systems). The ability of the FOST system to meet GDC 2, 4, and 5 is discussed in Section 9.5.4.1 of this report.

The FOST system provides a separate and independent fuel oil supply division for each diesel generator. The FOST system provides minimum storage capability for full-load operation of each diesel generator for 7 days without replenishment of fuel. GE submitted the acceptance criteria of the diesel generator support systems.

The ABWR design includes three standby diesel generators. Each diesel engine fuel oil and transfer

division consists of a day tank with sufficient capacity to supply fuel oil to power the diesel for 8 hours; a fuel oil storage tank with a capacity sufficient to power the diesel for 7 days; two fuel oil transfer pumps, an engine-driven and a redundant dc motor-driven fuel oil pump (both gravity fed) supplying fuel from the day tank to the engine fuel manifold; and associated piping, valves, instrumentation, and controls. Each FOST system division is independent and physically separated from the other two divisions, and each division is located in a separate quadrant of the reactor building. Thus, a single failure within any one of the divisions will affect only the associated diesel generator.

The SSAR states that selection of the fuel oil transfer pump is an interface requirement. Upon further evaluation, the staff determined that this requirement can be accomplished by identifying a COL action item in the SSAR for the applicant referencing the ABWR design to submit this information. Therefore, this interface requirement was reclassified as DFSER COL Action Item 9.5.4.2-1. Subsequently, GE added design information for one of the fuel oil transfer pumps. The system will consist of an engine-driven fuel oil transfer pump and a second electric motor-driven transfer pump. This is acceptable.

The staff noted as DSER (SECY-91-355) Open Item 81, that the type of motive power (required to be available during a LOOP) should be provided in the SSAR. GE amended the SSAR to identify the motive source for the pump as Class 1E bus power from its respective diesel generator. This resolved DSER Open Item 81.

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

Instrumentation provided for the FOST system includes level indication for the day tank, temperature sensors at the intake and discharge of the day tank, and pressure indication for the suction of the engine-mounted and dc motor-driven fuel oil pumps. In the DSER (SECY-91-355), the staff indicated as part of DSER Open Item 83 that the sensor on the tank discharge did not appear on the fuel oil system P&ID. GE submitted a revised figure showing this temperature sensor, which resolved this part of DSER Open Item 83. However, incorporation of the revised figure in the SSAR was identified as DFSER Confirmatory Item 9.5.4.2-1. Subsequently, GE modified Figure 9.5-6 in Amendment 25

of the SSAR which included this temperature sensor. Therefore, Confirmatory Item 9.5.4.2-1 is resolved.

Level sensors provide signals to start the fuel oil transfer pumps, one starting on low level, a second on low-low level. At the low level, a 60-minute supply (at full diesel generator load) of fuel oil is available for diesel generator operation. GE did not state whether storage tank level instrumentation was available. In describing its commitment for a stick gauge provision, GE stated that level switches are provided to monitor tank level. In the DSER (SECY-91-355), the staff identified inconsistencies in the listing of these level switches between parts of SSAR Sections 9.5.4 and 8.3. GE committed to include the level switches and the stick gauge in the appropriate parts of SSAR Section 9.5.4, Section 8.3.1.1.8.5, and the referenced figures. This commitment resolved DSER Open Item 82; however, incorporation of this information into the SSAR was identified as DFSER Confirmatory Item 9.5.4.2-2. GE submitted the necessary tank level information in SSAR Amendments 21, 22, and 25. Table 8.3-5 includes an alarm on low storage tank level, Section 9.5.4.5 discusses level alarms on the storage and day tanks, and Figure 9.5-6 includes the level sensor and stick gauge. The staff finds these modifications acceptable. Therefore, Confirmatory Item 9.5.4.2-2 is resolved.

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

SSAR Figure 9.5-6 depicts the standby diesel generator fuel oil system. From the review of this figure, the staff concluded in the DSER (SECY-91-355), that the fuel oil storage tanks and associated instrumentation should be added to the figure. In addition, the staff found discrepancies between the text and Figure 9.5-6, regarding the optional characterization of the electric fuel oil pump. Responding to RAI 430.273, GE stated, in part, that "two local fuel oil temperature indicators are provided (one in the suction line and one in the discharge line) from the day tank." Figure 9.5-6, however, shows only one temperature sensor. GE committed to include a revised Figure 9.5-6 in the SSAR that will include the fuel storage tanks and their associated instrumentation. This commitment resolved the other part of DSER Open Item 83. However, incorporation of these revisions into the SSAR was identified as DFSER Confirmatory Item 9.5.4.2-3. GE submitted the revised figure in Amendment 25 of the SSAR. The figure incorporates the tanks, their associated instrumentation, and temperature sensors at the suction and discharge of the day tanks, and

Auxiliary Systems

contains a footnote which allows the motor-driven pump to be added as an option. The staff finds the revisions acceptable. Therefore, Confirmatory Item 9.5.4.2-3 is resolved.

Section III.5 of SRP Section 9.5.4 addresses the need to minimize the creation of sediment turbulence at the bottom of the fuel oil storage tank during refueling. To ensure continuous operation of the diesel generator while refueling, the ABWR design relies on duplex filters and strainers between the storage tank and the day tank and at the fuel oil pump suction to remove any sediment. The SSAR suggests that refueling would probably occur while the day tank is full, which would allow time for sediment to settle before fuel is transferred from the storage tank to the day tank.

In the DSER (SECY-91-355), the staff stated that refueling procedures should be established as an interface requirement to verify that the day tank is full before refilling the storage tank, thereby minimizing the likelihood of sediment obstruction of fuel lines and any deleterious effects on diesel generator operation. This was identified as DSER Open Item 84. Upon further evaluation, the staff determined that this requirement can be accomplished by adding a COL action item requiring these procedures. As a result, this interface was reclassified as DFSER COL Action Item 9.5.4.2-2. GE included this information in the SSAR, which is acceptable.

GE described a program to ensure that the diesel fuel oil is tested and maintained according to the appropriate ASTM and ANSI requirements. Fuel oil is to be sampled and tested monthly for quality and contaminants. New fuel will be visually inspected before being added to the storage tank, and will be analyzed within 2 weeks for other required properties. Fuel oil not meeting all requirements will be replaced within a week. The system will be tested as part of the required diesel generator tests and hydrostatically tested before startup. Each fuel storage tank will be tested to ASME requirements every 10 years. The system design fuel oil quality and tests meet the requirements of RG 1.137. Based on the above information, the staff concludes that the FOST meets the requirements of GDC 17.

The design of the FOST system will meet the requirements of GDC 17, as related to the capability of the fuel oil system to meet independence and redundancy requirements, RG 1.9 and RG 1.137. As discussed in Section 9.5.4.1 of this report, this auxiliary system also meets the requirements of GDC 2, 4, and 5 and will incorporate the recommendations of NUREG/CR-0660. The system will also incorporate appropriate industry standards, i.e., ANSI N195-1976 and IEEE Standard 387.

Therefore, the staff concludes that the diesel generator fuel oil storage and transfer system meets the guidelines of SRP 9.5.4 and is acceptable.

9.5.5 Diesel Generator Cooling Water System

The staff reviewed the DGCW system in accordance with the GDC in SRP Section 9.5.5.

The function of the diesel generator cooling water system is to maintain the temperature of the diesel engine within a safe operation range under all load conditions and to maintain the engine coolant preheated during standby conditions. The system should be designed to meet the requirements of GDC 2 (protection against natural phenomena), GDC 4 (protection from environmental and dynamic effects of equipment failure), GDC 5 (sharing of structures, systems, and components between units), GDC 17 (availability of electric power systems), GDC 45 (inspection of cooling systems), and GDC 46 (testing of cooling systems). The ability of the ABWR diesel generator cooling water system to meet GDC 2, 4, and 5 is discussed in Section 9.5.4.1 of this report.

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

The ABWR design includes three standby diesel generators, each having a physically separate and independent engine cooling water division, as described in the preceding paragraph. Each cooling water division is powered from the respective diesel generator's safety-related Class 1E motor-control center. Therefore, the system meets the redundancy and single-failure criteria requirements of GDC 17.

During operation of the diesel generators, the temperature of the diesel engine cooling water is regulated by three-way temperature control valves. When the standby diesel generators are idle, the cooling water is heated by an

electric heater and maintained at 49 °C (120 °F), assuming ambient temperature of 16 °C (60 °F).

In the DSER (SECY-91-355), the staff stated that GE established interface requirements for specific information on the design and capability of the cooling water system and the keep-warm system including the following:

- (1) jacket-water circulating pump characteristics (NPSH and motive-power source, i.e., shaft, engine, etc.)
- (2) the keep-warm system description (design may or may not include a keep-warm pump)
- (3) temperature sensor selection (Amot-type or equivalent)
- (4) heat removal capability of system (to be based on maximum permissible diesel engine overload output)
- (5) expansion tank size
- (6) expected water loss over 7-day period and system volume capacity needed to ensure adequate volume is available to maintain system water level and pump NPSH without refill.

These requirements were identified as DFSER Open Item 9.5.5-1. Upon further evaluation, the staff determined that these requirements do not meet the definition of interface requirements implied in 10 CFR Part 52 and can be accomplished by adding a COL action item in the SSAR requiring the COL applicant referencing the ABWR design to submit this information. GE included this information in the SSAR, which is acceptable. This resolved DFSER Open Item 9.5.5-1.

In the DSER (SECY-91-355), the staff discussed a discrepancy between the text of SSAR Section 9.5.5 and Figure 9.5-7 regarding the circulating water pump. Section 9.5.5 described the jacket-circulating water pumps as engine- and motor-driven while Figure 9.5-7 described both as being motor-driven. (This discrepancy was considered part of DSER Open Item 85.) Additionally, the interface criteria of Section 9.5.13.6 stated that the selection of the motive power for these pumps was an interface requirement. This disagreed with the text of Section 9.5.5 and Figure 9.5-7, which clearly specified motive power for the pumps, although in an inconsistent manner. (This was the other part of DSER Open Item 85.) GE committed to submit consistent information regarding the power source for the jacket-cooling water pumps. The references to specific power sources were to be removed, and the selection of the power supply was to

be incorporated as DFSER COL Action Item 9.5.5-1. These changes resolve DSER Open Item 85; however, incorporation of these changes into the SSAR was identified as DFSER Confirmatory Item 9.5.5-1. The staff reviewed Amendment 23 of the SSAR and concludes that the modification to Section 9.5.13.6, requiring the COL applicant to submit information on the motive power for the cooling water pumps, is acceptable. Therefore, Confirmatory Item 9.5.5-1 is resolved.

The interface criteria for selecting this valve did not specify an Amot-type or equivalent valve. The staff concluded in DSER (SECY-91-355) Open Item 86 that GE should include this selection information as DFSER COL Action Item 9.5.5-1 for temperature sensor selection. GE committed to incorporate the reference to an Amot-type temperature sensor, or equivalent into the SSAR. Therefore, DSER Open Item 86 was resolved. However, incorporation of this reference into the SSAR was identified as Confirmatory Item 9.5.5-2 in the DFSER. The staff reviewed Amendment 23 of the SSAR and concludes that the modification to Section 9.5.13.6, requiring the COL applicant to submit an Amot-type temperature sensor or its equivalent, is acceptable. Therefore, Confirmatory Item 9.5.5-2 is resolved.

The SSAR states that the diesel engine has the capability to operate at full load for 2 minutes without secondary cooling. This will ensure that the diesel engine can operate at full load in excess of the time required to restore cooling water (RSW and RCW), which are sequenced onto the emergency power supply within 1 minute after a LOPP.

The DGCW system conforms with RG 1.9 for engine cooling water protective interlocks. All trips are bypassed during LOCA conditions except low cooling water pressure and low differential pressure of secondary cooling water. Both of these trips are 2-out-of-2 logic trips (the diesel generator system protective interlocks are discussed in Section 8.3.3.7 of this report). The cooling water system is provided with an expansion tank and expansion tank vent line, both of which are to be located above the system piping and pump location. A static head will ensure that the pumps and piping are filled with water. On the basis of the discussion for resolution of DSER (SECY-91-355) Open Items 85 and 86, the system meets the requirements of GDC 44 regarding the provision for cooling systems.

The SSAR includes a commitment that the operating procedures for the diesel generator will require the loading of the engine up to a minimum of 40-percent of full load (or lower as specified in the manufacturer's recommendation) for 1 hour after up to 8 hours of

Auxiliary Systems

continuous no-load or light-load operation. Such no-load or light-load conditions would exist for a LOCA with offsite power available. Procedures including this criterion will specifically meet the guidance listed in Item III.7 of SRP Section 9.5.5.

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

The design criteria and bases for the DGCW system will conform to GDC 17 and 44 for redundancy, physical independence, and cooling capability, and GDC 45 and 46 for inspection and testability of the system. As stated in Section 9.5.4.1 of this report, the design meets the requirements of GDC 2, 4, and 5, for the protection of equipment from environmental and dynamic effects and sharing of system components. In addition, GE has provided adequate guidance for an applicant referencing the ABWR design to ensure compliance with the recommendations of NUREG/CR-0660. The system meets the requirements of GDC 45 and 46 for inspection and testability. The design of the DGCW system will meet the requirements of GDC 44 for system operability. Therefore, the staff concludes that the system design meets the guidelines as specified above of SRP Section 9.5.5 and is acceptable.

9.5.6 Diesel Generator Starting Air System

The staff reviewed the DGCA system in accordance with SRP Section 9.5.6.

The design function of the DGSA system is to provide a supply of compressed air for starting the emergency diesel generator engines without external power. The air storage system is to perform its function in a manner that ensures that the time interval between a diesel engine start signal and a "ready-to-load" status is less than 20 seconds. The system is to be designed to meet the requirements of GDC 2 (protection from natural phenomena), GDC 4 (protection from the environmental and dynamic effects of equipment failure), GDC 5 (sharing of SSCs between units), and GDC 17 (availability of electric power systems). Compliance with the requirements of GDC 2, 4, and 5 is discussed in Section 9.5.4.1 of this report.

The ABWR design includes three emergency diesel generators, each of which has its own starting air division,

separate and independent of the starting air divisions for the other two diesel generators. Each starting air division consists of two 100-percent capacity sections, each of which is capable of supplying sufficient air for five automatic or manual start attempts without recharging the air-receiver tanks. Each starting air division consists of two air-compressors, two air-receivers, four air-admission valves (two redundant air-admission valves on each of two engine starting air manifolds), and associated piping and valves to connect system components.

One division of the DGSA system consists of two compressors and two air receivers. Controls are provided to automatically start and stop each air compressor to maintain the required pressure in each air receiver. Each compressor can be manually started to restore pressure in the air-receivers if needed. Each division is equipped with an air-receiver low-pressure alarm, which is indicated locally and displayed in the control room as part of a diesel generator trouble alarm. Each receiver is also equipped with a blowdown connection, which would be used periodically to manually blow down the receiver to remove any accumulated water from the tank.

In Amendment 16 of the SSAR, GE responded to Q430.285, stating that each air-dryer system includes an air dryer equipped with pre-filters and after-filters. SSAR Figure 9.5-8 did not specifically identify pre-filters and after-filters for the air dryers. Addition of these filters to the P&ID (or a statement specifically identifying the filters as an integral part of the air-dryer component) was identified as DSER (SECY-91-355) Open Item 88. GE committed to revise SSAR Figure 9.5-8 to incorporate the pre- and after-filters into the design. This modification resolved DSER Open Item 88. However, incorporation of the modification into the SSAR was identified as DFSER Confirmatory Item 9.5.6-1. Subsequently, the staff reviewed Amendment 25 of the SSAR and concluded that the addition of note 4 on Figure 9.5-8 clarified the provisions for pre- and after-filters on the air dryers. Therefore, Confirmatory Item 9.5.6-1 is resolved.

Each diesel generator is completely separate and independent of the others so that a malfunction or failure in one starting air division does not impair the starting capability of the other diesel generators. Therefore, the design meets the independence and redundancy requirements of GDC 17.

GE identified several design parameters as future interface requirements to be determined once a diesel generator vendor has been selected. Interface requirements to be specified included the devices to crank the engine, the duration of the cranking cycle, and the number of engine revolutions per start attempt. These interface requirements

would dictate design parameters such as the volume and design pressure of the air receivers (sufficient for five start cycles for each receiver) and compressor size (sufficient discharge flow to recharge the system in under 30 minutes). Once established, these interface criteria would provide an adequate basis for selecting component capacities. This was identified as DSER (SECY-91-355) Open Item 89 and incorporation of this information into the SSAR was identified as DFSER Confirmatory Item 9.5.6-2. Upon further evaluation, the staff has determined that this requirement can be accomplished by including a COL action item in the SSAR requiring the COL applicant to submit this information. GE included this information in the SSAR. Therefore, DSER Open Item 89 and DFSER Confirmatory Item 9.5.6-2 are resolved.

The air compressor's air-storage tanks, valves, and piping (up to the first connection on the engine skid) are designed in accordance with the requirements of ASME Code Section III, Class 3, requirements and are seismic Category I.

The DGSA system description did not include a reference to coolers at the discharge of the air compressors, although SSAR Figure 9.5-8 includes after-coolers located downstream of the starting air compressors. This discrepancy was identified as DSER (SECY-91-355) Open Item 90. GE committed to incorporate the coolers into the description of the starting air system. This modification resolved DSER Open Item 90; however, incorporation of the modifications into the SSAR was identified as DFSER Confirmatory Item 9.5.6-3. The staff reviewed Amendment 23 of the SSAR and concluded that Section 9.5.6.2 contained the required information regarding the coolers. The staff reviewed Figure 9.5-8 (Amendment 25 of the SSAR) and concluded that the after-coolers were adequately identified. The staff finds the SSAR modifications acceptable. Therefore, Confirmatory Item 9.5.6-3 is resolved.

The SSAR states that the starting air quality complies with the diesel engine manufacturer's recommendation regarding dew point, as opposed to the requirements stated in SRP Section 9.5.6 II.4.j. The staff will determine whether the system's air quality complies with SRP Section 9.5.6.II.4.j on a plant-specific basis.

Based on the above review, the staff concludes that the DGSA system meets the requirements of GDC 17 as it relates to the availability of electric power. As discussed in Section 9.5.4.1 of this report, this auxiliary system also meets the requirements of GDC 2, 4, and 5 and will incorporate the recommendations of NUREG/CR-0660.

The system, therefore, meets the guidelines of SRP 9.5.6 and is acceptable.

9.5.7 Diesel Generator Lubrication System

The design safety function of the DGLS system is to provide a supply of filtered lubrication oil to the various moving parts of the diesel engine (including pistons and bearings) during engine operation and during periods of standby to enhance first-try-start reliability. The basis for acceptance in the review was conformance of the design to GDC 17, regarding redundancy and physical independence, and the guidance and additional acceptance criteria of SRP 9.5.7. The staff discusses the ability of the system design to meet the requirements of GDC 2 (protection from natural phenomena), GDC 4 (protection from environmental and dynamic effects of equipment failure), and GDC 5 (sharing of SSCs between units) and the recommendations of NUREG/CR-0660 in Section 9.5.4.1 of this report.

The major components of the lubrication system include the engine lube oil pump (within the engine frame), an engine-driven pump, an oil cooler, a generator shaft lube oil cooler, an electric lube oil heater, a keep-warm circulating pump, oil filter, and strainer. Local alarms signal low oil pressure, high oil temperature, and low oil level. These signals are part of a general diesel generator trouble alarm located in the control room. The low lubrication oil level alarm is identified in SSAR Table 8.3-5, and refill is described in SSAR Section 9.5.7 as being performed on indication of low level (a lube oil supply pump actuates on a low level indication in the engine sump). In DSER (SECY-91-355) Open Item 91, the staff stated that SSAR Figure 9.5-9 did not show any level indication for the lube oil system. GE committed to correct the figure to include the level instrumentation, which had been mistakenly identified as a flow transmitter. This correction resolved DSER Open Item 91; however, incorporation of the modifications into the SSAR was identified as DFSER Confirmatory Item 9.5.7-1. GE supplied the revised figure in SSAR Amendment 25. The staff finds the revised figure acceptable. Therefore, Confirmatory Item 9.5.7-1 is resolved.

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

The system is designed to provide sufficient lubricating oil to support full-load diesel operation for 7 days.

Auxiliary Systems

In DSER (SECY-91-355) Open Item 92, the staff requested that GE submit the following specific design criteria: pump flows, operating pressure, temperature differentials, cooling system heat removal capabilities and electric heater characteristics for the DGLS. GE stated that the "lubrication system design criteria will be furnished as an interface criteria after selection of the diesel vendor is finalized." Upon further evaluation, the staff determined that this requirement can be accomplished by identifying a COL action item in the SSAR requiring the COL applicant to submit this information. As a result, the interface was reclassified as DFSER COL Action Item 9.5.7-1. The staff reviewed Amendment 23 of the SSAR (Section 9.5.13.5) and concludes that the COL action item would ensure that the COL applicant submits the required information. GE included this COL action item in the SSAR. This is acceptable.

GE stated that the protective features to prevent crankcase explosions and features to mitigate the consequences of such an event (such as relief ports) are vendor specific and would also be included as interface criteria after a diesel engine vendor is selected. The diesel generator would be protected from crankcase explosions by activating vacuum blowers to maintain the crankcase at negative pressure and shutting down the diesel on high-pressure conditions (unless a LOCA signal is present). Inclusion of these design criteria was identified as Open Item 92 in the DSER (SECY-91-355). GE committed to incorporate the above system design characteristics into the SSAR for use when the diesel vendor is selected. This information was to be incorporated into Section 9.5.8 of the SSAR and resolved DSER Open Item 92. Incorporation of the modifications into the SSAR was identified as DFSER Confirmatory Item 9.5.7-2. Upon further evaluation, the staff determined that these requirements should be included in the system design. GE modified SSAR Section 9.5.8 (Section 9.5.8.2) and Table 8.3-5 to include these design features. Therefore, Confirmatory Item 9.5.7-2 is resolved.

The DGLS is designed to maintain lubrication oil temperature and circulate heated lubrication oil under pressure to the moving parts of the diesel engine while the engine is in the standby mode. A lube oil priming pump will operate intermittently to keep the lube oil piping pressurized. This same pump is used in conjunction with the lube oil heater to maintain system temperature. On low lube oil temperature, both the heater and priming pump will automatically start thereby circulating heated oil throughout the system. The priming pump discharge pressure switch is Class 1E. Based on the above information, the staff concludes that the DGLS conforms to the recommendations of NUREG/CR-0660, for enhancing the starting reliability of the diesel engine.

All DGLS piping and components will be designed in accordance with ASME Code Section III, Class 3, requirements or ANSI B31.1 guidance and will be seismic Category I. GE did not designate the diesel engine interface for the lubrication system because the diesel engine is vendor-specific. Therefore, the components to be designated to meet ASME Code requirements, have not been separated from those required to meet the ANSI standard. To meet NRC requirements, all components up to the diesel engine interface must meet the ASME Code requirements. The NRC staff has accepted the ANSI classification for engine-mounted components if they are pressure tested to 1.5 times design pressure and the test is documented. Recognizing that the keep-warm heater and the priming pump do not have to be nuclear safety grade, the staff identified classification of components in the lubrication system as Open Item 93 in the DSER (SECY-91-355). The staff established Open Item 94 to have GE clarify which components meet the ASME Code requirements and which meet the ANSI requirements with the pressure testing provision. GE committed to revise the response to RAI 430.271 to clearly state which components are to meet the ASME Code requirements and which are to meet the ANSI requirements. Engine-mounted components are to be designed to the ANSI standards, all others, except the keep warm-heater and the priming pump, are to meet the ASME standard. The proposed modification resolved DSER Open Items 93 and 94; however, incorporation of the changes into the SSAR was identified as DFSER Confirmatory Item 9.5.7-3. The staff reviewed the revised response to RAI 430.271 in Amendment 22 of the SSAR and the revisions to Figure 9.5-9 in Amendment 25 of the SSAR. The staff found that the modifications to the RAI and the figure clearly state which components are to meet the ASME Code requirements and which are to meet the ANSI requirements. Therefore, Confirmatory Item 9.5.7-3 is resolved.

The DGLS conforms to RG 1.9 for the protective interlocks for the diesel engine lubrication system. All trips associated with the lubrication system are bypassed during LOCA conditions. The diesel generator system protective interlocks are discussed in Section 8.3 of this report.

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

The staff concludes that the design of the DGLS meets GDC 2 (protection from natural phenomena), GDC 4 (protection from environmental and dynamic effects of equipment failure), and GDC 17 (availability of electric

power), the guidelines of SRP Section 9.5.7, and the recommendations of NUREG/CR-0660 (GDC 5 does not apply to the ABWR design because it is a single-unit design). The DGLS design, therefore, is acceptable.

9.5.8 Diesel Generator Combustion Air Intake and Exhaust System

The design function of the DGCA system is to supply filtered air for combustion to the engine and to dispose of the engine exhaust to the atmosphere. Acceptance is based on conformance of the design to GDC 17, regarding redundancy and physical independence, the guidelines of SRP Section 9.5.8, the recommendations of NUREG/CR-0660, and industry codes and standards. The staff also assessed the ability of the system to supply sufficient combustion air and release sufficient exhaust gases to enable the emergency diesel generator to perform on demand. The compliance of the system design with the requirements of GDC 2 (protection from natural phenomena), GDC 4 (protection from environmental and dynamic effects of equipment failure), and GDC 5 (sharing of SSCs between units) is discussed in Section 9.5.4.1 of this report.

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

The staff designated selection of a combustion air flow capacity sufficient to ensure complete combustion as an interface requirement and as Open Item 95 in the DSER (SECY-91-355). Upon further evaluation, the staff determined that this requirement can be accomplished by including a COL action item in the SSAR requiring the COL applicant to submit this information. The interface was reclassified as DFSER COL Action Item 9.5.8-1. GE included this information in the SSAR, which is acceptable.

The DGCA meets the guidelines of RG 1.9, Revision 3 (July 1993) as it relates to system protective interlocks. All DGCA piping and components are designed to seismic Category I and ASME Code Section III, Class 3, requirements. Engine-mounted piping and components beyond the engine interface are considered part of the engine assembly and are seismic Category I as part of the diesel engine package.

The combustion air intakes are located on the side of the reactor building and are protected (by vertical grills) from tornado missiles. The intakes are located 11.5 m (37.7 ft) above grade and are designed to minimize any effects from dust and debris through the use of vertical grills set into the reactor building wall with filters located behind the grills. The intakes are protected from flooding by their location.

The diesel generator exhausts are partly housed within the reactor building with the exhaust silencer located on the roof of the reactor building. The design basis for the silencer requires that it be seismically qualified and able to withstand the effects of tornados. In DSER (SECY-91-355) Open Item 96, the staff stated that the means of protection from tornado missiles had not been adequately discussed. GE committed to update the SSAR to state that the silencers are seismically mounted and bolted in a horizontal position. However, this design change did not adequately address protection of the silencers from tornado missiles. Therefore, this issue was identified as DFSER Open Item 9.5.8-1. In SSAR Section 9.5.8, GE committed to house the system in a seismic Category I structure to protect against tornado-missiles. This resolved DSER Open Item 96; however, confirmation that the SSAR had been updated was identified as DFSER Confirmatory Item 9.5.8-1. The staff concluded that the system complies with GDC 4, RG 1.115, Revision 1 (July 1977) and RG 1.117, Revision 1 (April 1978) and the recommendations of NUREG/CR-0660. The staff reviewed Amendment 26 of the SSAR, which clarified that all parts of the system, except for the silencers, are housed in a seismic Category I structure (the reactor building). The exhaust silencers are seismically mounted and bolted down horizontally. Therefore, the silencers are unlikely to become missiles. The silencers are widely spaced around the top of the building so that it is unlikely that a single missile could damage more than one division of the system. A damaged silencer will not prevent a diesel from performing its safety function unless debris clogs the exhaust pipe. In such a case, the other divisions will be relied on to perform the safety function. The staff finds the additional information sufficient to conclude that the design includes adequate protection for the intake and exhaust system. Therefore, DFSER Open Item 9.5.8-1 and DFSER Confirmatory Item 9.5.8-1 are resolved.

Auxiliary Systems

Combustion air is not taken from the diesel generator room. Instead, combustion air and ventilation air enter the reactor building through common filters into the air inlet cubicle. Before the air enters the diesel generator room, it enters separate inlet plenums for ventilation and combustion. This design complies with NUREG/CR-0660 for combustion air and dust and dirt control in the portion of the reactor building housing the diesel generators.

The SSAR did not describe the design of the portion of the DGCA which extends from the crankcase vacuum blowers to the outside environment. DSER Open Item 97 (SECY-91-355) requested additional information regarding this area, identified in SSAR Figure 9.5-6. GE committed to modify the response to RAI 430.294 to state that this part of the system consists of piping only. This information resolved DSER Open Item 97; however,

incorporation of the modification into the SSAR was identified as DFSE Confirmatory Item 9.5.8-2. The staff reviewed Amendment 26 of the SSAR. Amendment 26 contained the modified response to RAI 430.294 stating that the gases are exhausted through a 150 mm (6-in.) pipe which passes through the reactor building wall. This additional information, along with Figure 9.5-6, adequately addresses the staff's concern. Therefore, this confirmatory item is resolved.

Based on the above review, the staff concludes that the DGCA system meets the requirements of GDC 17 as it relates to the availability of electric power. As discussed in Section 9.5.4.1 of this report, this auxiliary system also meets the requirements of GDC 2, 4, and 5 and will incorporate the recommendations of NUREG/CR-0660. The system, therefore, meets the guidelines of SRP 9.5.8 and is acceptable.

10 STEAM AND POWER CONVERSION SYSTEM

10.1 Summary Description

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

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

10.2 Turbine Generator

10.2.1 Turbine Generator System

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

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

The turbine generator is equipped with an electrohydraulic control system that will perform two basic functions: (1) turbine speed control for a variety of operating load conditions for which a digital control and monitoring system will be used and (2) turbine overspeed protection. The design functions of the turbine speed control system are (1) to control turbine speed throughout the normal range of load conditions and ensure that a full-load turbine

trip will not cause the turbine to overspeed beyond its design overspeed and (2) to provide turbine overspeed protection to minimize the probability of the generation of turbine missiles, in accordance with GDC 4. The turbine control system is, therefore, important to the overall safe operation of the plant.

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

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

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

Protection of safety-related SSC from turbine generator missiles is also assured by proper turbine-generator

Steam and Power Conversion System

orientation. This is discussed in SSAR Section 3.5.1.1.3 and reviewed in Section 3.5.1.3 of this report.

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

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

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

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

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

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

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

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

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

10.2.2 Turbine Rotor Integrity

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

assembly. The staff reviewed the measures taken by the applicant to assure turbine rotor integrity and reduce the probability of turbine rotor failure.

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

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

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

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

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

The combined stresses of low low-pressure turbine rotor at design overspeed resulting from centrifugal forces, interference fit, and thermal gradients will not exceed 75 percent of the minimum specified yield strength of the material. The design overspeed of the turbine will be

5 percent above the highest anticipated speed resulting from a loss of load. In the DFSER, the staff stated that the applicant or licensee referencing the ABWR standard plant design should provide the basis for the turbine design overspeed to the NRC. This requirement should be included in the turbine ITAAC and was identified as DFSER Open Item 10.2.2-2. The staff, upon further consideration, reclassified this as a COL action item. GE addressed this item in the SSAR Section 10.2.5, which states that the COL applicant will provide the basis for the turbine overspeed design to the staff for review. This is acceptable to the staff and therefore, Open Item 10.2.2-2 is resolved.

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

The turbine rotor design will be a solid forged monoblock rotor rather than one with shrunk-on disks. The current practice employed by the turbine manufacturers is to bore the center of the monoblock rotor to remove metal impurity and permit internal inspection. A forged or welded rotor will not be as susceptible to stress corrosion cracking as experienced in the shrunk-on disks. However, the one-piece rotor design requires stringent part-machining inspections. Therefore, the applicant referencing the ABWR design must submit inspection requirements for a one-piece rotor to the NRC staff for review and approval before plant operation. Further, the applicant must submit an actual turbine inspection schedule following the third refueling outage. The actual turbine inspection schedule should be based on a probability calculation of turbine missile generation. The calculated probability for turbine missile generation is expected to be less than or equal to $1.0E-4$ per year since GE specified that the turbine be favorably oriented. The NRC-approved methodology for calculating probability for turbine missiles is discussed in NUREG-1048 (Supplement 6, July 1986). This was identified as DFSER COL Action Item 10.2.2-1. GE addressed this item in the SSAR Subsection 3.5.1.1.1.3 to state that the COL applicant will submit for NRC approval, within 3 years of obtaining a COL, a turbine maintenance program including

Steam and Power Conversion System

probability calculations of turbine missile generation based on methodology approved by the NRC. Further, during the preservice inspection of the turbine, each machined turbine rotor is subjected to 100-percent ultrasonic examination and surface visual examinations, using established acceptance criteria. The ISI program for the turbine assembly includes the disassembly of the turbine and complete inspection of all normally inaccessible parts, such as couplings, coupling bolts, turbine shafts, low-pressure turbine brackets, and low-pressure and high pressure rotors. This is acceptable to the staff.

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

10.3 Main Steam Supply System

10.3.1 System Description and Operation

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

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

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

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

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

pneumatic pressure and are analysed to demonstrate structural integrity under SSE loading conditions.

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

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

In addition to meeting the acceptance criteria of SRP Section 10.3, the MS system for the ABWR must also be capable of mitigating the radiological consequences of an accident that could result in potential offsite exposures comparable to the dose reference values specified in 10 CFR Part 100. Most of the currently licensed BWRs rely on the MSIV leakage control system to mitigate the

radiological consequences of MSIV leakage following a design-basis loss-of-coolant accident (LOCA) and to stay within 10 CFR Part 100 limits if the MSIV leakage rate exceeded the technical specification limit of 0.3 m³/hr (11.5 ft³/hr). The ABWR will not have an MSIV leakage control system and, therefore, will rely on the MS system coupled with the main condenser (MC) to contain MSIV leakage, relying on plateout and holdup of radioactive iodine and to limit the radiological consequences to within 10 CFR Part 100 limits.

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

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

The steam lines in the reactor building (including the containment and some portion of the steam tunnel) and in the steam tunnel portion in the control building, are located in seismic Category I, flood-protected and tornado-protected structures. Thus, these portions of the MSSS

Steam and Power Conversion System

meet the requirements of GDC 2 and the guidelines of RG 1.29, "Seismic Design Classification," (Rev. 3) Positions C.1 and C.2.

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

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

Confirmatory Item 10.3.1-2. GE has included this information in SSAR Section 3.6.5.1. Therefore, Confirmatory Item 10.3.1-2 is resolved.

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

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

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

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

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

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

10.3.2 Steam and Feedwater System Materials

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

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

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

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

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

10.4 Other Features

10.4.1 Main Condenser

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

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

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

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

AIM

Association for Information and Image Management

1100 Wayne Avenue, Suite 1100
Silver Spring, Maryland 20910
301/587-8202

Centimeter

Inches

MANUFACTURED TO AIM STANDARDS
BY APPLIED IMAGE, INC.

6 of 10

Steam and Power Conversion System

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

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

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

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

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

Based on this information, the staff concludes that the main condenser design includes provisions which assure that failures of the system will not result in excessive releases of radioactivity to the environment, do not cause unacceptable condensate quality, or result in flooding of

areas housing safety-related equipment and therefore meets the requirements of GDC 60.

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

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

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

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

10.4.2 Main Condenser Evacuation System

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

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

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

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

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

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

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

10.4.3 Turbine Gland Sealing System

The staff reviewed the TGSS in accordance with SRP Section 10.4.3. Acceptance is based on TGSS meeting the

requirements of GDC 60 for controlling the releases of radioactive materials to the environment and the requirements of GDC 64 for monitoring the release of radioactive material to the environment.

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

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

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

Steam and Power Conversion System

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

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

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

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

10.4.4 Turbine Bypass System

The staff reviewed the TBS in accordance with SRP Section 10.4.4. The design is acceptable if, in accordance with GDC 4, failure of the system (due to a pipe break or

system malfunction) does not adversely affect safety-related systems or components and if, in accordance with GDC 34, the system can shut down the plant during normal operations. Use of this system will eliminate the need to rely solely on safety systems.

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

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

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

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

The TBS will not serve or support any safety function. No safety-related equipment is located inside the TB. All high-energy lines associated with the TBS are located in

the TB. Therefore, failure of the TBS are any safety-related equipment or hamper the capability for safe shutdown of the plant.

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

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

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

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

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

10.4.5 Circulating Water System

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

The CWS is partially within the ABWR scope. The in-scope portion of the system includes all piping, valves, instrumentation, and controls within the TB. All other

equipment outside the TB, including the CWS pumps is outside the ABWR scope and is the responsibility of the COL applicant.

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

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

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

The CWS pumps will be vertical, wet pit-type, capable of delivering approximately 45,430 m³/hr (200,000 gpm) per pump. The discharge line of each pump will be equipped with a butterfly valve to allow isolation and maintenance of any one pump while the others are in operation. The CWS pumps will be tripped and the pump and the condenser isolation valves closed on a high-high level condenser pit signal. A condenser pit high-level alarm will actuate in the control room.

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

In the DFSE, the staff stated that GE had not included an analysis of flooding in the TB and could affect safety-related equipment. The staff requested GE to provide a flood analysis which characterized the nature of the hazards and the design features to protect safety-related equipment from flooding in the TB. This was identified as DFSE Open Item 3.4.1-1. Subsequently, GE provided a flood analysis for the TB. The major flood hazard in the TB is from a failure in the CWS, which is an open-cycle system. Leak detectors in the condenser pit will alert the control room and automatically isolate the CWS on indication of building flooding. A postulated failure of the isolation function can result in flooding of the TB up to grade level. A non-watertight truck door at grade level

Steam and Power Conversion System

will allow release of the flood water onto the ground. As stated in Section 3.4.1 of this report, the below-plant-grade tunnel connecting the radwaste, reactor, and TB is sealed to prevent water from entering the reactor building. The staff finds that the design adequately protects safety-related SSCs from the effects of flooding as a result of pipe failures in the CWS. GE has also included this information in the SSAR. Open Item 3.4.1-1 is resolved.

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

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

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

10.4.6 Condensate Purification System

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

The staff has reviewed the design of the sampling equipment, sampling locations, and instrumentation to monitor and control the CCS parameters and finds the

design acceptable. However, in the DSER (SECY-91-355), the staff identified that SSAR Section 10.4.6 contained insufficient information for the staff to evaluate conformance with SRP Section 10.4.6 in the following areas:

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

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

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

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

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

10.4.7 Condensate and Feedwater System

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

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

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

trains. Downstream of the high-pressure feedwater heaters, the feedwater is combined into a common header that discharges into the reactor through two parallel 56-cm (22-in.) nominal diameter feedwater lines, as stated in SSAR Section 5.4.9.3.

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

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

Steam and Power Conversion System

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

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

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

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

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

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

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

10.4.8 Power Cycle Heat Sink

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

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

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

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

Steam and Power Conversion System

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

Because the PCHS is a non-safety-related system and is not required to remove reactor heat, the requirements of

GDC 44 do not apply. All active and passive system components will be accessible for inspection, maintenance, and testing during normal power operation. Therefore, the inspection and testing requirements of GDC 45 and 46 are met.

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

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

11 RADIOACTIVE WASTE MANAGEMENT

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

The staff reviewed the applicant's design, design criteria, and design bases for the radioactive waste management systems for the ABWR design. The acceptance criteria that the staff used for this evaluation are set forth in Section II of Standard Review Plan (SRP) Sections 11.1, 11.2, 11.3, 11.4, and 11.5, which include 10 CFR 50.34(a) as it relates to the technical information contents in safety analysis reports, 10 CFR 20.106 as it relates to radioactivity in effluents to unrestricted areas and 10 CFR Part 71, as it relates to packaging of processed solid wastes. In lieu of 10 CFR 20.106, the staff used 10 CFR 20.1302, which is the current requirement. The staff also used compliance with 10 CFR 50.34(f)(2)(xvii) as it relates to instrumentation for monitoring noble gases and continuous sampling of radioiodines and particulates in gaseous effluents during an accident and onsite capability to analyze and measure these samples, as acceptance criteria for the gaseous effluent monitoring and sampling systems. Additionally, the staff used compliance with 10 CFR 61.56 as it relates to waste characteristics for the waste products that result from solid waste processing. The above SRP sections, additionally, include the following general design criteria (GDC) of 10 CFR Part 50, Appendix A, as acceptance criteria for radioactive

waste management systems and liquid and gaseous process and effluent monitoring and sampling systems: GDC 3 as it relates to providing protection to gaseous waste handling and treatment systems from the effects of an explosive mixture of hydrogen and oxygen; GDC 60 as it relates to the radioactive waste management systems being designed to control releases of radioactive materials to the environment; GDC 61 as it relates to the liquid and gaseous radioactive waste management systems and ventilation systems for the fuel storage and handling areas being designed to assure adequate safety under normal and postulated accident conditions; and GDC 63 and 64 as they relate to solid radioactive waste management system and liquid and gaseous process and effluent monitoring and sampling systems being designed to monitor radiation leakages and radioactivity releases to the environment. The compliance of radioactive waste management systems and liquid and gaseous process and effluent monitoring and sampling systems with the above regulations are discussed in the sections that follow. Because specific compliance with Appendix I to 10 CFR Part 50 and the guidelines given in American National Standards Institute (ANSI) N13.1, "Guide to Sampling Airborne Radioactive Materials in Nuclear Facilities," Regulatory Guide (RG) 1.21, "Measuring and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants," and RG 4.15, "Quality Assurance for Radiological Monitoring Programs (Normal Operation)--Effluent Streams and the Environment," is not within the scope of ABWR design, the staff will review individual combined license (COL) applications for the ABWR design to ensure their conformance with these documents. Therefore, this was identified as draft final safety evaluation report (DFSER) COL Action Item 11.0-1. By amended standard safety analysis report (SSAR) GE included COL License Information (Section 11.2.5.1, Item 1) which states Chapter 11 that the COL applicant will show compliance with 10 CFR Part 50, Appendix I and the guidelines given in ANSI N13.1 and the RGs 1.21 and 4.15. This approach by GE is acceptable. GE has also included this action item in the final certified SSAR.

Certified Design Material

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

Radioactive Waste Management

11.1 Source Terms

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

11.2 Liquid Waste Management System

11.2.1 System Description and Review Discussion

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

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

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

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

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

* Excludes dissolved noble gases and tritium

** Includes chemical wastes

Radioactive Waste Management

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

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

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

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

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

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

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

normal waste generation rate for the LCW system will be about 58 m³/day (15,200 gpd); GE estimates 55 m³/day (14,530 gpd). The capacity of the limiting processing equipment is 720 m³/day (190,080 gpd). The difference between the expected normal waste generation rate and the design process flow rate provides adequate reserve for processing a surge in LCW generation rate.

The HCW subsystem will collect and process water of relatively high conductivity, such as the wastes from the floor drains (from drywell, reactor, radwaste, turbine, and service buildings). The HCW subsystem will also collect and process chemical wastes from chemical laboratories and laboratory drains. The wastes will be collected in one of two parallel HCW collector tanks, chemically adjusted to a suitable pH for evaporation, and concentrated in one of two parallel forced-circulation concentrators or evaporators to reduce the volume of water and decontaminate the distillate. The distillate will be demineralized by the HCW demineralizer and normally will be routed to the LCW system upstream of the polishing demineralizer. During normal processing, the LCW polishing demineralizer will be bypassed and the processed HCW will be directed to the LCW sample tanks. The processed stream will then be routed to the condensate storage tank for reuse. A small fraction of the processed HCW may be discharged from one of the LCW sample

tanks should the plant's water balance dictate such a discharge. The staff estimates that approximately 10 percent of the processed HCW will be discharged. The distillate from the HCW demineralizer also can be routed to an HCW distillate tank to be reprocessed by the HCW demineralizer, if required. The concentrated waste from the evaporator will be routed to the concentrated waste storage tank for further processing by the solid waste system. The staff estimates that the normal waste generation rate for the HCW system will be approximately 24 m³/day (6300 gpd); GE estimates 15 m³/day (4000 gpd). The capacity of the limiting processing equipment in this system is 142 m³/day (37,440 gpd), leaving adequate reserve for processing a surge in the HCW generation rate.

The detergent waste subsystem will collect and process detergent wastes from personnel showers and laundry operations. Detergent wastes will be collected in the single hot shower drain (HSD) receiver tank, processed through one or two HSD filters, and routed to the HSD sample tank before discharge. GE estimates that the normal generation of detergent waste will be approximately 11 m³/day (3000 gpd); the staff's estimate is 4 m³/day (1000 gpd). In a June 7, 1990, submittal, GE further stated that, if storm drains are included, the waste

Radioactive Waste Management

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

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

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

* In accordance with RG 1.143

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

*** Design pressure

† Design temperature

generation for this subsystem will be approximately 31 m³/day (8300 gpd). The storm drain water will be normally nonradioactive, but can become radioactive on contact with radioactive liquids. The storm drain water will be collected in one of the two HSD sample tanks and discharged after processing by the HSD filters, if needed. The staff estimates that all of the detergent wastes and storm drain water will be discharged. The capacity of the limiting processing equipment in this system is 284 m³/day (75,000 gpd), which, together with the system's tanks will ensure adequate margin to collect and process any surge in the waste generation.

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

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

11.2.2 Conclusion

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

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

The staff further concludes that there is reasonable assurance that the COL applicant will be able to meet the Appendix I to 10 CFR Part 50 dose guidelines for radioactive materials released through liquid effluents with regard to the minimum discharge canal flow rate of 340 m³/hr (1500 gpm) and the additional dilution credit of at least a factor of 10 between the point of release and the region in the unrestricted area where the water is used. The staff considers demonstrating compliance with 10 CFR Part 50, Appendix I, is the COL applicant's responsibility. So the staff will evaluate this compliance individually for each COL application. This was identified as DFSEER COL Action Item 11.2.2-1. By amended SSAR Chapter 11, GE identified COL License Information (Section 11.2.5.1, Items 1 and 4) which requires the COL applicant to demonstrate the above compliance. This approach by GE is acceptable. GE has also included this action item in the final certified SSAR.

The staff's conclusions are based on the following findings:

- (1) On the basis of ABWR parameters that govern reactor coolant system concentrations of radionuclides and design of the liquid radwaste treatment systems, as stated in Section 11.2.1 of

Radioactive Waste Management

this report, the staff estimates that the total of radioactive wastes discharged via the liquid effluent during normal plant operation including anticipated operational occurrences will be no more than 7,400 MBq/yr (0.2 Ci/yr) (excluding tritium) and 2.2×10^6 MBq/yr (59 Ci/yr) for tritium. This finding, in conjunction with SSAR Section 11.2.3.2 assumed minimum dilution flow rate of 340 m³/hr (1500 gpm) and at least an additional credit of a factor of 10 (as stated in SSAR Section 11.2.3.2) between the point of discharge and the region in the unrestricted area where water is used, provides reasonable assurance that the ABWR liquid waste management system will meet the applicable 10 CFR Part 50, Appendix I dose guidelines for liquid effluents.

- (2) GE's ABWR design has met the requirements of 10 CFR Part 20 that will have the minimum discharge flow rate of 340 m³/hr (1500 gpm) and for which liquid waste can only be discharged from either the HSD sample tank or the LCW sample tank. The staff has considered the potential consequences resulting from reactor operation and has determined that the concentrations of radioactive materials in liquid effluent averaged over a year, as permitted by 10 CFR 20.1302 for the above case, will be well below the limits in 10 CFR Part 20, Appendix B, Table 2, Column 2. Instantaneous discharge concentrations of the radionuclides in liquid effluents to an unrestricted area will also be within these limits because GE has stated in a submittal dated November 5, 1990, that the discharge rate via the single discharge line will be administratively controlled to conform to these limits. The staff will review the administrative controls to limit the instantaneous discharge concentrations of the radionuclides in liquid effluents to an unrestricted area to comply with the limits in 10 CFR Part 20, Appendix B, Table 2, Column 2, as explained in Section 11.2.1 of this report, on a plant-specific basis for the COL applicants. Therefore, this was identified as DFSER COL Action Item 11.2.2-2. By amended SSAR Chapter 11, GE included COL License Information Section 11.2.5.1, Item 5 which states the COL applicant will provide administrative controls to ensure the limits mentioned above. This approach by GE is acceptable. GE has also included this action item in the final certified SSAR.
- (3) GE's ABWR design has met the requirements of GDC 60 and 61 with respect to system design for controlling releases of radioactive materials to the environment. The staff considered the capabilities

of the proposed liquid radwaste treatment system to meet the demands of the plant resulting from anticipated operational occurrences and has concluded that the system's capacity and design flexibility are adequate to meet the anticipated needs of the plant as discussed in Section 11.2.1 of this report. The staff also reviewed GE's quality group classifications which are used for the system components and the seismic design applied to structures housing these systems. In the DFSER, the staff stated that quality assurance (QA) (Operation) provisions of the liquid radwaste systems will be reviewed individually for each COL application and identified this as DFSER COL Action Item 11.2.2-3. By amended SSAR Chapter 11, GE included COL License Information Section 11.2.5.1, Item 6 which states the COL applicant will provide QA (Operations) provisions of the liquid radwaste systems. This approach by GE is acceptable. GE has also included this action item in the final certified SSAR.

The design of the systems and structures housing these systems meets the applicable criteria given in RG 1.143, Revision 1. Specifically, the base mat and outside walls of the housing structures are seismic Category I to a height necessary to retain spilled liquids within the building. In the draft safety evaluation report (DSER) (SECY-91-235), the staff stated that the provisions incorporated in the ABWR design to control the release of radioactive materials in liquids resulting from inadvertent tank overflow were consistent with the criteria given in RG 1.143 except for the lack of a local alarm capability for the condensate storage tank (CST). This was identified as Open Item 70 in the DSER (SECY-91-235). GE responded in Amendment 20 to SSAR Section 9.2.9.2, Item (9), by stating: "Instrumentation shall be provided to indicate CST water level in the main control room. High water level will be alarmed both locally and in the main control room." This satisfies the staff concern regarding RG 1.143 requirements for a local high level alarm for CST which will be located in the radwaste building control room as stated in SSAR Section 11.2.1.2.1. Therefore, Open Item 70 is resolved.

IE Bulletin 80-05 identified an issue: cooling hot water in a low pressure tank could create a low vacuum condition and buckle the tank, releasing radioactive material or having other detrimental effects. In a fax dated May 21, 1992, GE stated that several low-pressure tanks that could contain primary system water have vents to prevent the development of a low vacuum condition. In the DFSER,

the staff stated that the above information resolved the tank failure concern in IE Bulletin 80-05, subject to documentation of the information in the SSAR. Therefore, the staff identified the resolution of the issue as DFSEER Confirmatory Item 11.2.2-1. By amended SSAR Section 11.2.4, GE included the above information by stating that the only tanks in the LWMs that can contain reactor water, diluted by other wastes are the LCW and HCW collector tanks and that these tanks are vented to preclude their vacuum collapse. This is acceptable to the staff, and therefore, this item is resolved.

The staff concludes that the liquid waste management system for the ABWR meets the acceptance criteria of SRP Section 11.2 and is, therefore, acceptable.

11.3 Gaseous Waste Management System

11.3.1 System Description and Review Discussion

The gaseous radioactive waste processing and plant ventilation systems are designed to collect, store, process, monitor, and discharge potentially radioactive gaseous wastes that will be generated during normal operation of the plant, including anticipated operational occurrences. The systems will consist of equipment and instrumentation necessary to reduce release of radioactive gases and particulates to the environment.

The principal sources of gaseous wastes in the plant will be the effluents from the offgas system, condenser mechanical vacuum pump, turbine gland seal system (TGSS), and ventilation exhausts from the radwaste building, containment purge, reactor building, and turbine building. All these effluents will be routed into the plant stack either directly or indirectly (TGSS and mechanical vacuum pump) and monitored continuously.

The major source of gaseous radwaste during normal plant operation before treatment will be the offgases from the main condenser air ejector. These will principally contain hydrogen and oxygen from the radiolytic decomposition of water, air from condenser in-leakage, fission and activation gases, and water vapor. To treat this effluent, the ABWR design uses an offgas processing system consisting of redundant catalytic hydrogen-oxygen recombiners, charcoal absorber delay beds, and a HEPA filter operating at ambient conditions.

The offgases will be diluted with sufficient steam in the last stage of the air ejector to reduce the hydrogen concentration to less than 4 percent by volume upstream of the recombiner. The offgases will be preheated in the first stage of the recombiners to approximately 177 °C (350 °F) to remove moisture before recombination and reduced in

hydrogen concentration to less than 1 percent by volume by the recombiner(s). The recombiner effluent will subsequently be cooled to between 57 °C (135 °F) and 68 °C (154 °F) by the offgas condenser. The offgas condenser will also include baffles to reduce moisture entrainment. The offgas stream will be further cooled to 18 °C (65 °F) by the cooler condenser. The pressure boundary of the system will be detonation resistant, with a design pressure of 2413 kPa (350 psig). Redundant, nonigniting, detonation-resistant hydrogen analyzers will monitor hydrogen concentration downstream of the recombiners and alarm both locally and in the control room when appropriate.

Fission and activation gases will be held for decay in the charcoal absorber system downstream from the offgas condensers. Before entering the delay beds, these gases decay for 2.5 minutes during their transit from the main condenser to the delay beds. The charcoal absorber beds consist of one guard bed absorber followed by four parallel trains of two absorber beds in series. The total mass of charcoal will be 114,000 kg (250,000 lb). The offgas system is also designed to prevent, monitor, and suppress the potential ignition and combustion propagation of charcoal in the charcoal absorber tanks, with the necessary temperature elements in the charcoal tanks and connections for nitrogen purge and blanketing.

Before discharge, the offgas system effluent stream will be passed through an HEPA filter assembly to remove particulates. The holdup times in the ambient offgas treatment system charcoal beds at 38 °C (100 °F) with a dew point of 18 °C (65 °F) were calculated according to the methodology of NUREG-0016, Revision 1, and GE proprietary report NEDO-10751. The staff estimates these times to be approximately 30 days for xenon, 44 hours for krypton, and 18 hours for argon; GE estimates 42 days for xenon and 46 hours for krypton. These beds will also absorb iodines from the treatment system effluent. The offgas system is designed to withstand a hydrogen explosion.

The ventilation exhausts from all plant areas such as the reactor building (RB) (which includes the primary containment when it is vented or purged, the fuel handling area, the area housing the emergency core cooling system equipment, other areas such as the standby gas treatment system (SGTS) rooms, the fuel pool cooling system equipment rooms, and areas housing nonessential equipment), the service building controlled area, the radwaste building, and the turbine building, will be directed to the plant vent and monitored continuously before being released to the environs. The RB areas mentioned above will be serviced either directly or indirectly (for the primary containment purging or venting) by the secondary

Radioactive Waste Management

containment ventilation system during normal plant operation. However, ventilation exhausts from RB areas serviced by the RB safety-related electrical equipment heating, ventilation, and air conditioning (HVAC) system and the RB safety-related diesel generator HVAC system, the turbine island serviced by the electrical building ventilation system, and the control building areas serviced by the control building safety-related equipment area HVAC system will be discharged directly to the environs unmonitored. The lack of monitoring for the exhaust from these areas is evaluated in Section 11.5 of this report which concludes that this is acceptable. Although SSAR Section 9.4.8.1.2 states that the service building controlled area ventilation exhaust will be monitored before its release, SSAR Table 11.5.1 does not indicate any explicit monitoring provision for the subject exhaust. Therefore, the staff concluded in Open Item 71 in the DSER (SECY-91-235) that this discharge will not be monitored. In its response of December 19, 1991, to the issues raised in the DSER, GE stated: "The service building ventilation exhaust is sent to the plant release point where the offgases are monitored and sampled during release." By amended SSAR Section 11.5.2.2.4, GE further clarified the issue by stating that the exhaust from the service building controlled area will be discharged to the environs via the monitored plant vent. Therefore, the issue of service building ventilation exhaust monitoring is resolved. Open Item 71 in the DSER (SECY-91-235), is resolved.

On the basis of SSAR Section 9.4 and GE's responses dated May 23 and August 27, 1990, the staff stated in Open Item 72 in the DSER (SECY 91-235) that neither the mechanical vacuum pump exhaust nor the normal ventilation exhaust system of any building includes either charcoal adsorbers or HEPA filters to remove elemental and organic forms of iodine and particulates from the applicable effluent stream. In calculating the gaseous effluents, the staff, therefore, assumed that all the exhausts will be discharged to the environs untreated. In its response of December 19, 1991, GE confirmed the staff's assumption. The staff's assumption does not invalidate its conclusion that the gaseous waste management system discussed in Section 11.3.2 of this report is acceptable. Therefore, the issue of lack of charcoal absorbers and HEPA filters is resolved and Open Item 72 in the DSER (SECY-91-235) is resolved.

The calculated release values for iodines and particulate in Table 11.2 of this report reflect the above assumption. SSAR Sections 6.5.1, 9.4.5.1.3, and 9.4.5.1.5 state that, if high radiation is detected in the secondary containment exhaust or in the refueling floor atmosphere, the secondary containment normal ventilation system will be secured and the exhaust discharged through the safety-related SGTS, which consists of charcoal adsorbers and HEPA filters.

The SGTS exhaust also will go through the monitored plant vent.

The plant stack and the major streams feeding the plant stack (offgas system and building ventilation systems) will be monitored to facilitate appropriate corrective action in a timely manner to prevent offsite release exceeding applicable limits. Additionally, the offgas treatment system includes an automatic control feature to terminate the post-treatment release if it exceeds a preset radiation level in the effluent. However, as part of Open Item 73 in the DSER, the staff questioned whether the monitors for the secondary containment exhaust will be sufficiently sensitive to detect a high-radiation level in the primary containment purge exhaust as specified by Branch Technical Position CSB 6-4, "Containment Purging During Normal Plant Operations." In SSAR Section 11.5.2.1.2, GE confirmed the sensitivity of these monitors to detect a high-radiation level in the primary containment purge exhaust. This part of Open Item 73 in the DSER (SECY-91-235) is resolved.

As the second part of Open Item 73 in the DSER (SECY-91-235), the staff noted that while SSAR Section 9.4.4.2.1.1 states that the turbine building ventilation exhaust will be monitored before its discharge to the monitored plant vent, SSAR Table 11.5-2 did not indicate any such process monitoring provision for that exhaust. GE amended SSAR Tables 11.5-1 and 11.5-2 to indicate that the turbine building ventilation exhaust will be monitored. This part of Open Item 73 is resolved. Both parts of Open Item 73, in DSER (SECY-91-235) are, therefore, resolved.

As stated above, all airborne radioactivity releases except offgases from the incinerator will be through the monitored plant vent. The plant vent is located on the reactor building at 76 m (249 ft) above grade and is the tallest point on the site. Incinerator offgases will also be monitored prior to their release to the atmosphere, through incinerator exhaust stack.

11.3.2 Conclusion

In evaluating the gaseous radwaste management system, the staff used the SRP criteria pertaining to (1) the capability of the system to maintain releases below the limits in 10 CFR Part 20 during periods of fission-product leakage at design levels from the fuel, (2) the capability of the system to meet the processing demands of the station during anticipated operational occurrences, (3) the quality group and seismic design classification applied to the equipment and to components and structures housing the system, (4) the design features that are incorporated to control the releases of radioactive materials in accordance

with GDC 60, and (5) the potential for gaseous releases resulting from hydrogen explosions in the gaseous radwaste system. The staff also reviewed the capability of the offgas system to limit the whole-body dose to less than 10 percent of the 10 CFR Part 100 limits for an individual exposure of 2 hours at the nearest exclusion area boundary (EAB) as a result of radioactive releases from a postulated offgas system leak or failure as assumed in BTP ETSB 11-5, Revision 0, July 1981. The staff reviewed all the applicable information provided in the amended SSAR as well as GE's submittals dated May 23, June 29, August 22, September 14, and October 26, 1990, December 19, 1991, and June 23, 1993.

The staff concludes that the gaseous radioactive waste management system design for the ABWR meets the applicable requirements of 10 CFR 20.1302 and GDC 3, 60, and 61 with regard to radioactivity in gaseous effluents released to unrestricted areas, fire protection, control of releases of radioactive materials, and radioactivity control in the gaseous waste management system and ventilation system associated with fuel storage and handling areas.

The staff expects that the COL applicant will provide an operational demonstration that the system design complies with Appendix I to 10 CFR Part 50 numerical guidelines for offsite radiation doses as a result of gaseous or airborne radioactive effluents during normal plant operation, including anticipated operational occurrences. Therefore, this was identified as DFSEER COL Action Item 11.3.2-1. By amended SSAR Chapter 11, GE included COL License Information in Section 11.3.11.1, which states that the COL applicant will demonstrate this above compliance. This approach by GE is acceptable. GE has also included this action item in the final certified SSAR.

Nonetheless, the staff has evaluated the ABWR design to determine if there is reasonable assurance that the COL applicant will be able to meet the Appendix I dose guidelines for design objectives. The ingestion, inhalation, and external irradiation of ground contamination pathway doses to applicable organs resulting from release of radioactive iodines, radioactive material in particulate form, and tritium and carbon-14 via airborne effluents depend on a number of site-dependent parameters. The population exposures (person-rem) and associated cost-benefit analysis are also site dependent. Therefore, the staff considered only if the standard design for the gaseous waste management system complies with Appendix I guidelines for external doses to any individual in an unrestricted area as a result of noble gas radionuclides in gaseous effluents. The staff concludes that there is reasonable assurance that ABWRs at sites that have an atmospheric dispersion factor (χ/Q) equal to or

less than $9.8 \times 10^{-6} \text{ sec/m}^3$ will meet the above dose guidelines (.05 mSv (5 mrem) per year to the total body).

Using the assumptions given in BTP ETSB 11-5 for analyzing a postulated leak or failure of a waste gas system and the EAB 0-2 hour χ/Q of $1.37 \times 10^{-3} \text{ sec/m}^3$ (used in Chapter 15 of this report), the staff has determined that the wholebody dose at the EAB is less than 10 percent of the 10 CFR Part 100 limit. Therefore, the staff concludes that for all sites that have equal to or less than the above χ/Q at the EAB, the offgas system design will meet the above dose criterion and will be acceptable. This will be verified for each COL applicant.

These conclusions referred to above are based on the following findings:

- (1) The ABWR design meets the requirements of GDC 60 and 61 by ensuring that the gaseous waste management system includes the equipment and instruments necessary to detect and control the release of radioactive materials in gaseous effluents.
- (2) On the basis of expected radwaste inputs over the life of the plant, the staff has determined the releases of radioactive materials (noble gases, iodines, particulate, tritium and carbon-14) in gaseous effluents resulting from normal operation, including anticipated operational occurrences. The staff used the calculated releases for noble gases (Table 11.2 of this report) to determine the bounding value for χ/Q and assumed a 4-minute decay of the noble gas radionuclides during transit from the release point to the unrestricted area. The staff used the dose models and values for parameters given in RG 1.109 (Rev. 1) to evaluate compliance with Appendix I to 10 CFR Part 50. To calculate the external dose of noble gas radionuclides, the staff assumed a semi-infinite cloud model for the gaseous effluents. For the bounding χ/Q value quoted above, the staff calculated a total body dose (the limiting external dose) of 0.05 mSv/yr (5 mrem/yr), which meets the applicable Appendix I dose guideline.
- (3) The ABWR design meets the requirements of 10 CFR Part 20 because the staff has considered the potential consequences resulting from reactor operation with a postulated fission product release rate consistent with an offgas noble gas release rate of $3.7 \times 10^6 \text{ Bq/MWt-sec}$ ($100 \mu\text{Ci/MWt-sec}$) at 30 minutes decay for a BWR and estimated that, under these conditions, the concentration of radionuclides in gaseous effluents in unrestricted areas with a value of χ/Q that is equal to or less

Radioactive Waste Management

than 9.8×10^{-6} sec/m³ will be below the concentration values in 10 CFR Part 20, Appendix B, Table 2, Column 1. For KR-89 and XE-137, whose specific concentration values are not explicitly given in this table, the staff used relevant concentration values based on ratios of whole-body dose factors of noble gas radionuclides given in RG 1.109 (Rev. 1).

- (4) The staff has considered the capability of the proposed gaseous waste management system to meet the anticipated demands of the plant resulting from anticipated operational occurrences and concludes that the system capacity and flexibility of design are adequate to meet the anticipated needs of the plant. (See item 3 above.)
- (5) The staff reviewed the seismic design criteria including the quality group classifications used for the gaseous waste management system components and the structures housing the radwaste system and concludes that the design of the system and the structures meets the applicable criteria specified in RG 1.143.
- (6) The staff reviewed the provisions incorporated in the ABWR design to control releases resulting from hydrogen explosions in the gaseous waste management system (SSAR Section 11.3, GE submittal dated June 29, 1990). The staff concludes that the features built into the design are adequate to prevent the occurrence of an explosion or adequate to withstand the effects of an explosion in accordance with GDC 3. (See Section 11.3.1 of this report regarding hydrogen recombiners and hydrogen analyzers.)

The staff concludes that the gaseous waste management system for the ABWR meets the acceptance criteria of SRP Section 11.3 and is, therefore, acceptable.

11.4 Solid Waste Management System

11.4.1 System Description and Review Discussion

The solid radioactive waste management system will consist of the equipment and instrumentation necessary to collect, solidify, incinerate, package, and store radioactive wastes resulting from the operation of the reactor water cleanup system, the fuel pool cooling and cleanup system, the suppression pool cleanup system, the condensate polishing system, the liquid radwaste system, the building ventilation systems, the SGTS, the offgas system, and miscellaneous solid wastes (e.g., paper, rags, contaminated

clothing, gloves, shoe coverings) arising from the operation and maintenance of the plant. The solid radwaste management system is located in the radwaste building.

The ABWR solid waste system is designed to process two general types of solid wastes: wet solid wastes, which will be solidified or dewatered before being shipped off site, and dry solid wastes, which will be either incinerated or compacted and/or packaged before being shipped. Combustible dry wastes (e.g., rags, uniforms, paper) will be burned in an incinerator and discharged to an ash storage drum. The offgas from the incinerator will be passed through two ceramic filters in series and a HEPA filter before being released into the atmosphere through a monitored vent. On the basis of GE's submittal dated June 29, 1990, the staff estimates this release to be 592 MBq/yr (0.016 Ci/yr) and to be in particulate form. Incinerated ash will be discharged to an ash storage drum by ash discharge equipment located on the bottom of the incinerator. The description of the incinerator to be used, source of incinerator heat, storage facility for the heat source, and specific fire protection features to prevent any undue fire hazard resulting from incineration, were identified as DFSEER COL Action Item 11.4.1-1. By amended SSAR Section 11.4.3.1 and June 2, 1993 submittal, GE included COL License Information (Section 11.4.3-1, Item 1) which states the COL applicant will provide the above information. The staff finds that GE's identification of the COL License Information pertaining to the incinerator, and testing of the major components of the incinerator and description of the incinerator operation in the SSAR (Section 11.4.2.2.6, and Figure 11.2-2, Sheet 26) are acceptable. GE has also included this action item in the final certified SSAR.

Noncombustible dry solid wastes will be compacted and placed in dry active waste drums for shipment.

There will be two forms of wet wastes: (1) slurries of spent resins and sludges from filters and filter demineralizer backwashes and (2) concentrated wastes from the HCW concentrators of the liquid radwaste treatment system. As stated in Section 11.2.1 of this report, the spent resins and the sludges will be dewatered and the resulting slurry will be loaded in HICs for eventual shipment.

The concentrated waste from the HCW concentrators will be routed through a thin-film dryer for dewatering. The water from this operation will be routed back to the HCW collector tanks. Air will be exhausted through the radwaste building HVAC exhaust. The dewatered,

powdered waste will be pelletized, and the pellets will be mixed with cement glass in drums for eventual offsite shipment or an approved solidification process will be used. Air from the pelletizing and solidification process will be routed to the radwaste building HVAC exhaust via a particle filter and a HEPA filter.

SSAR Interface Requirement 11.4.3.1 stated the first COL applicant will provide detailed information to demonstrate that the wet waste solidification process using cement glass as the solidification agent will result in a product that complies with 10 CFR 61.56. In Open Item 74 in the DSER (SECY-91-235), the staff stated that the interface item should be modified to require all COL applicants to provide this information. After re-reviewing all interface items, GE committed to revise the SSAR to reclassify the need to demonstrate compliance of wet waste solidification product with 10 CFR 61.56, as a COL Action Item. In the DSER, the staff found the GE commitment acceptable and, therefore, identified demonstration of the above compliance as DFSEER COL Action Item 11.4.1-2. By amended SSAR Section 11.4.3.1, GE included COL License Information 11.4.3.1, Item (2) which states the COL applicant will provide detailed information to demonstrate that the wet waste solidification process will result in a product that complies with 10 CFR 61.56. This is acceptable. GE has also included this action item in the final certified SSAR.

On the basis of the Electric Power Research Institute report EPRI-NP-5528, Volume 1 and NUREG/CR-2907 (annual reports for 1986 and 1987, Volumes 7 and 8, only BWRs were considered), the staff estimates the processed wet wastes requiring shipment to be about 370 m³/yr (13,000 ft³/yr) containing approximately 4.4 x 10⁷ MBq (1,200 Ci). The spent resin and filter and filter/demineralizer sludge slurries will be stored in HICs before shipment. The solidified concentrates will be stored in 0.21 m³ (55-gal) drums before shipment. On the basis of NUREG/CR-2907 and GE's submittal dated June 29, 1990, the staff estimates that the processed dry wastes requiring shipment will be about 340 m³/yr (12,000 ft³/yr) containing approximately 4.4 x 10⁵ MBq (12 Ci). However, with incineration of combustible dry wastes, the shipment volume will be less than 340 m³/yr (12,000 ft³/yr). The processed dry wastes will be stored in boxes or in 0.21 m³ (55-gal) drums.

Because the establishment and implementation of a process control program (PCP) for solidifying the evaporator concentrates, using an approved solidification agent, and the dewatering process for the spent resins and filter sludges are dependent on the as-procured equipment for the ABWR standard design, the staff will review the PCP and the dewatering process for each COL applicant against

BTP ETSB 11-3. This was identified as DFSEER COL Action Item 11.4.1-3. By amended SSAR Section 11.4.3.1, GE included COL License Information in Section 11.4.3.1, Item 3 which states the COL applicant will provide a PCP for solidifying the evaporator concentrates using an approved solidification agent and the dewatering process for the spent resins and filter sludges. This is acceptable. GE has also included this action item in the final certified SSAR.

11.4.2 Conclusion

In evaluating the solid radioactive waste management system, the staff considered (1) system design objectives in terms of expected types, volumes, and activities of wastes processed for offsite shipment; (2) provisions for onsite storage of processed solid wastes before shipment; (3) procedures for disposal of incinerated waste; (4) system design to meet acceptance criteria of SRP Section 11.4; and (5) piping and instrumentation diagrams for the system.

On the basis of its review of amended SSAR Section 11.4 and GE's submittals dated June 7, June 29, October 26, and November 5, 1990, August 2, 1991, May 18, 1992, and June 2, 1993, the staff concludes that the solid waste management system design meets the requirements of 10 CFR 20.302(a), and GDC 60, 63, and 64. The design also complies with 10 CFR 61.56 and 10 CFR Part 71. The dewatered resin and filter sludge wastes will be Type B per 10 CFR 61.55 classification. As noted above, they will not be processed to a stable form. They will be stored in HICs to comply with 10 CFR 61.56 and 10 CFR Part 71 requirements. The conclusion on solid waste management system is based on the following findings:

- (1) The design includes equipment and instrumentation for processing, packaging, and storing of radioactive solid wastes before shipment off site. Dedicated radwaste storage areas in the radwaste building can accommodate 221 0.21 m³ (55-gal) drums and 13 boxes with storage capacity of approximately 62 m³ (2200 ft³) of processed wet waste (solidified concentrate) and dry solid wastes. However, before issuing the DSER (SECY-91-235), GE did not specify the capacity and maximum number of HICs that can be stored. Further, the staff was concerned that the dedicated radwaste storage areas mentioned above may not be able to accommodate the HICs before shipment. Therefore, the staff identified this concern as Open Item 75 in the DSER (SECY-91-235). By amended SSAR Chapter 11, GE responded to the above concern in SSAR Section 11.4.2.3.6. From its review, the staff finds that normally the HICs will

Radioactive Waste Management

be shipped promptly after being filled, in the event of a shipping delay they will be stored with shielding in the truck area. SSAR Section 11.4.2.3.6 states that the truck area can accommodate 5 HICs containing approximately 24 m³ (840 ft³) of processed (i.e., dewatered) resins and sludges and the shielding for the HICs. On the basis of the available storage space for processed wet and dry solid wastes given above and the estimates of annual shipment volumes for these wastes given in Section 11.4.1 of this report, the staff concludes that the available storage space is sufficient to accommodate one full offsite waste shipment of dry wastes and 30 days of wet waste at normal generation rate in accordance with BTP ETSB 11-3, Positions B.III.2 and 3. Therefore, Open Item 75 is resolved. The capacities of tanks accumulating spent resins and filter sludges also meet BTP Position B.II.1.

Besides the GE submittals and SSAR revisions discussed above, in a teleconference on May 20, 1992, GE stated that onsite storage of low-level waste beyond that discussed above would be addressed by the COL applicant. Therefore, the staff identified the GE position as DFSEER COL Action Item 11.4.2-1. By amended SSAR Section 11.4.3.1, GE included COL License Information in Section 11.4.3.1, Item 4 which states that the COL applicant will provide a discussion of onsite storage of low-level waste beyond that discussed in the SSAR. This is acceptable. The staff will review the COL applicant's discussion against the guidance provided in Generic Letter 81-38, "Storage of Low-Level Radioactive Wastes at Power Reactor Sites." (This guidance is similar to the one provided in Appendix 11.4-A to SRP Section 11.4.) GE has also included this action item in the final certified SSAR.

- (2) The system will have the capability to process the types and volumes of wastes expected during normal plant operation, including anticipated operational occurrences, in accordance with GDC 60. Provisions for handling wastes meet the requirements of 10 CFR Parts 20. Specifically, the staff has determined that the offgases (resulting from incineration of combustible wastes) that will be exhausted through ceramic filters and a HEPA filter to a monitored vent will have minimal effect on compliance with 10 CFR 20.1302 relating to concentrations of radionuclides in gaseous or airborne effluents in unrestricted areas for sites with a χ/Q equal to or less than 9.8×10^{-6} sec/m³. As stated in Section 11.3.2(3) of this report, the staff

concludes that the subject concentrations will be below the applicable regulatory limit. By identifying the type, the expected quantity, the curie content, and the manner of disposal of the combustible wastes that will be incinerated, GE complies with 10 CFR 20.302 with regard to disposal of incinerator offgases that lies within the ABWR scope.

- (3) The system for monitoring radiation levels and leakage complies with GDC 63 and 64. Radiation monitors at the end of a drum conveyor will monitor the radiation resulting from mixture in the drums and surface contamination of the drums. Devices such as position switches, weight elements, and level sensors will be used to prevent spillage while filling, pouring (solidification of evaporator concentrates), and overfilling the drums. Additionally, safety interlocks provided for the solidification process system will ensure that solidification will be performed only under certain conditions (identified in SSAR Section 11.4.2.2.5). In addition, the effluents resulting from the system inputs to the liquid radwaste and gaseous radwaste management systems are monitored by the respective monitors for these systems (together with other effluents from these systems).
- (4) SSAR Sections 11.4.1.2 and 11.4.2.1 and GE's submittals dated June 7 and 29, 1990, and June 2, 1993, state that the quality group classification, seismic design, and other design features (such as heat tracing concentrate piping and tanks, and flushing connections for all components and piping that contain slurries) meet the guidelines of RG 1.143 and BTP ETSB 11-3, Position B.V.
- (5) The staff finds that the proposed dewatering method for spent resin and filter sludges, namely, treatment by a thin-film dryer or by a vendor-supplied mobile dewatering system, is acceptable. However, as stated in Section 11.4.1 of this report, the staff will review the details demonstrating compliance of the dewatering process and solidification process with applicable positions of BTP ETSB 11-3 and 10 CFR 61.56, on a plant-specific basis for each COL application.
- (6) Since radioactive material packaging is within the scope of the COL applicant, GE has included COL License Information in SSAR Section 11.4.3.1, Item (5) which states that the COL applicant will demonstrate the compliance of all radioactive waste shipping packages with 10 CFR Part 71 requirements for packaging such wastes. The staff

finds this acceptable and will review compliance with 10 CFR Part 71 on a plant-specific basis for each COL application.

- (7) As part of Open Item 76 in the DSER (SECY-91-235), the staff stated that GE should identify the specific fire protection features available in the applicable area to prevent any undue fire hazard resulting from incineration. By amended SSAR Section 11.4.3.1, GE provided COL License Information in SSAR Section 11.4.3.1, Item 1 which addresses the issue raised in this part of Open Item 76. As discussed in Section 11.4.1 of this report, the staff finds this COL License Information acceptable. Therefore, part of Open Item 76 is resolved.
- (8) In the DSER (SECY-91-235), the staff stated that SSAR Section 11.4.2.3.5 (last paragraph), Table 11.4-2, and the response to Question 430.171 were inconsistent and confusing. For example, the section stated that Table 11.4-2 represents the shipped volume of solid wastes and that Table 11.4-3 gives the corresponding curie content; however, the two tables could not be correlated since Table 11.4-2 gave the shipped volume only for the solidified concentrates (not the total volume of all solid wastes) and Table 11.4-3 gave curie content for the spent resin and filter sludges. Therefore, the staff requested GE to correct this section and Table 11.4-2 (Open Item 76). GE provided the requested information in Table 11.4-3. This information provides the volumes of various kinds of solid wastes expected to be shipped annually and their corresponding total curie content. Therefore, this part of Open Item 76, in the DSER (SECY-91-235), is resolved. Therefore, this resolved both parts of Open Item 76.

On the basis of these findings, the staff concludes that the solid radwaste management system design for the ABWR meets the acceptance criteria of SRP Section 11.4 and complies with 10 CFR 61.56 and is, therefore, acceptable.

11.5 Process and Effluent Radiological Monitoring and Sampling Systems

11.5.1 System Description and Review Discussion

The process and effluent radiological monitoring systems are designed to provide information about radioactivity levels in systems throughout the plant, indicate radioactive leakage between systems, monitor equipment performance,

and monitor and control radioactivity levels in plant discharges to the environs.

On the basis of GE's telefax dated May 26, 1993, its submittal dated June 23, 1993, and SSAR Section 11.5.2.2.4, the staff finds that all airborne radioactive releases from the plant to the environment except the offgases from the incinerator to the environment will be exhausted through the plant vent. The major sources that will be combined and routed to the plant vent are the offgas exhaust, the radwaste building exhaust, the RB (secondary containment) exhaust, the service building controlled area exhaust, and the turbine building exhaust, which includes the gland seal system and the mechanical vacuum pump exhausts. A radiation-monitoring system (RMS) will monitor the plant vent discharge for gross radiation level and collect halogen and particulate samples. The offgases from the incinerator will be monitored and released to the environs via the incinerator exhaust stack. Besides the main plant vent gaseous effluent monitor and samplers, as indicated in ABWR SSAR Table 11.5-1, radiation monitors will be provided for monitoring the offgas post-treatment exhaust, TGSS exhaust, RB (secondary containment) exhaust, radwaste building vent exhaust, and turbine building exhaust. These monitors will be used to identify sources of airborne activity before mixing in the main plant vent. Gaseous process stream monitoring will include the offgas pretreatment RM, the carbon bed vault RM, and the control rod drive maintenance area exhaust RM.

The liquid effluent and process RMS include the liquid radwaste effluent and the RB closed cooling water system RMS.

The RMs, which will monitor the discharges from the gaseous and liquid radwaste treatment systems (i.e., offgas posttreatment effluent and processed liquid radwaste effluent), are designed to alarm and provide a signal to automatically close the waste discharge valve of the affected treatment system before exceeding the normal operation limits. The radiation monitor for the incinerator offgas discharge is also designed to alarm and initiate automatic termination of the exhaust to the environs before the radiation level in the exhaust exceeds the technical specification (TS) limit. The DSER stated that the normal operation limits will be specified in the ABWR Radiological Effluent Technical Specifications (RETS) and, therefore, identified it as TS Item 11.5.1-1. However, GL 89-01 allows the RETS to be relocated in the Offsite Dose Calculation Manual (ODCM) which is plant specific. Since the requirement for the ODCM will be included in Section 5.0 (Administrative Controls) of the plant specific TSs to be provided by the COL applicant, TS Item 11.5.1-1 is resolved. Before being discharged from

Radioactive Waste Management

the radwaste treatment systems, liquid in the tanks will be sampled and analyzed. Release and dilution rates will be specified on the basis of the results of these analyses.

In addition to the gaseous and liquid effluent and processing RMS, there are systems required to initiate appropriate protective action in case of postulated accidents. These systems include the main steamline RMS, the fuel area (of the reactor building) ventilation exhaust RMS, the control building HVAC RMS, the SGTS RMS (the exhaust from this system goes to the plant stack), and the containment space-refueling mode RMS.

The ABWR design includes provisions for grab sampling and analysis of liquid sources (e.g., reactor coolant crud and filtrate, liquid radwaste system tanks, condensate storage tank, reactor building cooling water (RCW) system, reactor water cleanup system) and both liquid and gaseous effluent and process streams, including reactor service water and the circulating water system decant line for determination of gross radiation level and identity and quantity of specific radionuclides in the applicable stream or source. The stream or the source sampled, the parameters analyzed, the analysis frequency, and the sensitivity for analysis are listed in SSAR Tables 11.5-4 through 11.5-7.

The ABWR design includes accident monitoring instrumentation for monitoring noble gases, iodines, and particulate in gaseous or airborne effluent streams during an accident. As stated in SSAR Sections 1.A.2.15 and Section 7.5, GE considers that such instrumentation provided for the ABWR is generally in accordance with RG 1.97, Revision 3, and that, therefore, it meets the guidelines of NUREG-0737, Action Item II.F.1, Attachments 1 and 2, which is incorporated into 10 CFR 50.34(f)(2)(xvii). See the detailed discussion of this item in the following section.

11.5.2 Conclusion

The staff reviewed the amended SSAR Section 11.5 and GE's submittals dated June 2, 1989, February 28, 1990, December 19, 1991, and June 23, 1993, and GE's telefaxes dated May 26 and September 24, 1993, with regard to the process and effluent radiological monitoring and sampling systems for the ABWR. The review included piping and instrumentation diagrams (P&IDs) for the liquid and gaseous radwaste systems; SSAR Tables 11.5-1 through 11.5-7, which list the liquid and gaseous process and effluent RMSs and summaries of radiological analysis for liquid and gaseous process and effluent stream samples; information provided in SSAR Section 7.5.1.1 and Tables 7.5-1 and 7.5-2 (i.e., tables

comparing ABWR design provisions for monitoring radioactive gaseous effluents during an accident with applicable RG 1.97 guidelines); and descriptions of the various building ventilation systems, the main condenser evacuation system, and the TGSS in so far as they relate to the radiation-monitoring provisions for these systems. However, in Open Item 77 in the DSER (SECY-91-235), the staff stated it could not review the location of the monitoring points relative to the effluent release points in the gaseous effluent streams because GE had not provided the corresponding P&IDs. Subsequently, GE submitted P&IDs that identify the location of monitoring points relative to the effluent release points. Therefore, Open Item 77 in the DSER (SECY-91-235), is resolved.

As stated at the beginning of Chapter 11 of this report, the staff will review specific compliance of the COL sampling program and quality assurance (QA) for radiological monitoring programs with ANSI N13.1 and RGs 1.21 and 4.15 on a plant-specific basis for each COL applicant. During such a plant-specific review, the staff will evaluate the plant-specific features and programs provided by the COL applicant to address the issue discussed in IE Bulletin 80-10 of possible contamination of nonradioactive systems and the resulting potential for unmonitored, uncontrolled release to the environment. Therefore, the staff's review of the sampling and analysis program for the ABWR standard design is limited to the identification of the streams required to be sampled, monitored and controlled.

The staff concludes that the design of the ABWR process and effluent radiological monitoring and sampling systems complies with the requirements (1) 10 CFR 20.1302 relating to radioactivity monitoring of effluents to unrestricted areas, (2) of GDC 60 that radioactive waste management systems be designed to control release of radioactive materials to the environment, and (3) of GDC 63 and 64 that radioactive waste management systems be designed to monitor radiation levels, leakage and releases to environment. The staff's conclusion is based on the following findings:

- (1) The design includes provisions for monitoring the radioactivity of effluents to unrestricted areas. The exhausts from certain areas of the RB (served by the safety related RB electrical equipment and safety related RB diesel generator HVAC subsystems) and the turbine island (served by the electrical building ventilation system) will be directly released to the environs unmonitored. In addition to the above areas, the staff stated in Open Item 78 in the DSER (SECY-91-235) that the exhausts from the battery rooms and lube oil area in the turbine island and the reactor internal pump

(RIP) control panel room in the reactor building were unmonitored. However, the SSAR currently shows the lube oil exhaust being monitored at the plant vent and does not show a battery room in the turbine island (Figures 9.4-2a and 2b). The SSAR also shows the RB RIP ASD HVAC system (RIP control panel room HVAC system renamed) as a closed cooling HVAC system with no outside air supply to the room or exhaust from the room to the environs (Figure 9.4-5). Further, by telefax dated June 9, 1992, GE proposed to revise Section 11.5.2.2.4 to state that the exhausts from the areas serviced by the HVAC systems mentioned above are not monitored since the subject areas do not contain any radioactive systems and that the only releases to the environment by these systems would first have to be brought into the areas by their own HVAC system's supply fans. On the basis of the above telefax design information, the DFSEER stated that lack of radiation monitoring of certain exhausts identified above is acceptable since GDC 64 does not require radiation monitoring of plant exhausts to the environs that do not have potential to be radioactive. Therefore, in the DFSEER, the staff re-classified this part of DSER (SECY 91-235) Open Item 78 as DFSEER Confirmatory Item 11.5.2-1 and required GE to incorporate the telefax design information in the applicable SSAR section. By submittal dated June 23, 1993, GE incorporated the subject information as a footnote to SSAR Section 11.5.2.2.4 after revising the subject information to include also the exhaust from the service building clean area since the exhaust from this area also qualifies for non-monitoring as explained above. In the above submittal, GE also stated that the exhaust from the area served by the control building essential electrical HVAC system will not be monitored prior to discharge to the environs though it contains (RCW) system components. This is because, the RCW, which is considered as a clean water system, is monitored to alarm at any radiation level in the system above background from potential leakage sources. Such contamination will require dumping the cooling water to radwaste treatment and replacing it with clean water. Therefore, the system will remain clean. Furthermore, at the system operating temperature below 35 °C (95 °F), airborne evolution of radioactivity from the cooling water system will be negligible. The staff has reviewed the above GE's justification for not monitoring this exhaust prior to its release to the environs and finds the justification, acceptable. GE has also included this information in the final certified SSAR. Therefore, Confirmatory Item 11.5.2-1 is resolved.

The staff further stated as part of Open Item 78 in the DSER (SECY-91-235) that GE had not provided sufficient information in its submittal dated April 26, 1991, to clarify that the service building ventilation system exhaust will be routed through the plant vent where radiation monitoring occurs. By amended SSAR Section 11.5.2.2.4, GE clarified that the service building controlled area ventilation system exhaust will be routed through the plant vent, where radiation monitoring occurs. This is acceptable. Therefore, this part of DSER (SECY-91-235) Open Item 78 is resolved.

- (2) The design includes provisions for monitoring process streams (e.g., offgas post-treatment exhaust, secondary containment exhaust, RCW system) and provisions for initiating appropriate action in case of postulated accidents. Automatic control features include termination of liquid effluent release, incinerator offgas release, and the offgas system release as appropriate, when the preset radiation level for the applicable stream is exceeded. The automatic control features also include securing the normal secondary containment ventilation system and initiation of the SGTS under certain conditions identified in the SSAR. Also, since GE indicated a single monitor for the gland seal exhaust and mechanical vacuum pump exhaust, and totally clean steam will not normally be supplied for sealing the TGSS, the staff required in the DSER (SECY-91-235) that plant procedures include manual switchover to the auxiliary (backup clean) steam source whenever the monitor indicates that the exhaust stream concentration exceeds a preset level. As discussed in Section 10.4.3 of this report, GE has committed to provide a COL action item stating that the COL applicant will provide the necessary procedures for switchover to the auxiliary steam system when monitored radiation level in the TGSS exhaust exceeds an acceptable preset level. The staff identified the development of the procedures as DFSEER COL Action Item 11.5.2-1. By amended SSAR Section 10.4.10, GE identified COL License Information in Section 10.4.10 for the TGSS effluents. The subject information among other requirements, spells out the requirement for switchover mentioned above. The staff agrees with GE's approach for providing sealing steam to the TGSS. GE has also included this action item in the final certified SSAR.

As part of Open Item 78 in the DSER (SECY-91-235), the staff questioned whether the secondary containment ventilation exhaust monitor will be sensitive enough to detect high-radiation level in the primary containment

Radioactive Waste Management

purge exhaust. In response to the above concern, GE has revised Section 11.5.2.1.2 to state that the detectors in the secondary containment ventilation exhaust will be sensitive enough to detect high-radiation levels during primary containment purge to alert the operator and to initiate appropriate measures. This part of Open Item 78 is resolved.

- (3) As stated earlier in this section, the design includes provisions for sampling and analysis of radioiodines, particulates, and tritium in the process and effluent streams (for tritium, only in the effluent stream). However, as part of Open Item 78 in the DSER (SECY-91-235), the staff noted that SSAR Table 11.5-5 did not include grab sampling and analysis provisions for the gland seal process stream. SSAR Table 11.5-7 also did not include sampling and analysis provisions for the plant stack exhaust. In response to the above concern, GE added the sampling and analysis provisions for the gland steam condenser and plant stack exhausts (SSAR Table 11.5-7). Therefore, this part of Open Item 78 is resolved.
- (4) SSAR Tables 7.5-1 and 7.5-2 provide design and qualification criteria for accident-monitoring instrumentation and the concentration ranges covered by the instrumentation. The staff finds that the design complies with RG 1.97 with regard to ranges and design and qualification criteria. However, as part of Open Item 78 in the DSER (SECY-91-235), the staff stated that neither these tables nor SSAR Section 11.5 contained sufficient information and that the following information was to be provided:
- type of instrumentation to be used, including the calibration frequency and technique
 - monitoring locations (or points of sampling), including description of methods used to ensure representative measurements and background correction (the P&IDs for building ventilation systems were not provided to determine monitoring locations relative to the applicable release points for the gaseous effluent streams)
 - location of instrument readout(s) and method of recording, including description of the method or procedure for transmitting or disseminating the data

- assurance of capability to obtain readings at least every 15 minutes during and following an accident
- description of procedures or calculational methods to be used for converting instrument readings to release rates per unit time, based on exhaust air flow and considering radionuclide spectrum distribution as a function of time after shutdown
- description of the sampling system design, including the sampling medium to demonstrate how the design meets the requirements identified in Clarification 2 of NUREG-0737, page II.F.1-7
- description of the sampling technique to be used under accident conditions to demonstrate how the technique meets the requirements identified in Clarification 3 of NUREG-0737, pages II.F.1-7 and II.F.1-8
- description of the sampling technique to ensure the system capability to collect and analyze or measure representative samples of radioactive iodines and particulate in plant gaseous effluents during and following an accident as identified in Table II.F.1-2 of NUREG-0737, page II.F.1-9

In response to the above request for information, GE provided COL License Information 11.5.6.1 through 11.5.6.5. The COL license information calls for the COL applicant to provide an operation and maintenance manual that describes or demonstrates (as appropriate) the following: calculation of radiation release rates from radiation measurements, sampling system design and its compliance with the shielding requirements identified in Clarification 2 of Attachment 2 to TMI Item II.F.1 of NUREG-0737, sampling technique and its compliance with the requirements identified in Clarification 3 of Attachment 2 to TMI Item II.F.1 of NUREG-0737, collection technique for extracting representative samples of radioiodines and particulates and calibration frequencies and techniques for the radiation sensors. The staff concludes that this part of Open Item 78 is resolved.

On the basis of these findings, the staff concludes that the design of the ABWR process and effluent radiological and sampling systems meets the acceptance criteria of SRP Section 11.5 and is, therefore, acceptable.

12 RADIATION PROTECTION

Standard safety analysis report (SSAR) Chapter 12 provides information on the radiation protection features and estimated occupation exposure associated with the advanced boiling water reactor (ABWR) design. The radiation protection measures for the ABWR are intended to ensure that internal and external occupational radiation exposures to plant personnel, contractors, and the general population, as a result of plant operations, including shutdown periods and anticipated operational occurrences (AOOs), will be within applicable limits of regulatory criteria and will be as low as is reasonably achievable (ALARA). The staff reviewed the SSAR for completeness against the guidelines of Regulatory Guide (RG) 1.70 (Rev. 3), "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," and against the criteria of Sections 12.1 through 12.5 of NUREG-0800, the standard review plan (SRP).

The staff reviewed GE's SSAR and supplemental information to determine if the ABWR design is sufficient to permit plant operations while maintaining radiation doses to personnel within the limits of 10 CFR Part 20, and if the design features are consistent with the guidelines of RG 8.8 (Rev. 3) "Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations will be As Low As Is Reasonably Achievable." On May 21, 1991, the Commission issued a revision of 10 CFR Part 20 that changed the system of radiation dose limitation. The previous occupational dose limit for whole-body radiation exposure was 12.5 mSv (1.25 rem) per quarter year with a provision to extend it to 30 mSv (3 rem) per year. The new limit is 50 mSv (5 rem) per year with a provision to extend it to 100 mSv (10 rem) per quarter year. The previous 10 CFR Part 20 limits for doses from licensed radioactive material inside the body (deposited through injection, absorption, ingestion, or inhalation) were separate from the dose limits for exposure to licensed sources outside the body. The new Part 20 limits the sum of the external whole-body dose (deep dose equivalent) and the committed effective equivalent doses resulting from radioactive material deposited inside the body. In addition, the new Part 20 requires that this sum (the total effective dose equivalent) be maintained ALARA for each individual. These changes to the regulation do not affect the acceptance criteria used by the staff to review the ABWR design. The SRP acceptance criteria provide assurance that the radiation doses resulting from exposure to licensed radioactive sources outside the body and inside the body can each be maintained well within the limits of 10 CFR Part 20 and ALARA. The balancing of internal and external exposure necessary to ensure that their sum is ALARA is an operational concern that will be reviewed in conjunction with a combined license (COL) application. The Part 20, as amended, contains a number of new programmatic requirements that do not affect plant

design. Programmatic and operational radiation protection concerns will be addressed by the COL applicant.

The staff finds that the radiation protection measures incorporated in the ABWR design will provide reasonable assurance that occupational doses can be maintained ALARA and below the limits of 10 CFR Part 20 during all plant operations.

12.1 Ensuring That Occupational Radiation Doses Are As Low As Is Reasonably Achievable

The staff reviewed the ABWR design to ensure that GE had either committed to following the criteria of the RGs and staff positions referenced in SRP Section 12.1 or provided acceptable alternatives.

12.1.1 Policy Considerations

SSAR Section 12.1.1 describes GE's policy to ensure that ALARA considerations are factored into each stage of the ABWR design process. GE committed to ensure that the ABWR will be designed and constructed in a manner consistent with RG 8.8 (Rev. 3). The ALARA policy was applied through detailed engineering reviews and design modifications to ensure that the resulting plant design can maintain radiation exposures ALARA. This policy is consistent with the guidelines of RG 8.8 and is acceptable.

The policy considerations regarding plant operations contained in RG 8.8 (Rev. 3), RG 1.8 (Rev. 2) "Qualification and Training of Personnel for Nuclear Power Plants," and RG 8.10 (Rev. 1), "Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As Is Reasonably Achievable," are outside the scope of this review of the ABWR design. The COL applicant referencing the ABWR design will address these operational policy considerations to ensure that radiation doses are ALARA. This was identified as draft final safety evaluation (DFSER) COL Action Item 12.1.1-1. In Amendment 23 to the SSAR, GE revised Section 12.1.1 to clarify the policy considerations that will be addressed by the COL applicant. This is acceptable to the staff.

12.1.2 Design Considerations

The ABWR design aims to minimize the costs, both in terms of maintenance time and radiation exposure, associated with plant operation. The ABWR design employs features that will (1) eliminate the need for certain maintenance, (2) facilitate the required maintenance, and (3) minimize the sources of radiation exposure in the plant.

Radiation Protection

The ABWR design includes several design features consistent with the guidelines in RG 8.8 (Rev. 3). The plant layout has shielded rooms or cubicles for components that are the source of high radiation levels. Components in redundant systems are located in separate shielded rooms or cubicles so that radiation levels associated with an operating train of equipment will not significantly contribute to the radiological hazards associated with performing maintenance on the redundant train. The design of shielded rooms or cubicles have labyrinth access ways to reduce scattered radiation in adjacent areas. Removable shielded walls or hatches will be provided if space limitations in a room or cubicle prevent an adequate laydown area for maintenance of equipment. Appropriate use of remote operators and instrumentation will minimize the need to enter shielded rooms or cubicles. The remote back-flushing capability for plant filters and demineralizers will use gravity drains and piping that slope toward the backwash tank to minimize traps that would become radiation hot spots. Grafoil valve stem packing will reduce leakage of contaminated water from reactor systems and will minimize the need to repack the stems of these valves. These design features are consistent with the guidance in RG 8.8 (Rev. 3).

In addition to these design features, operational experience with previous boiling water reactor (BWR) designs has been factored into the ABWR design in several areas. Many unique ABWR features, designed to eliminate difficulties encountered in operating current BWRs, should also reduce occupational radiation exposure. An example is the elimination of reactor coolant recirculation piping inside primary containment. Several BWRs have experienced significant stress corrosion cracking, requiring replacement of this piping at the cost of thousands of person-rem radiation doses. Eliminating the coolant recirculation piping from the ABWR reactor not only eliminates the radiation exposure associated with recirculation pipe inspection and replacement but should also reduce the source of radiation in the drywell, thus reducing exposure to personnel performing other maintenance activities in the primary containment.

Other examples of design features that will reduce radiation exposure include the control rod drive (CRD) mechanism, layout of the lower drywell, and safety relief valve (SRV) design and layout. Current BWRs have external restraints on CRDs to prevent a rod ejection in the event of a CRD housing failure, which have to be cleared out of the way during CRD maintenance. The internal CRD restraint feature in the ABWR design will allow easier CRD removal and reduce radiation exposure associated with CRD maintenance. The arrangement of the lower drywell will also reduce radiation exposures during CRD maintenance by allowing easy access to the

lower reactor vessel head for CRD and reactor internal pump (RIP) removal. A transport system is also provided to remove CRDs and RIPs from the drywell so that maintenance can be performed in a lower radiation area. Direct-action SRVs that require less maintenance than current pilot-operated valves are provided in the ABWR design. These SRVs are placed around the outside of the reactor vessel and have a dedicated hoist to facilitate maintenance.

In the draft safety evaluation report (DSER) (SECY-91-355), the staff identified two areas where the current BWR operating experience was not adequately accounted for in the ABWR design. These areas are the dose rates in the upper drywell during the transfer of irradiated spent fuel assemblies (SFA) (Open Item 112) and exposures resulting from a complete withdrawal of the traversing incore probe (TIP) (Open Item 106).

In a March 26, 1992, SSAR markup, GE provided additional information concerning the radiation protection design features of the ABWR TIP system in response to Open Item 106. These features include a shielded room for the TIP drive units and a separate shielded room for the parked TIP. Additional shielding is provided for the parked TIP and its drive cable to allow personnel to enter this room when the TIP is out of the reactor. The TIP drive units also have an electro-mechanical switch that will cut power to their drive spooler to prevent the activated portions of the TIP from being completely withdrawn into the drive housings. These features are designed such that radiation exposures resulting from TIP operations, and related abnormal AOOs, can be maintained ALARA. This was identified as DFSER Confirmatory Item 12.1.2-1. GE has also included this information in Amendment 20 of the SSAR and the staff finds it to be acceptable.

The potential for creating extremely high dose rates in the upper drywell during spent fuel handling operations and the potential for high dose rates around unshielded portions of the TIP conduit are discussed in Section 12.3.2 of this report.

12.1.3 Operational Considerations

Operational considerations regarding the implementation of a radiation protection program are outside the scope of this design certification review. The COL applicant referencing the ABWR design will address these operational considerations to the level of detail provided in RG 1.70 (Rev. 3). This was identified as DFSER COL Action Item 12.1.3-1. In Amendment 23, GE revised Section 12.1.3 of the SSAR to identify these operational considerations as an area to be addressed by the COL applicant. The staff finds it to be acceptable.

12.1.4 COL License Information

Section 12.1.4 of the DSER (SECY-91-355) identified three issues concerning compliance with RGs 8.10 (Rev. 1) and 1.8 (Rev. 2), and procedures for keeping occupational exposures ALARA, as outside the scope of this review. This was identified as DFSEER COL Action Item 12.1.4-1. In Amendment 20 to the SSAR, GE revised Section 12.1.4 to properly characterize these issues. The staff finds it to be acceptable.

In Open Item F1.9-1, the staff identified the need for GE to include a COL action item related to the use of appropriate materials in the ABWR design which would reduce the potential for personnel exposures. GE provided a submittal dated February 7, 1994, which included a markup of SSAR Section 12.3.7.4, that added a COL action item stating that the applicant, following the design commitments included in SSAR Section 12.3.1.1.2, is responsible for material selection to ensure that radiation exposures are ALARA. The staff found this commitment for a COL action item to be acceptable.

12.2 Radiation Sources

The staff has audited the contained source terms and airborne radioactive material source terms in Section 12.2 and Chapter 11 of the ABWR SSAR for completeness against the guidelines in RG 1.70, (Rev. 3), and against the criteria set forth in Section 12.2 of SRP. The contained source terms are used as the basis for designing radiation protection features (including radiation shielding) and for personnel dose assessment. Airborne radioactive source terms are used in the design of ventilation systems and personnel dose assessment. The staff reviewed the source terms in the SSAR to ensure that GE had either committed to following the criteria of RGs and staff positions contained in SRP Section 12.2 or provided acceptable alternatives. In addition, the staff selectively compared source terms for specific systems against those used for plants of similar design. The staff finds that source term descriptions in the SSAR are not adequate to meet the criteria of RG 1.70 (Rev. 3) and NUREG-0800.

At the current stage of the ABWR design, GE does not have the specifications for the as-built systems or the as-procured hardware that would be available for a completed plant. Therefore, GE cannot describe the radioactive system components, which will be significant in-plant radiation sources, to the level of detail specified in RG 1.70 and the SRP. Although these details, such as radioactivity content, source geometry, equipment leakage, and plant location, are needed for the staff to verify the adequacy of the radiation shielding, ventilation and airborne radioactivity monitoring systems, the staff has

determined that providing this information goes beyond the design requirements specified in 10 CFR Part 52. As an alternative, GE has provided a set of design acceptance criteria (DAC) that, if met, will verify the adequacy of the ABWR shielding design, plant ventilation design, and the design of the airborne radioactivity monitoring systems. Compliance with these DAC, as with other inspections, tests, analyses, and acceptance criteria (ITAAC), would be verified during plant construction prior to loading fuel into the reactor. The DAC in Table 3.2.a of the ABWR certified design material (CDM) specify the methods and assumptions for verifying the shielding design, including those for estimating the source terms to be used in the shielding analysis. Similarly, the DAC in Table 3.2.b of the CDM specify the acceptance methods for determining the airborne radioactivity concentrations used to verify the adequacy of the ventilation and airborne monitoring system designs. This alternative is acceptable to the staff.

12.2.1 Contained Sources and Airborne Radioactive Material Sources

GE describes radioactive sources in the ABWR design is contained in SSAR Chapters 11 and 12. Section 11.1 provides information on the radioactive source terms in reactor water and steam. Section 12.2 provides descriptions of plant components that will become significant sources of radiation during plant operations, including shutdown, and sources of airborne radioactive material.

During power operations, the greatest potential for personnel radiation dose is inside the primary containment drywell from nitrogen-16, noble gases, reactor neutrons, and prompt gammas. The steam and condensate systems outside the drywell are also significant sources of radiation because nitrogen-16 is generated during power operations. In other areas outside of the drywell, and inside the drywell after shutdown, the primary sources of personnel radiation exposure are the fission products in the coolant from fuel cladding defects and the activation products transported to and deposited in plant systems and components. Tables 12.1-7 through 12.2-30 in the SSAR list the radioactivity (or source terms) for typical components. These source terms are based on the assumed component geometry and locations listed in Table 12.2-5.

The estimates of concentrations of fission and activation products in the ABWR systems containing reactor water are based on American National Standards Institute/American Nuclear Society (ANSI/ANS)-18.1 "Radioactive Source Term for Normal Operation of LWRs" (1984), adjusted using the assumptions in RG 1.112 (Rev. 0), "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Power Reactors." Allowances

Radiation Protection

are included for the buildup of activation products resulting from corrosion and wear on the basis of operating experience of reactors of similar design. Neutron and prompt gamma source terms are based on reactor core physics calculations. The source terms used to determine in-plant post-accident radiation levels meet the provisions in RG 1.3, "Assumptions used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors," as further discussed in Item II.B.2 of NUREG-0737, "Clarification of the TMI Action Plan Requirements."

In Open Item 107 in the DSER (SECY-91-355), the staff identified several deficiencies in GE's description of the contained radioactive source terms for the ABWR: sources inside the drywell and in the turbine building were omitted, sources in vital areas after an accident were not described, and source characterization was insufficient. GE amended SSAR Section 12.3.5 to indicate that the post-accident sources of concern in plant vital areas are limited to gamma radiation shine from the reactor building and the radioactive material in the post-accident coolant and effluent monitoring systems. GE also amended the tables of source terms in Section 12.2 to include sources inside the drywell and turbine building. SSAR Tables 12.2-7 through 12.2-30 provided nominal source strengths on the basis of expected system configuration and approximate component geometry. As discussed in Section 12.2 of this report, the actual source terms used in confirmatory shielding calculations will be determined as required by the DAC in Table 3.2.a of the CDM. The staff considers this item resolved.

Almost all of the airborne radioactivity within the plant results from equipment leakage. As discussed below, the leakage of contaminated fluids from system components cannot be quantified at this stage in the ABWR design. GE has proposed DAC in Table 3.2.b of the CDM to ensure that the airborne source terms in each room and operating area of the plant are calculated prior to fuel load. The lack of airborne source term description in the SSAR was also identified as Open Item 107 in the DSER (SECY-91-355). On the basis of the following discussion of DAC, the staff considers this item resolved.

12.2.2 Certified Design Material

As discussed in Section 12.2 of this report, the SSAR does not provide system layouts within rooms or cubicles or information about the type and size of components in these systems. Without this as-built or as-procured information, source term parameters needed to calculate radiation shielding for these systems cannot be provided as specified in the SRP. Similarly, since leakage characteristics of this unidentified equipment are not known, the concentrations

of airborne radioactive material in equipment rooms or cubicles cannot be provided. As an alternative, GE provided DAC that require the COL applicant to determine source term parameters that will be verified during plant construction. DAC are discussed in Section 12.3.5 of this report. DAC describing the bases for the source term are consistent with the SRP acceptance criteria. Compliance with these DAC, supplemented by the information in SSAR Sections 12.2 and 12.3, is acceptable to adequately address the requirement to identify the kinds and quantities of radioactive materials expected to be produced by plant operation in 10 CFR 50.34(b)(3) and will ensure that the appropriate source terms (as supplemented by the guidance of RG 1.112 (Rev. 0), NUREG-0737, and ANSI/ANS 18.1) are used to demonstrate that the ABWR design meets the relevant requirements in 10 CFR Part 20 concerning the limitation of radiation dose to personnel; 10 CFR 50.34(f) and GDC 19 with respect to operator access to plant areas during and following a reactor accident; and GDC 61 regarding adequate shielding, containment and confinement of fuel storage and handling, radioactive waste, and other systems which may contain radioactivity. The adequacy and acceptability of the ABWR design descriptions and ITAAC (including DAC) are evaluated in Section 14.3 of this report.

12.2.3 COL License Information

In the DSER, the staff stated that two items were mischaracterized as interface items by the applicant: (1) the compliance with 10 CFR Parts 20 and 50 in Section 12.2.2.3 and (2) the determination of gamma shine from the turbine building in Section 12.2.1.3 of the SSAR. These SSAR sections referenced Section 12.2.3, which identified the issues as design interfaces. This was DSER Confirmatory Item 12.2.3-1. In an SSAR markup of Chapter 12 dated April 16, 1993, GE deleted the discussion of interfaces in Sections 12.2.1.3, 12.2.2, and 12.2.3. Section 12.2.3 has been revised to identify the issues regarding compliance with 10 CFR Parts 20 and 50 as COL license information. As indicated in Section 12.2.2.4 of the SSAR, gamma shine from the turbine building is addressed in the DAC listed in Table 3.7 (see Section 12.3.5.1 of this report). GE has included this information in the SSAR. This change is acceptable. Therefore, this confirmatory item is resolved.

12.3 Radiation Protection Design

The staff has audited the facility design features in the SSAR, including the shielding, the ventilation, and the radiation and airborne radioactivity monitoring instrumentation for completeness against the guidelines in RG 1.70 (Rev. 3) and against the criteria set forth in SRP Section 12.3. The staff reviewed these design features to

ensure that GE had either committed to following the criteria of RGs and staff positions referenced in SRP Section 12.3 or provided acceptable alternatives.

The staff concludes that GE has demonstrated that the ABWR design can meet the relevant requirements of 10 CFR Parts 20 and 50 and GDC 19 and 61 in all areas of the plant, as set forth below.

12.3.1 Facility Design Features

Several features in the ABWR design will help ALARA radiation doses associated with tasks such as maintenance, refueling, radioactive material handling, in-service inspection, decommissioning, and accident recovery (see Section 12.1.2 of this report). These features will facilitate access to work areas, reduce or allow the reduction of radioactive source intensity, reduce the occupancy requirements in high radiation fields, and provide for portable shielding, and remote-operation and instrumentation of radioactive systems. These ABWR features are consistent with the guidance of RG 8.8 (Rev. 3) and the SRP and are acceptable.

GE's drawings of the plant layout indicate six radiation zones which are the basis for occupancy and access restrictions for various areas within the plant during normal operations and accident conditions. Maximum design dose rates are established for each zone and are used as the basis for shielding each zone. This method of plant zoning is consistent with the guidance in RG 1.70 (Rev. 3) and the SRP and is acceptable.

In Open Items 108 through 110 in the DSER (SECY-91-355), the staff identified several deficiencies in SSAR Chapter 12 Figures 12.3-1 through 12.3-73, which depict plant radiation zones (during normal operations, normal shutdown, and accident conditions) and area radiation monitor locations. This was identified as DFSER Confirmatory Item 12.3.1-1. Amendments 21 and 22 to the SSAR provided more legible figures for the reactor, control, and radwaste buildings and resolved the discrepancies noted in the figures of the turbine building. These updated figures also indicate the normal controlled and uncontrolled access routes to the plant as well as the access and egress routes to and from plant vital areas under accident conditions. In response to a staff question, GE acknowledged the radiation zone designation above the spent fuel pool (from greater than 1.0 mSv/hr (100 mrem/hr)) was an error. GE revised the radiation zone designation for this area to less than 0.05 mSv/hr (5 mrem/hr). This dose rate for the area above the spent fuel pool is consistent with industry experience and, therefore, is acceptable to the staff. GE has also included

this information in the SSAR and the staff finds it to be acceptable.

The buildup of activation products from corrosion and wear is a major contributor to occupational radiation doses. As discussed in Section 12.1.4 of this report, the COL applicant is responsible for material selection to ensure that radiation exposures are ALARA. Design features provided in the SSAR to minimize exposure sources include a reduction of cobalt-bearing components used in reactor systems (activated cobalt is a major contributor to plant radiation levels) and pre-filming (establishing a corrosion resistant layer on internal surfaces) of reactor systems before plant operation to minimize activated material deposition on system interior surfaces. Main condenser tubes and tube sheets will be made of titanium alloys to minimize the introduction of foreign material (which become activated and/or promote corrosion) into the reactor system as a result of condenser tube leakage. Other features, such as the use of seamless piping, straight-through valve design wherever possible, butt-welded piping connections, and back-flushing connections on instrument lines, will minimize buildup of radioactivity in plant piping systems.

In Open Item 111 in the DSER (SECY-91-355) concerned the provision in the ABWR design to facilitate chemical decontamination of heat exchangers in systems that carry radioactive water. The staff identified DFSER Confirmatory Item 12.3.1-2 on the basis of an April 9, 1992, draft SSAR amendment. Subsequently, Amendment 20 to the SSAR stated that the reactor water cleanup (CUW) non-regenerative and regenerative heat exchangers have separate decontamination connections. Heat exchangers in the residual heat removal (RHR) system and the heat exchangers for RIP cooling have fittings that will allow flushing with clean water. GE's corrosion product control features are consistent with the guidance in RG 8.8 (Rev. 3) and the SRP and are acceptable. The staff finds it to be acceptable. This item is resolved.

The ABWR is designed so that operation will not require alternate high-radiation area controls (pursuant to 10 CFR 20.203(c)(5)), as used in current operating BWRs. All high radiation areas (with greater than 1.0 mSv/hr (100 mrem/hr)) can be locked to control unauthorized access. No credit is taken for the relief provided in Section 12.6 of the BWR standard technical specifications (i.e., area locked at 10.0 mSv/hr (1,000 mrem/hr)). This design position meets the requirements of 10 CFR Part 20 and is acceptable.

Radiation Protection

12.3.2 Shielding

Radiation shielding will protect personnel against radiation exposure inside and outside the plant during normal operation, including AOOs, and during reactor accidents. All radioactive sources will be shielded on the basis of the access and exposure level requirements of the designed radiation zoning. Concrete used for radiation shielding meets the design guidance provided in RG 1.69, "Concrete Radiation Shields for Nuclear Power Plants" (Rev. 0). GE performed shielding calculations with the QAD-F, GGG, and DOT.4 computer codes. These are commonly accepted shielding calculational codes referenced in the SRP and are acceptable.

GE has not provided the thickness of specific radiation shields, contrary to the guidance of RG 1.70 (Rev. 3) and the acceptance criteria of the SRP. GE's position is that, because the system layouts and the physical dimensions of the as-procured radioactive system components are not known, the shielding requirements for these systems cannot be provided at this stage of the ABWR design. Therefore, the staff cannot conduct confirmatory calculations of shielding effectiveness. As an alternative method, GE has provided DAC to verify the adequacy of the ABWR shielding design. The staff's review of these DAC is discussed in Section 12.3.5.1 below. This alternative is acceptable, and the staff considers this item resolved.

The adequacy of the shielding in the upper drywell was identified as an Open Item 112 in the DSER (SECY-91-355). The biological shield surrounding the reactor vessel did not cover a significant portion of the top of the reactor vessel. A fuel handling mishap resulting in dropping an SFA across the reactor flange would result in extremely high dose rates in the upper drywell with this design. In addition to the radiological hazard presented by this AOO, it appears that raising an SFA in proximity of the vessel wall could result in significant radiation dose rates in the upper drywell. Amendment 21 revised the SSAR to reflect a design change to the shielding in the upper drywell, raising the biological shield to within 4 inches of the upper drywell ceiling. This design change would provide sufficient shielding during the normal withdrawal of SFAs from the reactor. However, a dropped SFA resting across the reactor flange would still produce significant radiation streaming into the upper drywell. Personnel in the upper drywell during this AOO could receive lethal radiation doses before they could escape. This was DFSER Confirmatory Item 12.3.2-1. In response to the staff's concerns, GE revised the design change to add a shielding ledge to the opening in the upper drywell ceiling. This ledge significantly reduces the radiation streaming into the upper drywell from a SFA resting on the reactor vessel flange. According to the

staff's analysis, this shield design would reduce the dose rates in the upper drywell to less than 5 Sv/hr (500 rem) during a worse-case fuel-drop AOO. Therefore, the staff concludes that there is reasonable assurance that individuals could escape the upper drywell without receiving life-threatening radiation doses. Amendment 23, which revised the SSAR to reflect this final design, is acceptable to the staff. This item is resolved.

The shielding of the TIP system was identified as Open Item 106 in the DSER (SECY-91-355). As discussed in Section 12.1.2 of this report, the TIP drive and the TIP storage are located in separate shielded rooms. However, the conduit that guides the TIP from the reactor to its storage is virtually unshielded. This conduit shares the primary containment penetration with the lower drywell personnel access. Personnel at the lower drywell access hatch or in the access tunnel would be exposed to the unshielded activated TIP and the drive cable as they are retracted from the reactor core. DFSER Confirmatory Item 12.3.2-2 was identified on the basis of a March 26, 1992, draft SSAR amendment. Amendment 21 revised the description of the radiation design features associated with the TIP system. This amendment notes that the lower drywell access is located in a separate shielded room that can be locked to prevent access to these areas while the TIP is being withdrawn from the core. In addition, flashing alarms at the door to this room and at the lower drywell access hatch will warn personnel when power is applied to the TIP drives. Also, the TIP system will operate so that the TIP will be withdrawn in the high-speed mode, minimizing the transit time of the activated components through the unshielded portions of the system. These features ensure that the personnel radiation exposures resulting from the operation of the TIP system can be maintained ALARA and are acceptable. The staff finds it to be acceptable.

12.3.3 Ventilation

The ABWR ventilation systems are designed to protect personnel and equipment from extreme environmental conditions and to ensure that plant personnel are not inadvertently exposed to airborne contaminants exceeding the concentration limits given in 10 CFR Part 20. Design features intended to maintain personnel exposures ALARA include the following:

- Airflow between areas potentially having airborne contamination will always be from the area of lower potential contamination to the area of higher potential contamination.

- Negative or positive pressure will be used in areas to prevent exfiltration or infiltration of possible airborne contamination, respectively.
- The control room ventilation has dual fresh air intakes designed so that at least one intake will be free of contamination following a loss-of-coolant accident.

These design features are in accordance with the guidelines of RG 8.8 (Rev. 3) and are acceptable. However, as noted in Section 12.2 of this report, the expected leakage of radioactive fluids from plant systems cannot be determined at this stage of the ABWR design. Without this source term, GE is not able to provide the concentrations of airborne contamination in cubicles, rooms, and corridors as specified in the SRP. Therefore, the staff cannot verify that the plant ventilation system design meets the criteria in the SRP. This was identified as Open Item 113 in the DSER (SECY-91-355). As an alternative, GE provided DAC that require the COL applicant to calculate the expected concentrations of airborne radionuclides as specified in the SRP, to verify that adequate ventilation is provided. The staff identified DFSEF Confirmatory Item 12.3.3-1 on the basis of a May 1, 1992, draft SSAR amendment. Section 12.3.5.2 of this report contains the staff's evaluation of these DAC. Amendment 21 added Appendix 12A to the SSAR which describes the calculational methods and assumptions that will be used to satisfy the DAC in Table 3.2.b of the CDM. These calculational methods and assumptions are consistent with provisions of the SRP and are acceptable. The staff finds it to be acceptable. This item is resolved.

12.3.4 Area Radiation and Airborne Radioactive Monitoring Instrumentation

Open Item 114 in the DSER (SECY-91-355), questioned the description of the ABWR area radiation monitoring system. GE revised the SSAR with the following information. The area radiation monitoring system consists of 25 gamma sensitive detectors and their associated digital monitors. The detectors are in key locations of the plant and will have operating ranges (sensitivity) commensurate with the expected radiation levels in the areas. Monitored radiation levels will be recorded and indication will be provided in the control room. These area monitors will be powered from the non-1E vital 120-Vac bus. The monitors will have local audible alarms with adjustable settings (both up-scale and down-scale) to warn personnel of abnormal conditions such as higher-than-normal radiation levels or detector failure. High-range radiation accident monitors that meet the criteria of RG 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident" (Rev. 3),

are provided in the RHR equipment area. In addition, to assess the magnitude of the release of radioactive material from the core during an accident, four high-range gamma sensitive ion chambers in the containment will be able to measure up to 0.72 c/Kg per second (10^7 R/hr). The staff concludes that the area radiation monitoring system meets the applicable criteria in RG 8.8 (Rev. 3), RG 1.97 (Rev. 3), and the provisions in Item II.F.1.3 of NUREG-0737 that are required by 10 CFR 50.34(f)(2)(xvii)(D) and is acceptable. The COL applicant will address the operational considerations, such as monitor alarm set points, listed in RG 1.70 (Rev. 3) Section 12.3.4. This was DFSEF COL Action Item 12.3.4-1. GE has included this information in a markup of SSAR Section 12.3.4 dated April 16, 1993. GE has also included this action item in the SSAR and the staff finds it to be acceptable.

The staff noted in the DSER (SECY-91-355) that criticality accident monitors were not provided in the ABWR design to meet the requirements of 10 CFR 70.24 as provided in the SRP. In response to the staff's request, GE amended the SSAR to state that these monitors are unnecessary because the ABWR is designed to ensure subcritical conditions during fuel handling and storage. Several licensees of operating BWRs with similar design features and fuel handling procedures have received a license condition exempting them from this 10 CFR 70.24 requirement. The requirements of 10 CFR Part 70 are outside the scope of this review. The COL applicant will provide information showing that their plant meets the requirements of 10 CFR 70.24 or request an exemption. This was DFSEF COL Action Item 12.3.4-2. GE has included this action item in a markup of SSAR Section 12.3.4 dated April 16, 1993. GE has also included this action item in the SSAR and the staff finds it to be acceptable.

Monitoring of airborne radioactive materials in nuclear power plants typically is provided by fixed continuous air monitors. These monitors sample the ventilation air exhausted from plant areas having the highest potential for radioactivity release. Movable continuous air monitors are positioned in plant areas that have a potential for airborne radioactivity release during certain operating modes (i.e., an area where a radioactive system is opened during maintenance) to supplement the fixed monitors. GE has not described the airborne monitoring for the ABWR design. As discussed in Section 12.3.3 of this report, the expected concentrations of airborne radionuclides cannot be determined at the current level of the ABWR design detail. As an alternative, GE has provided DAC that would require the COL applicant to verify that airborne monitors provided in the final ABWR design meet the criteria of the SRP. The staff's review of these DAC is in

Radiation Protection

Section 12.3.5 below. The COL applicant will address the operational considerations, such as the procedures for operation and calibration of the monitors as well as the placement of the movable monitors, in the COL application. This was DFSEER COL Action Item 12.3.4-3. GE has included this information in a markup of SSAR Section 12.3.4 dated April 16, 1993. GE has also included this action item in the SSAR and the staff finds it to be acceptable.

12.3.5 Certified Design Material

The staff initially identified three areas where the level of design detail in the SSAR did not allow the staff to conclude that the ABWR design meets the acceptance criteria in Chapter 12 of the SRP. These areas are the adequacy of the plant radiation shielding, the adequacy of the plant ventilation system, and the adequacy of the plant airborne radionuclide monitoring system. As an alternative, GE provided DAC requiring the COL applicant to perform shielding analysis and airborne radionuclide concentration calculations that will be verified by the ITAAC during plant construction to verify that the final ABWR design is acceptable. Details of the staff's review of these DAC follow.

12.3.5.1 Plant Shielding DAC

Chapter 12 of the SSAR contains layout drawings of the plant that indicate the designed maximum radiation level (or zone) for each room, equipment cubicle, and operating space during normal power operations, shutdown operations and accident conditions. As discussed in Section 12.2 above, the piping layout and component selection have not been set for the ABWR systems; therefore, parameters such as source strength and geometry needed to verify the adequacy of the radiation shields around these systems are not available. In addition, nitrogen-16 gammas from the plant can significantly contribute to offsite dose rates. The adequacy of the plant shielding needed to comply with the radiation dose limits for individual members of the public in 10 CFR Part 20 cannot be verified since the turbine design and site-specific parameters such as distance to the site boundary are unknown.

GE has submitted DAC for plant shielding in Table 3.2.a of the CDM. These DAC require the COL applicant to verify the adequacy of (1) the shielding around rooms and spaces during normal operations and shutdown conditions, (2) the shielding and temporary shield space provided between plant systems during maintenance activities, (3) the shielding provided around vital plant areas during accident conditions (Three Mile Island (TMI) Action Plan Item II.B.2 (10 CFR 50.34(f)(2)(vii)), and (4) the plant

shielding needed to limit public dose. The staff's review indicates that the analysis assumptions, methods, and acceptance criteria in these DAC are consistent with the criteria in the SRP. Therefore, the staff concludes that compliance with these DAC, as supplemented by the information in SSAR Sections 12.3.2, is acceptable to adequately address the relevant requirements in 10 CFR 50.34(b)(3) and 10 CFR Part 20 concerning the limitation of radiation exposures to personnel, including the requirement to maintain doses ALARA as supplemented by the guidance in RG 8.8 (Rev. 3); 10 CFR 50.34(f) and GDC 19 with respect to operator access to plant areas during and following a reactor accident as supplemented by the guidance in Item II.B.2 of NUREG-0737; and GDC 61 regarding adequate shielding of fuel storage and handling, radioactive waste, and other systems which may contain radioactivity. GE provided a revised set of design descriptions and ITAAC (including DAC). The adequacy and acceptability of the ABWR Tier 1 material and ITAAC (including DAC) are evaluated in Section 14.3 of this report.

12.3.5.2 Ventilation and Airborne Monitoring DAC

The level of detail in the current ABWR design is not sufficient to provide the expected airborne concentrations in rooms and operating areas within the plant as specified in RG 1.70 (Rev. 3). Therefore, GE has not provided a description of the airborne monitoring system consistent with the criteria in the SRP.

GE has submitted DAC for ventilation and airborne monitoring in Table 3.2.b of the CDM. These DAC requires the COL applicant to calculate the expected concentrations of airborne radioactivity in each equipment cubicle, corridor, and operating area that require personnel access. These DAC also require an analysis by the COL applicant to identify those areas of the plant that require continuous monitoring of airborne radioactive materials. The staff's review indicates that the assumptions and acceptance criteria in these DAC are consistent with the criteria in the SRP. Therefore, the staff concludes that compliance with these DAC, as supplemented by the information in SSAR Sections 12.3.3 and 12.3.4 and Appendix 12A, will meet the relevant requirements in 10 CFR 50.34(b)(3) and 10 CFR Part 20 concerning the limitation of radiation exposures to personnel from airborne radioactive material, including the requirement to maintain doses ALARA as supplemented by the guidance in RG 8.8 (Rev. 3); GDC 61 regarding adequate shielding of fuel storage and handling, radioactive waste, and other systems which may contain radioactivity; and the requirements in 10 CFR Part 20, 10 CFR 50.34(f) and GDC 64 related to in-plant monitoring of airborne radioactive materials during routine operating conditions.

The adequacy and acceptability of the ABWR Tier 1 material and ITAAC (including DAC) are evaluated in Section 14.3 of this report.

12.3.5.3 Radiation Design Features

In the DSER (SECY-91-355), the staff identified a number of radiation design features to be addressed in the ABWR Tier 1 design description and ITAAC. This was identified as DFSER Open Item 12.3.5.3-1. GE provided a revised set of design descriptions and ITAAC (including DAC). The adequacy and acceptability of the ABWR Tier 1 material and ITAAC (including DAC) are evaluated in Section 14.3 of this report. On the basis of this evaluation, this item is resolved.

12.3.6 10 CFR 50.34(f): TMI-Related Items

SSAR Section 12.3 addresses two items from the TMI Action Plan (NUREG-0660), II.F.1.3 (10 CFR 50.34(f)(2)(xvii)(D), and II.B.2 (10 CFR 50.34(f)(2)(vii)). Item II.F.1.3 requires that high-range radiation accident monitors be provided in the containment. Item II.F.1.3 specifies that high-range dose monitors be capable of detecting dose rates up to 10^5 rads per hour. NUREG-0737 modified this position to specify that gamma sensitive monitors be capable of reading up to 10^7 R per hour. As discussed in Section 12.3.4 of this report, the ABWR design has four monitors (two in the drywell and two in the suppression chamber) that meet the provisions of Item II.F.1.3. Item II.B.2 specifies that radiation shielding be provided so that operators can access vital equipment in the plant during an accident without receiving excessive radiation dose. As discussed in Section 12.3.5.1 of this report, the DAC in Table 3.2.a of the CDM require the COL applicant to demonstrate compliance with Item II.B.2, as part of the analysis to verify the adequacy of the plant's radiation shielding.

The COL applicant will be responsible for demonstrating compliance with II.B.2 because the final hardware and system design specifications need as inputs to shielding calculations are not available now.

12.4 Dose Assessment

The staff has audited the dose assessment in Section 12.4 of the SSAR for completeness against the guidelines in RG 1.70 (Rev. 3) and against the criteria set forth in SRP Section 12.3.II.5. This review consisted of ensuring that GE had either committed to following the criteria of RGs and staff positions in Section 12.3 of the SRP or provided acceptable alternatives. In addition, the staff selectively compared the dose assessment made by GE for

specific functions against the experience of operating BWRs. Details of the review follow.

GE provided an assessment of the radiation dose that would be received by operating a plant of the ABWR design. Estimated person-rem doses for major work within areas of the plant during maintenance and refueling periods and for power operations are given in SSAR Table 12.4-1 and result in an estimated total annual dose of 0.989 person-sievert (98.9 person-rem).

In Open Item 115 in the DSER (SECY-91-355), the staff identified several deficiencies in GE's dose assessment, including mathematical errors, inconsistencies between text and tables, a lack of bases for assumptions, and a level of detail that was not consistent with the guidance in RG 8.19, "Occupational Radiation Dose Assessment in Light-Water Reactor Power Plants--Design Stage Man-Rem Estimates" (Rev. 1). GE amended the SSAR to correct the math errors and inconsistencies noted and to give the bases for the assumptions and values used in the assessment.

In GE's dose assessment, the stay times and frequencies for each task are based on a detailed task analysis of maintenance activities in a BWR with a similar containment design (MARK III containment). Average dose rates are based on past experiences for similar tasks. These values were then adjusted to account for ABWR design features to obtain the expected doses. The cumulative annual dose of 0.989 person-sievert (98.9 person-rem) for personnel operating an ABWR plant is consistent with the Electric Power Research Institute design guideline of 1.0 person-sievert (100 person-rem) per year and compares favorably with the average current BWR experience (which is more than twice the ABWR projected dose). Although not in the format specified in RG 8.19, this detailed dose assessment meets the intent of RG 8.19 and is acceptable. The staff considers this item resolved.

12.5 Organization

The organization required to implement an effective health physics program and ensure that radiation exposures are within the limits of 10 CFR Part 20 and are ALARA is outside the scope of this review. The COL applicant seeking an operating license by referencing the ABWR certified design will address this concern to the level of detail discussed in RG 1.70 (Rev. 3). This was DFSER COL Action Item 12.5-1. GE has included this information in a markup of SSAR Section 12.5.3.1 dated April 16, 1993. GE has also included this action item in the SSAR and the staff finds it to be acceptable.

Radiation Protection

12.5.1 10 CFR 50.34(f): TMI-Related Items

The regulation in 10 CFR 50.34(f)(2)(xvii) requires in-plant radiation and airborne radioactivity monitoring in accordance with Item III.D.3.3 of the TMI Action Plan. Item III.D.3.3 requires that operating reactors be capable of accurately measuring radio-iodine concentrations in plant areas under accident conditions. The NUREG-0737 clarification of Item III.D.3.3 specifies that this capability use portable instruments and includes requirements for training and procedures for the use of these instruments.

These programmatic requirements are outside the scope of this review and have been identified as items to be addressed by the COL applicant. This was identified as DFSBR COL Action Item 12.5.1-1. Appendix A to SSAR Chapter 1, Section 1A.3.3, as revised by Amendment 23, identifies post-accident radio-iodine monitoring as an issue to be addressed by the COL applicant. GB also has included this information in a markup of SSAR Section 12.5.3.2 dated April 16, 1993. GB has included this action item in the SSAR and the staff finds it to be acceptable.

13 CONDUCT OF OPERATIONS

13.1 Organizational Structure of Applicants

The staff completed its review of Standard Safety Analysis Report (SSAR) Section 13.1, "Organizational Structure of Applicants," and finds this section to be adequate. The staff agrees that the information related to the combined license (COL) applicant's organizational structure is outside the scope of the advanced boiling water reactor (ABWR) standard plant design. This information will be the responsibility of the COL applicant referencing the ABWR design at the COL phase described in 10 CFR 52.79(b).

13.2 Training

The staff's review of SSAR Section 13.2 was based on the current regulatory requirements in 10 CFR 52.47, 10 CFR 50.34(g), and 10 CFR 50.34(f) and the guidance in SRP Section 13, NUREG-0700, and NUREG-0933. The staff developed additional review criteria to provide a basis for the review of aspects of the ABWR human factors engineering (HFE) program that were not fully addressed by the previously mentioned documents. These criteria are contained in the staff's "HFE Program Review Model (PRM) and Acceptance Criteria for ABWR," which was forwarded to the Commission in SECY-92-299 dated August 27, 1992, and is attached as Appendix J of this report. The HFE PRM considered aspects of training as it pertained to the verification and validation (V&V) of the main control room and remote shutdown system designs.

GE has not included training development in the scope of the ABWR design certification application and has stated that development of training will be the responsibility of the COL applicant. However, information on the incorporation of operating experience (Three Mile Island (TMI) I.C.5) and preoperational and low power testing (TMI I.G.1) into training programs is provided by GE to the COL applicant in SSAR Section 13.2.

Although training is not considered to be part of the information required for certification of the ABWR design, the staff identified the following issues related to training during its review of SSAR Chapter 18, "Human Factors Engineering:" training materials, the use of a simulator for training, and the incorporation of operational experience into training programs. Each of these items is discussed below.

DSER Issue 18.22: Training Materials

In the DSER the staff stated that it "expects GE to develop and submit for certification a detailed program description for developing the training material as part of the design certification for the ABWR." Following issuance of the

DSER, the staff determined that training materials were already a part of the established licensing review process under 10 CFR Part 50 for the COL applicant and, therefore, did not need to be addressed in the ABWR design certification review under 10 CFR Part 52. The staff agrees that the submittal of training materials is beyond the scope of the ABWR design certification and finds GE's approach to be acceptable; therefore, this item is resolved.

TMI Action Item I.A.4.2

10 CFR 50.34(f)(2)(i) corresponds to TMI Action Item I.A.4.2, "Long Term Simulator Training Upgrade," with regard to simulator capabilities. GE states that "simulator facilities for use in performing operator training are outside the scope of the standard plant design certification." This is acceptable because training will be addressed by the COL applicant. This was draft final safety evaluation report (DFSER) COL Action Item 18.7.2.2-1. GE has included the COL action item, that the operator training program meets 10 CFR Part 50, as Item 18.8.8 in SSAR Section 18.8, and the staff finds this approach to be acceptable.

TMI Action Item I.C.5

The staff reviewed TMI Action Item I.C.5, "Feedback of Operating Experience," on the incorporation of operational experience into the training and procedure development programs. The staff determined that development of detailed procedures and training materials is beyond the scope of the ABWR design certification and is the responsibility of the COL applicant. GE has included the training and procedure development process as a COL license information item in SSAR Sections 13.2 and 13.5, and the staff finds this approach to be acceptable.

TMI Action Item I.G.1

The staff reviewed TMI Action Item I.G.1, "Training Requirements for Preoperational and Low-Power Testing." The staff determined that I.G.1 is beyond the scope of ABWR certification and is the responsibility of the COL applicant. GE has included the training requirements for preoperational and low power testing activities as a COL license information item in SSAR Section 13.2.3.2, and the staff finds this approach to be acceptable.

The staff also notes that V&V of training materials will be examined further during the V&V of the ABWR main control room and remote shutdown system as described in Appendix 18E of the SSAR. Therefore, the issue of training is resolved for the ABWR design certification.

Conduct of Operations

13.3 Emergency Planning

In Section 13.3 of the SSAR, GE states that emergency planning is not within the scope of the ABWR design and that the COL applicant will provide emergency plans in accordance with 10 CFR 50.33(g) and 52.72(d). The staff agrees that the requirement to provide the emergency plans is the responsibility of the COL applicant referencing the ABWR design and will depend significantly on both plant and site-specific characteristics. This was identified as DFSER COL Action Item 13.3-1. GE included this action item in Amendment 31 of the SSAR. The staff finds it to be acceptable.

Nevertheless, GE also acknowledges that there are design features, facilities, functions, and equipment necessary for emergency planning that must be considered in the design basis of a standard plant. These are addressed in SSAR Table 13.3-1, "ABWR Design Considerations for Emergency Planning Requirements," which specifies a technical support center (TSC) that complies with all of the TSC design requirements and is located in the service building adjacent to the control room. (The TSC is shown in SSAR Figure 1.2-19, "Control and Service Building, Arrangement Plan at Elevation 7900 mm.") GE further states that the TSC will contain the necessary facilities and equipment called for in Section 2 of NUREG-0696 ("Functional Criteria for Emergency Response Facilities," U.S. NRC, February 1981).

The staff performed its review in accordance with the requirements of 10 CFR 50.47(b), Appendix E to 10 CFR Part 50, and 10 CFR 50.34(f)(2)(xxv), which requires an onsite TSC, an onsite operational support center (OSC), and a nearsite emergency operations facility (EOF). The staff's review also considered the guidance provided in NUREG-0654 and NUREG-0696. It is the staff's position that the facilities and equipment for the ABWR standard plant TSC should be compatible with the control room and meet the applicable criteria of NUREG-0696. In this regard, the staff noted in the DFSER that whereas GE specified a TSC to support 20 people, NUREG-0696 specifies the following:

The TSC working space shall be sized for a minimum of 25 persons, including 20 persons designated by the licensee and five NRC personnel. This minimum size shall be increased if the maximum staffing level specified by the licensee's emergency plan exceeds 20 persons.

Therefore, the staff stated in the DFSER that the TSC for the ABWR standard plant should be sized for 25 persons and be compatible with the control room in order to meet the criteria of NUREG-0696. This was DFSER Open

Item 13.3-1. GE revised Table 13.3-1 in Amendment 25 to the SSAR to indicate that the TSC will be of sufficient size to support 25 people. The staff finds it to be acceptable. This item is resolved.

The ABWR standard plant design also includes considerations for decontamination of onsite individuals in the service building adjacent to the main change rooms as shown in Figure 1.2-20 of the SSAR. The staff finds these design considerations for an onsite decontamination facility acceptable for meeting the requirements of 10 CFR Part 50, Appendix E, Section IV.E.3.

GE considers other facilities that support emergency planning to be outside the scope of the ABWR standard plant design scope. These include an offsite EOF for the management of overall licensee emergency response, including coordination with Federal, State and local officials. The staff agrees that the EOF is not within the scope of the ABWR standard plant design, but must be addressed by the COL applicant referencing the ABWR standard plant design. This was identified as DFSER COL Action Item 13.3-2. GE included this action item in Amendment 31 of the SSAR. The staff finds it to be acceptable.

GE originally considered an onsite OSC (assembly area) separate from the control room and TSC where licensee operations support personnel report in an emergency to be outside the scope of the ABWR standard plant design. However, the staff noted in the DFSER that an OSC should be provided as part of the ABWR standard plant design. This was DFSER Open Item 13.3-2. GE revised Table 13.3-1 in Amendment 25 to the SSAR to state that the ABWR standard plant will comply with all of the OSC design requirements, and that the lunch room adjacent to the TSC in the service building will be identified as the OSC. The staff finds it to be acceptable. This item is resolved.

Certified Design Material

The requirements for the TSC and OSC were not covered in the Tier 1 design descriptions or inspections, tests, analyses, and acceptance criteria (ITAAC). This was DFSER Open Item 13.3-3. GE has provided a revised set of design descriptions and ITAAC. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Section 14.3 of this report. On the basis of this evaluation, this item is resolved.

10 CFR 50.34(f):TMI-Related Items

SSAR Section 13.3 addresses TMI Action Plan (NUREG-0660) Item III.A.1.2(10 CFR 50.34(f)(2)(xxv)).

This item requires the COL applicant to upgrade its emergency support facilities by establishing a TSC, OSC, and a near site EOF for command and control, support, and coordination of onsite and offsite functions during reactor accident situations. As discussed in Section 13.3 of this report, the ABWR design provides for a TSC and an OSC. The near-site EOF is considered by the staff not to be within the scope of ABWR standard plant design and will be addressed by the COL applicant referencing the ABWR standard plant design (DFSER COL Action Item 13.3-2). GE has included this action item in the SSAR. The staff finds it to be acceptable.

13.4 Review and Audit

The staff determined that the review and audit information is outside the scope of the ABWR standard plant design. The COL applicant will provide the necessary information on reviews and audits for plant operation. It will be reviewed in detail during the COL stage. GE has included this action item in the SSAR and the staff finds it to be acceptable.

13.5 Plant Procedures

The staff's review was based on the current regulatory requirements established in 10 CFR 52.47, 10 CFR 50.34(g), and 10 CFR 50.34(f) and the guidance contained in SRP Section 13, NUREG-0700 and NUREG-0933. The staff developed additional review criteria to provide a basis for the review of aspects of the ABWR HFE program that were not fully addressed by the previously mentioned documents. These criteria are contained in the staff's HFE PRM, which is attached as Appendix J of this report. The HFE PRM considered aspects of plant procedures as they pertained to the V&V of the main control room and remote shutdown system designs.

GE has not included procedure development in the scope of its ABWR design certification application and has identified procedure development as a COL responsibility in SSAR Section 13.5. GE's description of this COL license information item is consistent with the staff's HFE PRM - Element 7, procedures, developed to support the review of the ABWR HFE effort. The HFE PRM is described in Appendix J of this report.

Although plant procedures are not considered to be part of the information required for certification of the ABWR design, the staff identified several issues related to procedure development during the review of SSAR Chapter 18, "Human Factors Engineering." Items regarding procedure development as well as differences between Element 7 of the HFE PRM and SSAR Section 13.5 are discussed below.

(1) General Criterion 5 of Element 7 of the HFE PRM

General Criterion 5 of Element 7 of the HFE PRM states: "All procedures shall be verified and validated. A review shall be conducted to assure procedures are correct and can be performed. Final validation of operating procedures shall be performed in a simulation of the integrated system as part of V&V activities described in Element 8." Although GE has not included a requirement for final validation of operating procedures, in its design scope, as a COL action item, GE has stated that procedures will be available for the human-system interface (HSI) V&V activities (as specified in SSAR Section 13.5.3.3, - Item e). These activities are described further in SSAR Appendix 18E and SSAR Table 18E-4. The details of GE's V&V process are provided in SSAR Table 18E-4 and require that final procedures be included by the COL applicant in the V&V activities for both the main control room and the remote shutdown system. On the basis of SSAR Section 13.5.3.3 and the inclusion of procedures in SSAR Table 18E-4, the staff finds specification of this criterion in the SSAR as a COL action item to be acceptable.

(2) General Criterion 6 of Element 7 of the HFE PRM

General Criterion 6 of Element 7 of the HFE PRM states: "An analysis shall be conducted to determine the impact of providing computer-based procedures and to specify where such an approach would improve procedure utilization and reduce operating crew errors related to procedure use." This activity is not part of GE's design scope. GE states that an analysis of computer-based procedures will be conducted as part of the task analysis to be conducted by the COL applicant and evaluated further as part of the HSI. The description of the HSI requirements is contained in Table 3.1 of the ABWR certified design material. Because the computerization of procedures is an aspect of the HSI design implementation, the incorporation of the analyses as part of HSI Element 4 is acceptable. Therefore, the staff finds the specification of COL completion of this criterion to be acceptable.

Although plant procedures are not considered to be part of the information required for design certification of the ABWR, the staff identified the following issues related to procedures during its review of GE's HFE program: procedure development (DSER — SECY-91-320 — Issue 18.21), guidelines for updating procedures (HF-4.4), short-term accident procedures review (TMI Action

Conduct of Operations

Item I.C.1), incorporation of operating experience (TMI Action Item I.C.5) and a long-term plan for upgrading procedures (TMI Action Item I.C.9). Each of these items is addressed below.

DSER (SECY-91-320) Issue 18.21: Procedure Development

In the DSER, the staff stated that system-level operating procedures would be developed concurrent with the development of the ABWR systems design. These procedures and the associated task analyses on which the HSI requirements are based are not included in the design certification; thus, they could not be evaluated.

The staff determined that development of detailed procedures and associated training materials is beyond the scope of the ABWR design certification and is the responsibility of the COL applicant. This was DFSEER Confirmatory Item 18.9.2.2.7-1. GE has included the COL license information in the SSAR Section 13.5 which adequately addresses procedure development for the certification of the ABWR design. The staff finds GE's approach to be acceptable. Therefore, Confirmatory Item 18.9.2.2.7-1 is resolved.

USI/GSI Item HF-4.4 and TMI Action Items I.C.1, I.C.5 and I.C.9

The staff reviewed Item HF-4.4, "Guidelines for Upgrading Other Procedures," TMI Action Item I.C.1, "Short-Term Accident and Procedures Review," TMI Action Item I.C.5, "Feedback of Operating Experience," and TMI Action Item I.C.9, "Long Term Plan for Upgrading Procedures." It was determined that development of detailed procedures is beyond the scope of the ABWR certification and is the responsibility of the COL applicant. GE has included the procedure development process as a COL license information item in SSAR Section 13.5.3. SSAR Section 13.5.3.1 states that the methods and criteria for the development, V&V, implementation, maintenance and revision of procedures will include consideration of TMI Action Items I.C.1, I.C.5, and I.C.9. The staff finds GE's approach to be acceptable. The staff interprets this process to include the analysis of HF-4.4.

The staff also notes that V&V of plant procedures will be examined further during the V&V of the ABWR control room and remote shutdown system as described in GE's certified design material, Table 3.1, "Human Factors Engineering," and in Appendix 18E of the SSAR. Therefore, GE's proposed treatment of plant procedures is acceptable for the purposes of ABWR design certification.

13.6 Physical Security

13.6.1 Preliminary Planning

SSAR Section 13.6 states that preliminary planning is not required because the security plan will be the responsibility of COL applicants who reference the ABWR standard plant design. The staff finds this approach to be acceptable. Since the COL application will include a physical security plan, safeguards contingency plan, and a guard qualification and training plan, a preliminary planning submittal is not necessary for the design certification.

13.6.2 Security Plan

The SSAR states that the development of the security plan is beyond the scope of the ABWR standard plant design. The staff finds this approach to be acceptable. In addition to the action items listed in Section 13.6.3 of the SSAR, the COL applicant will provide site-specific security, contingency, and guard training plans in accordance with 10 CFR 50.34 and 10 CFR Part 73. This was identified as DFSEER COL Action Item 13.6.2-1. By SSAR Amendment 25, GE identified in Section 13.6.3.8 that the COL applicant will provide site-specific security, contingency, and guard training plans. The staff finds it to be acceptable.

The staff requires that at least 60 days before loading fuel, the COL applicant will confirm that the security systems and programs described in its physical security plan, safeguards contingency plan, and guard qualification and training plan have achieved operational status and are available for NRC inspection. Operational status means that the security systems and programs are functioning in entirety as they would when the reactor is operating and will remain so. The COL applicant's determination that operational status has been achieved must be based on tests conducted under realistic operating conditions of sufficient duration to demonstrate (1) that the equipment is properly operating and capable of long-term, reliable operation; (2) that procedures have been developed, approved, and implemented; and (3) that personnel responsible for security operations and maintenance have been appropriately trained and have demonstrated their capability of performing their assigned duties and responsibilities. This was identified as DFSEER COL Action Item 13.6.2-2. GE has included this action item described above in the SSAR and the staff finds it to be acceptable.

13.6.3 Control of Access to Areas Containing Vital Equipment

Section 13.6.3 of the SSAR identifies a number of interfaces between the ABWR standard plant design and the remainder of the plant that must be addressed by COL applicants who reference the ABWR standard plant design.

The staff reviewed the interfaces and determined that these were not interfaces as described in 10 CFR 52.47 but were actions to be accomplished as part of the COL application. The staff found them to be acceptable as COL action items subject to the addition of the requirement that the applicant provide plant-specific security, contingency, and guard training plans in accordance with 10 CFR 50.34 and 10 CFR Part 73. This was identified as DFSER COL Action Item 13.6.3-1. GE has included this action item as described above in the SSAR and the staff finds it to be acceptable.

13.6.3.1 Introduction

Section 13.6.3.1 of the SSAR states that SSAR Section 13.6.3 deals with the control of access to areas containing vital equipment.

13.6.3.2 Design Bases

Section 13.6.3.2 of the SSAR states that security functions described in Section 13.6.3 are incorporated into the overall ABWR design so that the plant is in compliance with the requirements of 10 CFR Part 73.

The Electric Power Research Institute (EPRI) Advanced Light Water Reactor (ALWR) Utility Requirements Document (Volume II, Revision 1, Chapter 11, Section 8.4.1) specifically requires the protected area lighting to be powered from an uninterruptible power source. In its response to a request for additional information (RAI) Q910.18, GE identified a site security load on the non-Class 1E vital (uninterruptible) load list (SSAR Table 20B-1), but the staff considered the description of this interconnection to be insufficiently defined. In response to staff comments, GE added SSAR Section 19B.3.12, which clarified the connection between the security system uninterruptible power requirements (to be later so determined by the plant-specific security system designer as to meet required security system performance) and the non-Class 1E vital power supply capacity. SSAR Section 19B.3.12 requires that the site security system be powered from a non-Class 1E vital (uninterruptible) ac power source. The protected area boundary lighting subsystem is the only exception. At the discretion of the COL applicant, the protected area boundary lighting subsystem may be powered from a non-Class 1E vital

(uninterruptible) ac power source or from an interruptible power source provided adequate compensatory measures are established by the site's physical security implementation plan. The staff considers this resolution acceptable and this portion of DFSER Open Item 1.1-1 is resolved.

In NRC Information Notice 83-83, the staff suggested that new plant designs that make extensive use of solid-state devices in instrument and control circuits may experience reactor system malfunctions and spurious actuation as a result of portable communication devices in their vicinity. In RAI Questions (Q)910.10 and Q910.17, the staff asked that radio-frequency interference design criteria be established to ensure that security personnel within the reactor and control building could maintain radio communication without adversely affecting plant operation. GE's response to Q910.17 referenced discussions of system tolerance to electromagnetic interference. The staff finds that the plant security systems criteria in SSAR Section 9.5.13.11 and the amendment to SSAR Appendix 7A, in response to a concern raised in DSER Section 7.1.3.3, adequately resolve staff's concern.

13.6.3.3 Vital Areas

Section 13.6.3.3 of the SSAR itemizes by location the plant equipment to be considered vital equipment pursuant to 10 CFR 73.2 and the vital areas containing that equipment. SSAR Figures 13.6-1 through 13.6-14 outline the vital areas.

In RAIs Q910.9, Q910.11, and Q910.20, the staff questioned the completeness of the list of vital equipment in SSAR Section 13.6.3.3. GE clarified this list on February 22, 1991, after discussions with staff. The staff is satisfied that the list of vital equipment includes all active and passive plant equipment essential to safe shutdown of the reactor, including necessary support systems, the reactor vessel and the remainder of the reactor coolant system pressure boundary within the primary containment, the suppression pool, spent fuel in the fuel pool, and any associated piping, equipment, and controls whose failure could result in an offsite release in excess of 10 CFR Part 100 limits. The staff finds this to be compatible with NRC Review Guideline 17 (January 23, 1978, memorandum from R. Clark to safeguards licensing staff). Prior to issuance of a COL, the staff's review of the designation of equipment as vital in plant-specific applications will focus on plant support equipment outside the scope of the certified ABWR design. In addition, 10 CFR 73.55(e) requires that the central alarm station be considered a vital area and secondary power supply system for alarm annunciator equipment and that non-portable communications equipment be located in

Conduct of Operations

vital areas. The secondary alarm station also is typically on site and treated as a vital area. Vital area classification of the central and secondary alarm stations was identified as DFSER COL Action Item 13.6.3.3-1. In SSAR Amendment 25, GE stated in Section 13.6.3.8 that the COL applicant will provide site-specific security, contingency, and guard training plans. Specifically, as stated in Section 13.6.3.10, the COL applicant will provide the classification of the control and secondary alarm stations. Vital area classification will be addressed at the time of the plant-specific security plan review. The staff finds it to be acceptable.

The staff expects that at least 60 days before loading fuel, a licensee will have confirmed that no portion of as-built vital systems is located outside of designated vital areas or can be prevented from performing their safety functions from outside the vital areas (e.g., by reach-rod valve manipulation). This verification should include piping, valves, and motor control centers that are required for maintaining boundary integrity, for performing the safe reactor shutdown cooling function, and for isolating safety-related equipment from non-safety-related equipment. This was identified as DFSER COL Action Item 13.6.3.3-2. In SSAR Amendment 25, GE specified in Section 13.6.3.11 that the COL applicant will confirm that locations of vital systems and system operations meet the above requirements. The staff finds it to be acceptable.

The plant-specific licensing review of the security and contingency response plan also will include an evaluation of whether the security response force's capability to interdict the violent external assault postulated in 10 CFR 73.1(a)(1)(i) properly accounts for the minimum penetration delay provided by the vital area barriers and doors. This was identified as DFSER COL Action Item 13.6.3.3-3. In SSAR Amendment 25, GE stated in Section 13.6.3.12 that the COL applicant will provide an evaluation of the interdiction capability of the security response force. The security response force's capability to interdict the violent external assault postulated in 10 CFR 73.1(a)(1)(i) will be addressed at the time of the plant-specific security plan review. The staff finds it to be acceptable.

13.6.3.4 Methods of Access Control

SSAR Section 13.6.3.4 describes, in general terms, the types of door controls that will be used to control access to vital areas. In response to RAIs Q910.12, Q910.21, and Q910.22, GE added statements to the SSAR that all doors and hatches connecting vital to non-vital areas are to be alarmed and emergency egress will not require keys or card readers. The staff finds this approach to be acceptable.

13.6.3.5 Access Control and Security Measures Through Exterior Doors to the Nuclear Island

SSAR Section 13.6.3.5 describes the specific security measures at portals into the reactor and control buildings from exterior areas and facilities of the remainder of the plant. For the DSER (SECY-91-235), the staff stated the types of door controls specified in SSAR Section 13.6.3.4 were generally acceptable, but insufficient detail was provided to determine compatibility with RG 5.12, Revision 0. This description of access control methods also did not address the positive control requirement of 10 CFR 73.55(d)(7)(i)(B) and the record-keeping requirement of 10 CFR 73.70(d), which requires logging individuals' times of entry to and exit from each vital area. This was identified as DSER (SECY-91-235) Open Item 36 and DFSER COL Action Item 13.6.3.5-1. In SSAR Amendment 25, GE stated in Section 13.6.3.13 that the COL applicant will demonstrate that door controls are compatible with RG 5.12, the positive control requirement of 10 CFR 73.55(d)(7)(i)(B) and the record-keeping requirement of 10 CFR 73.70(d). This issue will be considered during review of the plant-specific security plan. The staff finds it to be acceptable.

In RAI Q910.19, the staff asked why the parameters for environmental conditions in SSAR Appendix 3I should not apply to the design and qualification of security access control components. TMI Action Item II.B.2 (NUREG-0737) identifies areas for which environmental qualification of equipment necessary to ensure post-accident access may need to be considered. Although the "security center" is not safety related, it is included in NUREG-0737 because access to it may be necessary to give access to the rest of the plant. In NRC Information Notice 86-106, Supplement 2, the staff discussed an event at the Surry Power Station in which condensed steam saturated a security card reader and caused a short circuit in the card reader system for the entire plant. As a result, key cards would not open doors controlled by the security system. In the same event, a security communications system radio repeater was temporarily degraded because a thick layer of ice had formed on it from actuation of a carbon dioxide discharge nozzle. In response to RAI Q 910.19, GE stated that (1) this equipment is not safety related and is not required to operate under accident conditions, (2) the card reader design is required to preclude the possibility of failure of one card reader affecting the operation of any other card reader, (3) card reader doors are required to have a key-operated override, and (4) emergency exits are required to be designed so that personnel can exit without using keys or card readers. The staff considers this response to be consistent with currently accepted industry practice and is, therefore, acceptable.

Furthermore, SSAR Chapter 19, Appendix 19B.3.10, requires a COL applicant to evaluate the effects of the security system on required operator actions during all emergency modes of operation. The staff's position is that this analysis should include consideration of an emergency requiring evacuation of the control room in the control building to the remote shutdown panel in the reactor building. This was identified as DSER (SECY-91-235) Open Item 37 and DFSER COL Action Item 13.6.3.5-2. In SSAR Amendment 25, GE stated in Section 13.6.3.8 that the COL applicant will provide site-specific security, contingency, and guard training plans. Evaluation of compliance with the vital equipment prompt access requirements of 10 CFR 73.55(d)(7)(ii) will be resolved during review of the plant-specific security plan. The staff finds it to be acceptable.

13.6.3.6 Bullet-Resistant Walls and Doors, Security Grills, and Screens

SSAR Section 13.6.3.6 discusses bullet-resistant walls and doors and security grills and screens incorporated into the building design, with the stated intent of minimizing forcible access to the control room. In its responses to RAIs Q910.13, Q910.23, and Q910.24, GE did not resolve staff uncertainty as to the adequacy of barriers in all man-sized openings in physical barriers that separate other vital from non-vital areas. Also, the staff position on the effectiveness of the ventilation system barriers described in SSAR Section 13.6.3.6 remained as described in RAI Q910.13; that is, consideration may need to be given to how accessible, isolated, and hidden from view these barriers will be, as well as whether they can be penetrated with hand tools available on site. While SSAR Section 13.6.3.6 only addresses the main control room heating, ventilation, and air conditioning (HVAC) ducting and exterior air exhaust systems, SSAR Chapter 19, Appendix 19B.2.4(13) was changed to include the EPRI ALWR requirements on utility port openings (e.g., HVAC, cooling, and piping) through all vital or protected area boundaries, in accordance with EPRI

Evolutionary Requirements Document, Chapter 9, Section 5.2.5.1, Revision 0. Specifically, the SSAR states that the ABWR design will minimize the use of utility port openings through all vital or protected area boundaries and will provide security access control of these utility ports. GE's change in Appendix 19B.2.4, which clarifies that the ABWR design will comply with the above ALWR requirements, satisfactorily resolved this issue.

Certified Design Material

The staff expects that at least 60 days before loading fuel, the COL applicant referencing the ABWR design will have confirmed that the as-built bullet-resistant feature of walls and doors and the penetration-resistant feature of barriers in HVAC ducting and exhausts, committed to in SSAR Section 13.6.3.6, have been installed in all locations required by the commitment. This inspection requirement should be included in appropriate building ITAAC and was identified as DFSER Open Item 13.6.3.6-1. In its submittal dated May 30, 1992, GE stated that the issue was not a Tier 1 ITAAC issue. Additional review by the staff agreed with GE that the issue did not meet the criteria to be a Tier 1 ITAAC requirement. This issue is discussed in detail in Section 14.3.2.5. Therefore, this item is resolved.

13.6.3.7 Compatibility With the Remainder of the Plant

SSAR Section 13.6.3.7 states that access control for the remainder of plant buildings, including the turbine building side of the main steam tunnel, must be compatible with the site-specific physical security program which will be the responsibility of the COL applicant who references the ABWR standard plant design. The staff agrees and notes that an acceptable security barrier to bar unauthorized access from the turbine building into the steam tunnel must also permit venting of steam into the turbine building in accordance with SSAR Section 9A.3.2 of Appendix A to Chapter 9.

14 INITIAL TEST PROGRAM

14.1 Preliminary Safety Analysis Reports Information

Section 14.1 of the standard safety analysis report (SSAR) is specified in Regulatory Guide (RG) 1.70 and the standard review plan (SRP). However, the preliminary safety analysis report information specified by RG 1.70 is only required for facilities licensed under 10 CFR Part 50, and is not required for a design certification under 10 CFR Part 52. This is because under Part 52, the staff gives its final design approval (FDA) for the design, rather than its preliminary design approval as done under Part 50. Thus, the design information in the SSAR is final rather than preliminary, and therefore, no information is required to be in this section of the ABWR SSAR.

The staff provided its preliminary evaluation of GE's draft certified design material in Section 14.1 of the DFSER. Subsequently, as requested by the staff, GE provided SSAR information discussing the certified design material in a new Section 14.3 of the SSAR. The staff's review of the certified design information is summarized in Section 14.3 of this report, which conforms with SSAR Section 14.3.

This section is not applicable to the ABWR application. The DFSER included a discussion of piping design in this section. This discussion has been moved to Section 3.12 of this report.

14.2 Initial Plant Test Programs

Introduction

Chapter 14, Section 14.2, of the ABWR SSAR, "Specific Information to be Included in Final Safety Analysis Reports," describes the plant initial test program (ITP) for the ABWR. The GE ABWR ITP consists of a series of tests categorized as construction, preoperational, or initial startup tests. The construction acceptance tests determine installation and functionality of equipment. Preoperational tests are those tests normally conducted prior to fuel loading to demonstrate the capability of plant systems to meet performance requirements. Initial startup tests begin with fuel loading and demonstrate the capability of the integrated plant to meet performance requirements. This report documents the staff's evaluation of the GE ABWR ITP.

The staff performed the review of the plant ITPs in accordance with Section 14.2, of NUREG-0800, the SRP, and RG 1.68 (Rev. 2), "Initial Test Program for Water Cooled Nuclear Power Plants." The staff reviewed thirteen areas relating to initial plant test programs, described in Chapter 14 of the SSAR, and submitted by GE as part of its

design certification application. These areas of review are listed below:

- Summary of the Test Program and Objectives
- Organization and Staffing
- Test Procedures
- Conduct of Test Program
- Review, Evaluation, and Approval of Test Results
- Test Records
- Conformance of Test Program With Regulatory Guides
- Utilization of Reactor Operating and Testing Experience in the Development of Test Program
- Trial Use of Plant Operating and Emergency Procedures
- Initial Fuel Loading and Initial Criticality
- Test Program Schedule
- Individual Test Descriptions
- Combined License (COL) Information - ITP

The acceptance criteria used by the staff for this review were contained in SRP Section 14.2, Subsection II.

Background

A meeting was held with GE, on May 7, 1991, in order to discuss a list of questions, comments, and errata information generated by the staff's initial review of the SSAR and GE's potential responses to these items. Subsequently, a draft SSAR amendment was submitted by GE via a letter dated May 20, 1991, from R.C. Mitchell (GE) to C.L. Miller (NRC), in response to these items.

The DSER (SECY-91-355), dated November 5, 1991, consisting of acceptable and open items, was subsequently forwarded to GE for its use in further revising the SSAR. In October 1991, GE formally submitted SSAR Chapter 14, Amendment 20. GE subsequently submitted a response to the DSER open items via letter dated March 11, 1992, from R.C. Mitchell (GE) to R.C. Pierson (NRC) and additional SSAR markups were subsequently provided to the staff. These changes were incorporated in SSAR Amendments 21, 22, and 23.

Initial Test Program

Evaluation

The staff reviewed the GE ABWR ITP in accordance with SRP Section 14.2. This evaluation includes information in staff requests for additional information (RAIs) following its earlier review, GE's letter of May 20, 1991, in response to the staff's RAIs and the staff's findings regarding each response. The staff also evaluated GE's responses to DSER open items as contained in its letter of March 11, 1992. This evaluation also includes all other information supplied to the staff as docketed markups to the SSAR, as well as changes contained in subsequent SSAR amendments. This evaluation for the ABWR included open items, confirmatory items, and resolution of those sections of the SSAR that the staff initially found to need modification or additional information.

Based on the review of the GE ABWR ITP description in SSAR Chapter 14 and the responses to the RAIs and the DSER open items, DFER open items and confirmatory items, the staff concludes that the ITP was generally comprehensive and covered all areas of staff concern. The staff also conducted an in-depth review of system-specific testing requirements within each test abstract. The staff concludes that GE provided a sufficient level of detail to adequately describe system-specific test prerequisites and acceptance criteria. The staff also reviewed cross references to acceptance criteria information in other parts of the SSAR and finds them acceptable.

14.2.1 Summary of Test Program and Objectives

As stated in SSAR Section 14.2.1, the objectives of the ITP will be to

- ensure that construction is completed and acceptable
- demonstrate the capability of structures, components, and systems to meet performance requirements
- effect fuel loading in a safe manner
- demonstrate, where practical, that the plant is capable of withstanding anticipated transients and postulated accidents
- evaluate and demonstrate, to the extent possible, that plant operating procedures provide assurance that the operating group is knowledgeable about the plant and procedures and fully prepared to operate the facility in a safe manner
- bring the plant to rated capacity and sustained power operation

14.2.1.1 Construction Test Objectives

SSAR Section 14.2.1.1 states that construction tests will be performed to demonstrate that components and systems are correctly installed and operational. These tests will include, but will not be limited to; flushing and cleaning, hydrostatic testing, initial calibration of instrumentation, checks of electrical wiring and equipment, valve testing, and initial energization and operation of equipment and systems. Completion of this phase will ensure that systems are ready for preoperational testing.

14.2.1.2 Preoperational Test Objectives

SSAR Section 14.2.1.2 states that preoperational tests will be conducted before fuel loading to verify that plant systems are capable of operating in a safe and efficient manner compatible with the system design bases. The general objectives of the preoperational test phase will be to:

- ensure that design specification and test acceptance criteria are met
- provide documentation of the performance and safety of equipment and systems
- provide baseline test and operating data on equipment and systems for future reference
- run-in new equipment for a sufficient period so that any design, manufacturing, or installation defects can be detected and corrected
- ensure that plant systems operate together on an integrated basis to the extent possible
- give maximum opportunity to the permanent plant operating staff to obtain practical experience in the operation and maintenance of equipment and systems
- help demonstrate safe and efficient system operating and surveillance testing procedures to the extent possible
- demonstrate that safety systems and equipment are operated to allow fuel loading and entry to the startup phase

14.2.1.3 Startup Test Objectives

SSAR Section 14.2.1.3 states that after the preoperational test phase has been completed, the startup phase will begin with fuel loading and extend to commercial operation. The tests conducted during the startup phase consist of major

and minor plant transients, steady-state tests, and process control system tests. These tests will be directed toward demonstrating correct performance of the nuclear boiler and the various plant systems while at power and the validation of analytical models used in the design.

The general objectives of the startup phase will be to:

- achieve an orderly and safe initial core loading
- accomplish all testing and measurements necessary to ensure that the approach to initial criticality and subsequent power ascension is safe and orderly
- conduct low-power physics tests sufficient to ensure that test acceptance criteria have been met
- conduct initial heatup and hot functional testing so that hot integrated operation of all systems is shown to meet test acceptance criteria
- conduct an orderly and safe power ascension program, with requisite physics and systems testing, to ensure that integrated plant operation at power meets test acceptance criteria
- demonstrate, to the extent possible, the adequacy of various component, system, and plant procedures
- conduct a successful warranty demonstration

14.2.1.4 Conclusion

The staff finds that the objectives for the ITP meet the acceptance criteria in SRP Section 14.2 and are acceptable.

14.2.2 Organization and Staffing

14.2.2.1 Normal Plant Staff

SSAR Section 14.2.2.1 states that the normal plant staff responsibilities, authorities, and qualifications are given in SSAR Chapter 13. DFSE Confirmatory Item 14.2.2.1-1 addressed an issue that plant organizational staff was outside the scope of the ABWR SSAR. GE added information in Amendment 30 of SSAR Section 14.2.2.1 to state that plant organizational staff will be provided by the COL applicant. The staff finds this acceptable. This resolved DFSE Confirmatory Item 14.2.2.1-1.

14.2.2.2 Startup Group

SSAR Section 14.2.2.2 states that the startup group will be an ad hoc organization created to ensure that the ITP is conducted in an efficient, safe, and timely manner. The

startup group will be responsible for planning, executing, and documenting all startup and testing activities that occur between the completion of the construction phase and commencement of commercial operation of the plant. Upon completion of the startup program, the startup group will be dissolved and the normal plant staff will assume complete responsibility for the plant. The normal plant staff will be included in as many aspects of the test programs as is practicable, considering their normal duties in the operation and maintenance of the plant.

14.2.2.3 General Electric Company

SSAR Section 14.2.2.3 states that GE will be the supplier of the boiling water reactor (BWR) nuclear steam supply system and is responsible for generic and specific BWR designs. During the construction and testing phases of the plant cycle, GE personnel will be on site to offer consultation and technical direction with regard to GE-supplied systems and equipment.

14.2.2.4 Others

SSAR Section 14.2.2.4 states that other concerned parties outside the plant staff organization--such as the architect-engineer, the constructor, the turbine-generator supplier, and vendors of other equipment--will be involved in the testing program to various degrees.

14.2.2.5 Interrelationships and Interfaces

SSAR Section 14.2.2.5 states that the effective coordination between the various site organizations involved in the test program will be achieved through the startup coordinating group (SCG) that will be composed of representatives of the plant owner/operator, GE, and others. The duties of the SCG will be to review and approve project testing schedules and to effect timely changes to construction or testing in order to facilitate execution of the preoperational and initial startup test programs.

14.2.2.6 Conclusion

The staff finds that the organization and staffing plan meet the acceptance criteria in SRP Section 14.2 and are acceptable. As discussed in Section 14.2.2.1, the normal plant staff aspects will be evaluated during the COL review.

14.2.3 Test Procedures

SSAR Section 14.2.3 states that testing during all phases of the ITP will be conducted using detailed, step-by-step written procedures to control the conduct of each test.

Initial Test Program

Such test procedures will specify testing prerequisites, describe desired initial conditions, include appropriate methods to direct and control test performance (including the sequencing of testing), specify acceptance criteria by which the test will be evaluated, and provide for or specify the format by which data or observations will be recorded. The procedures will be developed, reviewed, and controlled by personnel with appropriate technical backgrounds and experience in accordance with the startup administrative manual. This will include participation of principal design organizations in the establishment of test performance requirements and acceptance criteria. GE will provide the COL applicant with scoping documents (i.e., preoperational and startup test specifications) containing testing objectives and acceptance criteria applicable to its scope of design responsibility. Such documents also will include, as appropriate, delineation of specific plant operational conditions at which tests will be conducted, testing methodologies to be used, specific data to be collected, and acceptable data reduction techniques as well as any reconciliation methods needed to account for test conditions, methods, or results if testing is performed at conditions other than representative design operating conditions. Available information on operating and testing experiences of operating power reactors will be factored into test procedures as appropriate.

The staff finds that the content, development, and review of test procedures meet the criteria in SRP Section 14.2, and are acceptable. However, a COL applicant will need to provide the following for staff review:

- The scoping document (i.e., preoperational and startup test specifications) containing testing objectives and acceptance criteria applicable to its scope of design responsibility. This was DFSER COL Action Item 14.2.3-1.
- The scoping document that delineate and any other documents which delineate plant operational conditions at which tests are to be conducted, testing methodologies to be utilized, specific data to be collected, and acceptable data reduction techniques to be reviewed by the staff at the time of combined operating license. This was DFSER COL Action Item 14.2.3-2.
- The scoping document that delineate any reconciliation methods needed to account for test conditions, methods or results if testing is performed at conditions other than representative design operating conditions. This was DFSER COL Action Item 14.2.3-3.
- The approved preoperational test procedures approximately 60 days before their intended use and

startup test procedures approximately 60 days before fuel loading. This was DFSER COL Action Item 14.2.3-4.

GE incorporated the above COL action items in SSAR Sections 14.2.3 and 14.2.13. This is acceptable.

14.2.4 Conduct of Test Program

SSAR Section 14.2.4 states that the ITP will be conducted by the startup group in accordance with the startup administrative manual. This manual will contain the administrative procedures and requirements that govern the activities of the startup group and their interfaces with other organizations. The startup administrative manual will receive the same level of review and approval as do other plant administrative procedures. It will define the specific format and content of preoperational and startup test procedures as well as the review and approval process for both initial procedures and subsequent revisions or changes. The startup manual also will specify the process for review and approval of test results and for resolution of failures to meet acceptance criteria and of other operational problems or design deficiencies. It will describe the various phases of the ITP and establish the requirements for progressing from one phase to the next as well as those for moving beyond selected hold points or milestones within a given phase. It will also describe the controls in place that will ensure the as-tested status of each system is known and track modifications, including retest requirements, deemed necessary for systems undergoing or already having completed specified testing. Additionally, the startup manual will delineate the qualifications and responsibilities of the different positions within the startup group.

Staff Evaluation of DSER Items

In the DSER, the staff determined that GE should specify whose approval must be obtained before increasing power to the next higher test plateau. (This was incorrectly identified as an interface requirement in Section 14.2.5 of the DSER). GE indicated that such specifics will be a function of the plant owner/operator's unique organizational structure and detailed plant administrative procedures and thus left to the COL applicant. GE revised SSAR Section 14.2.13 to reflect that the COL applicant should specify whose approval is needed to proceed to the next testing plateau. This is acceptable.

The staff finds that the conduct of the test program meets the acceptance criteria in SRP Section 14.2 and is acceptable. However, a COL applicant will need to provide a startup administrative manual (procedures) and any other documents that delineate the conduct of the test

program to be reviewed by the staff at the time of the COL application. This was DFSER COL Action Item 14.2.4-1. GE included the above action item in SSAR Section 14.2.13 and the staff finds this acceptable.

14.2.5 Review, Evaluation, and Approval of Test Results

SSAR Section 14.2.5 states that individual test results will be evaluated and reviewed by cognizant members of the startup group. Test exceptions or acceptance criteria discrepancies will be communicated to the affected and responsible organizations who will help resolve the issues by suggesting corrective actions, design modifications, and retests. GE and others outside the plant staff organization, as appropriate, will have the opportunity to review the results for conformance to predictions and exceptions. Test results, including final resolutions, then will be reviewed and approved by designated startup group supervisory personnel.

Staff Evaluation of DSER Items

In the DSER, the staff determined that GE should specify whose approval must be obtained before increasing power to the next higher test plateau. GE indicated that such specifics will be a function of the plant owner/operator's unique organizational structure and detailed plant administrative procedures and are thus will be defined by the COL applicant. GE revised SSAR Section 14.2.13 to reflect that the COL applicant should specify whose approval is needed to proceed to the next testing plateau. This is acceptable.

The staff finds that the process for review, evaluation, and approval of test results meets the acceptance criteria described in SRP Section 14.2 and is acceptable. However, a COL applicant will need to provide a startup administrative manual (procedures) and any other documents that delineate the review, evaluation, and approval of test results for staff review. This was DFSER COL Action Item 14.2.5-1. GE included the above COL action item in SSAR Section 14.2.13 and the staff finds this acceptable.

14.2.6 Test Records

SSAR Section 14.2.6 states that the ITP results will be compiled and maintained according to the startup manual, plant administrative procedures, and applicable regulatory requirements. Test records that demonstrate the adequacy of safety-related components, systems, and structures will be retained for the life of the plant. Retention periods for other test records will be based on consideration of their

usefulness in documenting initial plant performance characteristics. As discussed in SER Section 14.2.4, the startup administrative manual is identified as a COL action item.

The staff finds that the test records compilation, maintenance, and retention program meets the acceptance criteria in SRP Section 14.2, and is acceptable.

14.2.7 Conformance of Test Program With Regulatory Guides

SSAR Section 14.2.7 lists the NRC RGs that will be used in the development of the ITP and states that the applicable tests will comply with these guides. The applicable revisions to these RGs are listed in SSAR Table 1.8-20.

In the DSER, the staff determined that GE needed to add additional RG references to Table 1.8-20 and SSAR Section 14.2.7. This was DSER Open Item 6.C. The DFSER tracked this issue as DFSER Confirmatory Item 14.2.7-1.

In its letter of May 20, 1991, GE agreed to amend the SSAR to include RGs 1.95, "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release," Revision 1, and 1.139, "Guidance for Residual Heat Removal," Revision 0, and to document the applicable revision number of each RG listed in SSAR Section 14.2.7 or to amend the section to reference Table 1.8-20 for the applicable revision numbers of the listed guide. GE made these changes in Amendment 18 of the SSAR. GE further agreed to correct the reference to RG 1.68.3, "Preoperational Testing of Instrument and Control Air," Revision 0, contained in SSAR Table 1.8-20 or Section 14.2.7, as appropriate, to Revision 0, issue date of April 1982. GE stated this change in its response on March 11, 1992. The staff determined that GE made the required changes in Amendment 21 of the SSAR. The staff finds this acceptable. This resolved the above portions of DFSER Confirmatory Item 14.2.7-1.

The staff's review identified that SSAR Section 14.2.7 lists RG 1.140 as applicable to the ABWR, however, SSAR Table 1.8-20 indicates RG 1.140 is not applicable to the ABWR. SSAR Section 14.2.12.1.34(3)(1) includes RG 1.140 guidance as acceptance criteria for visual inspection and airflow distribution, testing for penetration of dioctyl phthalate and bypass leakage testing for high-efficiency particulate air (HEPA) and charcoal absorber sections where installed. The staff found that the utilization of RG 1.140 guidance for normal ventilation exhaust system air filtration and absorber units was not clear. GE was asked to make the appropriate clarifications to Table 1.8-20, Section 14.2.7, and individual test

Initial Test Program

abstracts, as appropriate, to clearly and consistently describe the degree of conformance to RG 1.140, any exceptions taken, and to identify any aspects of RG 1.140 that are not applicable to the ABWR.

GE responded by deleting the reference to RG 1.140 in SSAR Amendment 30 from the list of acceptance criteria in SSAR Section 14.2.12.1.34 and from the list of RGs in SSAR Section 14.2.7 to make those sections consistent with Table 1.8-20. GE added RG 1.52 to Section 14.2.12.1.34 and Section 14.2.7 in SSAR Amendment 30 to address testing of engineered safety feature (ESF) heating, ventilation, and air conditioning (HVAC) systems that require in place testing of HEPA and carbon adsorber filters. The staff finds this acceptable. This resolved DFSEF Confirmatory Item 14.2.7-1.

14.2.8 Utilization of Reactor Operating and Testing Experience in the Development of Test Program

SSAR Section 14.2.8 states that since every reactor/plant in a GE BWR product line is an evolutionary development of the previous plant in the product line it is evident that the ABWR plants will benefit from the experience acquired with the successful and safe startup of more than 30 previous BWR plants. The operational experience and knowledge gained from these plants and other reactor types have been factored into the design and test specifications of GE-supplied systems and equipment that will be demonstrated during the preoperational and startup test programs. Additionally, reactor operating and testing experience of similar nuclear power plants obtained from NRC licensee event reports and through other industry sources will be used to the extent practicable in developing and carrying out the ITP.

The staff finds that the use of reactor operating and testing experience in the development of the test program meets the acceptance criteria in SRP Section 14.2, and is acceptable.

14.2.9 Trial Use of Plant Operating and Emergency Procedures

SSAR Section 14.2.9 states that, to the extent practicable, throughout the preoperational and initial startup test program, operating, emergency, and abnormal procedures will be incorporated, where applicable, in the performance of tests. The use of these procedures is intended to

- prove the adequacy of the specific procedure or illustrate changes that may be required

- provide training of plant personnel in the use of these procedures
- increase the level of knowledge of plant personnel on the systems being tested

GE further indicated, that to meet the above goals, test procedures will actually reference or extract steps from operating, emergency, or abnormal procedures.

The staff finds that the method of incorporating operating, emergency, or abnormal procedures into the test program meets the acceptance criteria in SRP Section 14.2, and is acceptable. However, as discussed in SSAR Section 14.2.3, a COL applicant will need to provide the approved preoperational test procedures 60 days before their intended use and the startup test procedures approximately 60 days before fuel loading.

14.2.10 Initial Fuel Loading and Initial Criticality

SSAR Section 14.2.10 states that the fuel loading and initial criticality will be conducted in a very controlled manner in accordance with specific written procedures as part of the startup test phase. The NRC approves fuel loading after it has verified that prerequisite testing has been satisfactorily completed or after the COL applicant provides appropriate justification to proceed with fuel loading and complete the preoperational testing after fuel loading.

The intent of the testing program is to complete all preoperational tests and approve the results before commencement of fuel loading. However, there may be unforeseen circumstances that arise that would prevent this from occurring, but that would not necessarily justify the delay of fuel loading.

In the DFSEF, the staff determined that SSAR Section 14.2.10 should address the completion of preoperational testing (including the review and approval of test results required before fuel loading). If portions of any preoperational tests are intended to be conducted, or their results approved, after fuel loading, GE should (1) list each test, (2) state which portions of each test will be delayed until after fuel loading, (3) provide technical justification for delaying these portions, and (4) state when each test will be completed and the results approved.

GE revised SSAR Section 14.2.10 in Amendment 18 to require that the above stated conditions be appropriately documented should the COL applicant decide to request permission from the NRC to proceed with fuel loading under such circumstances. In the DFSEF, the staff found that the methods of controlling pre-fuel load checks, initial

fuel loading, precritical testing and initial criticality outlined in Section 14.2.10 met the acceptance criteria in SRP Section 14.2, and were acceptable. However, a referencing applicant will need to provide a startup administrative manual (procedures) and any other documents that delineate the methods of controlling pre-fuel load checks, initial fuel loading, precritical testing and initial criticality. These are to be reviewed by the staff before issuing a COL. This was DFSER COL Action Item 14.2.10.4-1. GE included this COL action item in SSAR Sections 14.2.10 and 14.2.13. The staff finds this acceptable.

14.2.10.1 Pre-Fuel Load Checks

SSAR Section 14.2.10.1 states that once the plant has been declared ready to load fuel, there are a number of specific checks that must be made before proceeding. These include a final review of the preoperational test results and the status of any design changes, work packages, and/or retests that were initiated as a result of exceptions noted during this phase. Also, the technical specifications (TS) surveillance program requirements, as described in the ABWR TS Chapter 16, will be instituted at this time to assure the operability of systems required for fuel loading. Just before the initiation of fuel loading, the proper vessel water level and chemistry will be verified and the calibration and response of the nuclear instruments will be checked. This was DFSER TS Item 14.2.10.1-1.

The staff reviewed GE submittals dated March 31 and April 16, 1993, which contained GE's response to resolve TS Item 14.2.10.1-1. SSAR Section 14.2.12.2.3(2), prerequisites (c) and (d) will require neutron detectors instrument channels to be properly calibrated and operable prior to fuel loading. A cross-reference to a response dated March 31, 1993, of TS LCO 3.3.1.3, Surveillance Requirements (SR) for startup range nuclear monitor instrumentation, excluded calibration of neutron detectors. The staff determined that calibration of neutron detector instrumentation is required before fuel loading.

The ABWR Standardized Technical Specification (STS) bases section for SR 3.3.2.1.6 states "the neutron detectors are excluded from channel calibration because they cannot readily be adjusted. The detectors are fission chambers that are designed to have relatively constant sensitivity over the range, and with an accuracy specified for a fixed useful life." The staff determined that the startup test program prerequisite requirements verify calibration of the neutron detectors instrumentation channels before fuel loading and before startup. Neutron sources would be used to verify proper calibration and instrument response of the detectors. The ABWR STS do not require the detectors to be calibrated during startup, therefore, the staff determined

that the cross reference to the STS in SSAR Section 14.2.12.2-3 should be deleted. The staff verified that the cross reference to the STS was deleted from acceptance criteria 14.2.12.2-3(2)(c) and (d) in SSAR Amendment 31. The acceptability of the ABWR TS is discussed in Chapter 16 of this report. This resolved DFSER TS Item 14.2.10.1-1.

14.2.10.2 Initial Fuel Loading

SSAR Section 14.2.10.2 states that fuel loading will require the movement of the full core complement of assemblies from the fuel pool to the core, with each assembly being identified by number before being placed in the correct coordinate position. The procedure controlling this movement will specify that partial core shutdown margin and subcritical checks be made at predetermined intervals throughout the loading, thus ensuring safe loading increments as described in test abstract 14.2.12.2.3, Fuel Loading. In-vessel neutron monitors will provide continuous indication of the core flux level as each assembly is added. A complete check will be made of the fully loaded core to ascertain that all assemblies are properly installed, correctly oriented, and occupying their designated positions.

14.2.10.3 Pre-Criticality Testing

The control rods shall be verified functional and scram tested with the fuel in place. The post-fuel-load flow test of the reactor internals vibration assessment program, if applicable, shall be conducted at this time. Additionally, a final verification shall be made that the required TS surveillances have been performed.

14.2.10.4 Initial Criticality

SSAR Section 14.2.10.4 states that during initial criticality, the full core shutdown margin shall be verified for the fully loaded core as described in startup test abstract 14.2.12.2.4, "Full Core Shutdown Margin Demonstration." SSAR Section 14.2.10.4 also states that initial criticality shall be achieved in an orderly, controlled fashion following specific detailed procedures in an approved rod withdrawal sequence. Core neutron flux shall be continuously monitored during the approach to criticality and periodically compared to predictions to allow early detection and evaluation of potential anomalies.

14.2.11 Test Program Schedule

SSAR Section 14.2.11 states that the schedule, relative to the initial fuel load date, for conducting each major phase of the ITP will be provided by the COL applicant. This includes the time table for generation, review, and

Initial Test Program

approval of procedures as well as the actual testing and analysis of results. As a minimum, at least 9 months before the fuel loading date should be allowed for conducting the preoperational phase and at least 3 months should be allowed for conducting the startup and power ascension testing that commences with fuel loading. To allow for NRC review, test procedure preparation for power ascension will be scheduled so that approved procedures are available approximately 60 days prior to fuel load. Although there will be considerable flexibility available in the sequencing of testing within a given phase, testing should be performed as systems are turned over from construction. However, the interdependency of systems should also be considered so that common support systems, such as electrical power distribution, service and instrument air, and the various makeup water and cooling water systems, will be tested as early as possible. Sequencing of testing during the startup phase will depend primarily on specified power and flow conditions and inter-system prerequisites. To the extent practicable, the schedule should establish that, before exceeding 25-percent power, the test requirements will be met for those plant structures, systems, and components (SSCs) that are relied on to prevent, limit, or mitigate the consequences of postulated accidents. Additionally, testing will be sequenced so that the safety of the plant is never totally dependent on untested systems, components, or features.

Staff Evaluation of DSER Items

In the DSER, the staff determined that SSAR Section 14.2.11 should be modified to include the following:

- A figure that illustrates the power-flow operating map
- A table that lists the startup tests and states at which test condition(s) each test is to be conducted

The staff stated in DSER Open Items 116 and 117 that SSAR Figure 4.4-1 did not provide sufficient detail regarding test condition identification to determine that each startup test will be conducted at appropriate power-flow conditions in accordance with RG 1.68, "Initial Test Programs for Water Cooled Nuclear Power Plant" (Rev. 2), Appendix A.5, "Power Ascension Tests." In addition, the SSAR did not contain a table of startup tests.

GE subsequently submitted a power-to-flow operating map, SSAR Figure 14.2-1, that provides an appropriate indication of test conditions and GE provided a table of startup tests, SSAR Table 14.2-1, "Startup Test Matrix." This was acceptable subject to incorporation into a future SSAR revision and staff review. This resolved DSER

Open Items 116 and 117, and became DFSER Confirmatory Item 14.2.11-1. The staff finds that the power-to-flow map and the table of startup tests that were incorporated into Amendment 23 of the SSAR meet the acceptance criteria of SRP Section 14.2, and are acceptable. This resolved DFSER Confirmatory Item 14.2.11-1.

GE indicated that SSAR Section 14.2.11 would be revised to include the following additional information on the test program schedule. Power ascension testing will be conducted in essentially three phases: (1) initial fuel loading and open vessel testing, (2) testing during nuclear heat up to rated temperature and pressure, and (3) power operation testing from 5-percent to 100-percent rated power. Further, power operation testing will be divided into three sequential testing plateaus as shown on SSAR Figure 14.2-1. The testing plateaus consist of low-power testing at less than 25-percent power, mid-power testing up to 75-percent power between approximately the 50-percent and 75-percent rod lines, and high-power testing along the 100-percent rod line up to rated power. Thus, there will be a total of five different testing plateaus designated as shown on SSAR Figure 14.2-1. Table 14.2-1 indicates in which testing plateaus the various power ascension tests will be performed. Although the order of testing within a given plateau will be somewhat flexible, the normal recommended sequence of tests will be; (1) core performance analysis, (2) steady state tests, (3) control system tuning, (4) system transient tests, and (5) major plant transients (including trips). Also, for a given testing plateau, testing at lower power levels generally should be performed before that at higher power levels. The detailed testing schedule will be generated by the COL applicant and will be made available to the NRC before actual implementation. The schedule then will be maintained to reflect actual progress and subsequent revised projections. The information above was acceptable subject to incorporation into a future SSAR revision. This was DFSER Confirmatory Item 14.2.11-2.

GE revised the test program schedule in SSAR Section 14.2.11 and Table 14.2-1 in SSAR Amendments 23 and 33. The staff finds the changes acceptable. This resolved DFSER Confirmatory Item 14.2.11-2.

Additionally, a COL applicant will need to provide a startup administrative manual (procedures) and any other documents that delineate the test program schedule for staff review. GE identified this as a COL action item in a response dated April 6, 1993. This information was incorporated into SSAR Amendment 31. This was DFSER COL Action Item 14.2.11-1. GE included this COL action item in SSAR Section 14.2.3, and the staff finds this acceptable.

14.2.12 Individual Test Descriptions

14.2.12.1 Preoperational Test Procedures

SSAR Section 14.2.12.1 states that the general descriptions relate the objectives of each preoperational test. During the final construction phase, it may be necessary to modify the preoperational test methods as operating and preoperational test procedures are developed. Consequently, methods in the descriptions are general, not specific.

Specific testing to be performed and the applicable acceptance criteria for each preoperational test will be documented in detailed test procedures to be made available to the NRC approximately 60 days before their intended use. Preoperational testing will be in accordance with the detailed system specifications and associated equipment specifications for equipment in those systems (provided as part of scoping documents to be supplied by GE and others as described in SSAR Subsection 14.2.3). The tests will demonstrate that the installed equipment and systems will perform within the limits of these specifications. To allow for verification that the detailed test procedures are developed in accordance with established methods and appropriate acceptance criteria, the plant and system preoperational test specifications will also be made available to the NRC.

The preoperational tests anticipated for the ABWR standard plant design are listed and described in the SSAR Subsection 14.2.12.1. The staff finds that the scope of preoperational tests described in SSAR Subsection 14.2.12.1 meets the acceptance criteria of SRP Section 14.2, and is acceptable. Testing of systems out of the scope of the SSAR are discussed in SSAR Section 14.2.13 along with other COL information related to the ITP.

14.2.12.2 General Discussion of Startup Tests

SSAR Section 14.2.12.2 discusses those tests proposed and expected to comprise the startup test phase. For each test a general description is provided for test purpose, test prerequisites, test description, and test acceptance criteria, where applicable. Because changes will occur as the test program is developed and implemented, the descriptions remain general in scope. However, an attempt is made in describing a test to identify those operating and safety-oriented characteristics of the plant design that are being explored and evaluated.

Where applicable, the relevant acceptance criteria for the test are discussed. Some of the criteria relate to the value of process variables assigned in the design or analysis of

the plant, component systems, and associated equipment. If a criterion of this nature is not satisfied, the plant will be placed in a suitable hold condition until resolution is obtained. Tests compatible with this hold condition may be continued. Following resolution, applicable tests or portions of these tests may be repeated to verify that the requirements of the criterion are ultimately satisfied. Other criteria may be associated with expectations relating to the performance of systems. If this type of criterion is not satisfied, operating and testing plans would not necessarily be altered. However, investigations of the measurements and of the analytical techniques used for the predictions would be started. Specific actions for dealing with criteria failures and other testing exceptions or anomalies will be described in the startup administrative manual.

The specifics of the startup tests relating to test methodology, plant prerequisites, initial conditions, acceptance criteria, and analysis techniques will come from the appropriate design and engineering organizations of the COL applicant in the form of plant, system, and component performance and testing specifications. The COL applicant shall provide test documents for the Office of Nuclear Reactor Regulation review as discussed in SSAR Section 14.2.3.

14.2.12.3 Staff Evaluation of DSER Items

In the DSER, the staff determined that testing of systems outside the scope of the ABWR standard plant design should be included or referenced in SSAR Section 14.2.12. (This was not tracked as a separate open item.)

GE revised Section 14.2.12.1 in SSAR Amendment 18 to state that testing of systems outside the scope of the ABWR standard plant design are discussed in SSAR Section 14.2.13. The staff finds this acceptable.

In the DSER, the staff determined that SSAR Section 14.2.12 test abstracts should be modified to address the following concerns:

- (1) Several preoperational and startup test prerequisites include the requirement that interfacing support systems will be available.

The staff asked GE to identify which support systems will be required for each test and to specify which individuals or groups will be authorized to make this determination. (This was incorrectly identified as an interface requirement.)

GE stated that the interfacing support system requirements will be specified in the detailed test

Initial Test Program

procedures (and operating and maintenance procedures, if appropriate) that are required by RG 1.68 to be made available to NRC personnel at least 60 days before the intended use of preoperational tests procedures and 60 days before fuel loading for startup test procedures. Additionally, the startup administrative manual will delineate how such determinations of operability and availability will be authorized. Thus, these details are the responsibility of the COL applicant. This was DFSER COL Action Item 14.2.12.3-1. GE included this action item in Section 14.2.13, and the staff finds this acceptable.

In the DSER, the staff determined that the level of detail in the test abstracts was insufficient to determine conformance with RG 1.68, Position C.2. This was DSER Open Item 118. The individual test abstracts in SSAR Sections 14.2.12.1.1, 14.2.12.1.4, 14.2.12.1.7, 14.2.12.1.11, 14.2.12.1.12, 14.2.12.1.13, 14.2.12.1.18, 14.2.12.1.21, 14.2.12.1.22, 14.2.12.1.43, 14.2.12.1.44, 14.2.12.1.45.1, 14.2.12.1.45.2, 14.2.12.1.45.3, 14.2.12.1.45.4, 14.2.12.1.53, 14.2.12.1.59, 14.2.12.1.67, 14.2.12.1.68, and 14.2.12.1.69 did not specify basic systems required to be available, interface systems, or criteria required as prerequisite or initial conditions for the preoperational tests. Specific prerequisites should be addressed in these individual test abstracts. This was tracked in the DFSER as Open Item 14.2.12.3-1.

The staff completed its review of GE's February 12, 1993, submittal, which addresses DFSER Open Item 14.2.12.3-1. The staff determined that the level of detail for specific preoperational test prerequisites was sufficient with the exception of two individual test abstracts. GE provided revisions to the prerequisite sections of preoperational test abstracts 14.2.12.1.4 and 14.2.12.1.44 in responses dated May 13 and 21, 1993. The staff finds both of these revisions acceptable. The revisions were incorporated into SSAR Amendment 30. This resolved DFSER Open Item 14.2.12.3-1.

- (2) The use of the word "should" in most, if not all test abstracts, is not a commitment by the COL applicant to perform certain tasks. It should, therefore, be reevaluated and revised accordingly (i.e., "will," "must"). This was DSER Open Item 119.

In the DFSER, the staff verified that GE incorporated the word change from "should" to either "will" or "shall" into most test abstracts. This was acceptable subject to incorporation into a future SSAR revision. This resolved DSER Open Item 119, and was tracked in the DFSER as DFSER Confirmatory Item 14.2.12.3-1. The above word change was incorporated into the applicable test abstracts in SSAR Amendment 23. The staff finds this acceptable. This resolved DFSER Confirmatory Item 14.2.12.3-1.

- (3) Several preoperational and startup test abstracts included imprecise acceptance criteria (e.g., "applicable intervals," "applicable design specifications," "specified amounts," "specified tolerances," "perform as specified," and "function properly"). This was DSER Open Item 120.

In the DFSER, the staff determined that GE should address in the individual test abstracts the bases for determining acceptable system and component performance. Acceptable criteria includes specific references to RGs, TS, assumptions used in the safety analysis, other ABWR SSAR sections, and applicable codes and standards.

GE indicated that SSAR Chapter 14 was written primarily to document the appropriate testing commitments contained in RG 1.68. In its May 20, 1991, submittal, GE indicated that precise acceptance criteria would be provided as part of the inspections, tests, analyses, and acceptance criteria (ITAAC) effort. However, GE did not provide this information in either the ITAAC or in SSAR Chapter 14. This was tracked in the DFSER as DFSER Open Item 14.2.12.3-2.

The staff reviewed the individual test abstracts submitted on February 12 and May 13, 1993, to assess the adequacy of the test abstract coverage of system-specific test requirements. Clarifications and additional cross-reference information for roughly half of the preoperational tests were determined necessary. The staff requested GE to supplement the SSAR with this additional information in order to determine that these tests adequately address system-specific test requirements.

GE submitted Amendment 30, Revision 0, dated June 7, 1993, to the staff which provided cross-reference acceptance criteria information for all but two of the preoperational test abstracts. The two tests were the integrated emergency core cooling

systems (ECCS) loss of offsite power (LOOP)/loss of coolant accident (LOCA) preoperational test and the extraction steam system preoperational test. GE provided the necessary cross references for these test abstracts in Revision 1 of SSAR Amendment 30. The staff found all of the cross-references acceptable. Additional information on system-specific testing requirements were also provided for all test abstracts. The staff finds this information acceptable. This resolved DFSER Open Item 14.2.12.3-2.

- (4) SSAR Section 14.2.12.2 states that failure to satisfy some acceptance criteria (e.g., those related to values of process variables important to plant design) will result in the plant being placed in a suitable hold position until resolution is obtained, while failure to satisfy other acceptance criteria (e.g., expectations relating to system performance) may only result in the need for further data analysis. This was DFSER Open Item 121.

In the DFSER, the staff determined that the distinction between these types of acceptance criteria was unclear, and that GE should clearly address the various types of acceptance criteria and the resultant actions for each type if unsatisfactory test results were obtained.

SSAR Section 14.2.12.2, as modified by GE, stated: "Specific actions for dealing with criteria failures and other testing exceptions or anomalies will be described in the startup administrative manual." This response to the staff's open item was not acceptable. Of 35 individual startup test abstracts in SSAR Section 14.2.12.2, 33 did not specify the required actions or precautions for dealing with criteria failures and other testing exceptions or anomalies. GE did not adequately modify Section 14.2.12.2 or the individual test abstracts to address the subject acceptance criteria or actions. This was DFSER Open Item 14.2.12.3-3.

The staff requested GE provide a general distinction between startup test level 1 and level 2 acceptance criteria, and to specify individual startup test abstract acceptance criteria as level 1 or level 2. Level 1 criteria relates to the value of the process variables assigned in the design or analysis of the plant, component systems, or associated equipment. If a level 1 criteria is not satisfied, the plant will be placed in a hold condition until resolution is obtained. A level 2 criterion is associated with expectations relating to the performance of systems.

If a level 2 criteria requirement is not satisfied then an engineering evaluation must be completed to verify that overall system performance is acceptable.

The staff reviewed a submittal by GE on April 16, 1993, of SSAR Section 14.2.12.2 which identified the distinction between startup test level 1 and level 2 acceptance criteria. The distinction between levels 1 and 2 acceptance criteria is found to be acceptable. This information was incorporated into SSAR Amendment 30.

The staff's review of individual startup test abstracts, submitted by GE on June 7, 1993, identified that more precise acceptance criteria has been provided and the criteria is acceptable. The staff also determined that the system-specific test requirements are adequately addressed. GE incorporated the individual startup test abstract acceptance criteria information, the startup test levels 1 and 2 acceptance criteria, and the system-specific startup test requirements into SSAR Amendment 30. The staff finds the startup testing abstracts as described in SSAR Amendment 30 acceptable. This resolved DFSER Open Item 14.2.12.3-3.

- (5) GE should identify startup tests listed in SSAR Section 14.2.12.2 that are not essential to the demonstration of conformance with design requirements for SSCs and design features that

- will be used for safe shutdown and cooldown of the reactor under normal plant conditions and for maintaining the reactor in a safe condition for an extended shutdown period
- will be used for safe shutdown and cooldown of the reactor under transient (infrequent or moderately frequent events) conditions and postulated accident conditions and for maintaining the reactor in a safe condition for an extended shutdown period following such conditions
- will be used for establishing conformance with safety limits or limiting conditions for operation that will be included in the facility TS
- are classified as engineered safety features (ESFs) or will be used to support or ensure the operations of ESFs within design limits

Initial Test Program

- are assumed to function or for which credit is taken in the accident analysis for the facility, as described in the SSAR
- will be used to process, store, control, or limit the release of radioactive materials

GE stated that the test abstracts contained in SSAR Section 14.2.12.2 are intended to meet the requirements of RG 1.68, updated and/or modified as necessary to reflect the actual ABWR design. In the DSER it was stated that a required screening will be performed by GE to identify and document any testing that is currently specified for systems that are not essential for demonstrating conformance with the aforementioned criteria. This was DSER Open Item 122.

GE revised SSAR Section 14.2.13 (in draft) to state that criteria contained in the RG 1.68, Position C.1, will be used to determine if any testing is currently specified for systems that are not essential for demonstrating conformance with the aforementioned criteria. The staff found this item acceptable subject to incorporation into a future SSAR revision. This resolved DSER Open Item 122 and was tracked in the DFSE as DFSE Confirmatory Item 14.2.12.3-2.

The testing described in Section 14.2.12.2 includes testing of a limited number of ABWR SSCs, and design features that do not meet the referenced RG 1.68 criteria. The staff subsequently determined that it was premature to develop a specific list of tests that do not meet the criteria in RG 1.68, Position C.1. The staff has not yet defined the applicable COL license conditions related to such tests (i.e., timely notification to the NRC for major test changes that affect systems that meet the criteria in RG 1.68, Position C.1). This aspect will be further reviewed for a COL applicant as part of the COL action items listed in SSAR Section 14.2.13. Therefore, GE did not include this information in the SSAR. The staff finds this acceptable. This resolved DFSE Confirmatory Item 14.2.12.3-2.

14.2.12.4 Conformance of the ABWR with RG 1.68, Revision 2

In the DSER, the staff's review of the preoperational and startup test phase descriptions disclosed that the operability of several of the systems and components listed in RG 1.68 may not be adequately demonstrated by the tests described in the SSAR. This was DSER Open Item 123.

Each item is evaluated separately below. This evaluation resolved DSER Open Item 123.

Staff Evaluation of DSER Items

The staff determined that GE should either expand the test descriptions to address the following items, insert cross-references in SSAR Section 14.2.12 if complete test descriptions for the following items are provided elsewhere in the SSAR, or modify SSAR Section 14.2.7 or Table 1.8-20 of the SSAR, as appropriate, to provide technical justification for any exception to RG 1.68. It was stated in the DSER that the following items should be reflected in a subsequent amendment to the SSAR. (Note: Each item is numbered in accordance with RG 1.68.)

- 1.a.(2)(d) Supports and restraints for discharge piping of safety relief valves (SRVs)

GE stated in its response of May 20, 1991, that a statement had been added to SSAR Section 14.2.12.1.1 indicating that testing of SRV discharge piping supports and restraints is specifically covered by that testing described in SSAR Section 14.2.12.1.51.

GE incorporated into SSAR Section 14.2.12.1.51 cross-references to SSAR Sections 3.9.2.1 and 5.4.14.4. The references were acceptable subject to incorporation into a future SSAR revision. This was DFSE Confirmatory Item 14.2.12.4-1.

The staff reviewed these cross-references in SSAR Amendment 23, Chapters 3 and 5, respectively and finds them to be acceptable. This resolved DFSE Confirmatory Item 14.2.12.4-1.

- 1.a.(4) Pressure boundary integrity tests

GE stated in its response of May 20, 1991, that integrity tests of the reactor coolant pressure boundary are specified in SSAR Section 5.2.4.6.2.

The staff verified this and the staff also determined that a cross-reference to Section 5.2.4.6.2 was incorporated into Subsection 14.2.12.1.1 in SSAR Amendment 18. The staff evaluated this reference in Chapter 5 of this report, and finds this acceptable.

- 1.c Protection of facility for anticipated transients without a scram (ATWS)

GE stated in its response of May 20, 1991, that ATWS protection functions will be tested as part of the respective systems that perform such functions (i.e., standby liquid control system, rod control and information system

(RCIS), fine motion control rod system, recirculation flow control system). For the purpose of more explicitly demonstrating compliance with RG 1.68, GE revised the appropriate parts of SSAR Section 14.2.12.1 (in SSAR Amendment 18) as shown below to more specifically indicate where ATWS-related testing requirements are being fulfilled, particularly those related to the alternate rod insertion function.

- 14.2.12.1.3(3)(a) - Recirculation Flow Control
- 14.2.12.1.6(3)(b) - CRD (Control Rod Drive) System
- 14.2.12.1.7(3)(b) - RCIS

The staff finds this acceptable.

- 1.h.(4) Demonstration that containment hydrogen monitoring is functional without the operation of the hydrogen recombiner

GE stated in its response of May 20, 1991, that in the ABWR design, containment hydrogen monitoring is accomplished separately from the hydrogen recombiners. Therefore, the specific test described in RG 1.68 is not applicable. Proper functioning of containment hydrogen monitors is verified by the testing described in SSAR Section 14.2.12.1.26. The staff finds this acceptable.

- 1.h.(9) Demonstration that containment recirculation fans can operate in accordance with design requirements at the containment design peak accident pressure

GE stated in its response of May 20, 1991, that the ABWR design does not use containment recirculation fans during normal operation or accident conditions. Therefore, the specific test described in RG 1.68 is not applicable. The staff finds this acceptable.

- 1.i.(1) Containment design over pressure structural tests (and vacuum tests)

GE stated in its response of May 20, 1991, that containment structural integrity testing requirements are specified in SSAR Section 3.8.1.7.1. GE also incorporated into Section 14.2.12.1.40.2 in SSAR Amendment 18 a cross-reference to Section 3.8.1.7.1. This issue was evaluated in SER Chapter 3. The staff finds this acceptable.

- 1.i.(12) Failed fuel detection system

GE stated in its response of May 20, 1991, that the ABWR design failed fuel detection function will be performed by the leak detection and isolation system and the process radiation monitoring system. In particular, gross fuel

failure would be detected first by the main steamline radiation monitors and then by the offgas pre-treatment radiation monitors. In addition, the normal reactor water sampling system will allow for identification of trends indicative of possible fuel failure. Testing of the applicable features of the associated systems, as described in SSAR Sections 14.2.12.1.13 and 14.2.12.1.23, will ensure proper operation of the failed fuel detection function.

GE revised Section 14.2.12.2.1 in SSAR Amendment 18 to include specific reference to proper operation of failed fuel detection functions. The staff finds this acceptable.

- 1.i.(15) Automatic dispatcher control systems

GE stated in its response of May 20, 1991, that automatic load following will be performed by the automatic power regulator for which testing is described in SSAR Section 14.2.12.1.17. This system will have the capability, if enabled, to accept external demand signals (e.g., from the load dispatcher). If the COL applicant decides to seek approval for using this capability, designation of the appropriate testing will be included in that application.

GE revised SSAR Section 14.2.13 to include automatic dispatcher control systems as a responsibility of the COL applicant. This is acceptable. This was also DFSE COL Action Item 14.2.12.4-1. GE incorporated this action item in SSAR Section 14.2.13, and the staff finds this acceptable.

- 1.k.(2) Personnel monitors and radiation survey instruments

GE stated in its response of May 20, 1991, that traditional preoperational testing of personnel monitors and radiation survey instruments is not appropriate in the ABWR design because these instruments will be subject to very specific calibration programs. It is the responsibility of the plant operator to verify and maintain the proper calibration and operation of such devices. Therefore, GE revised the SSAR to indicate that any required testing of personnel monitors and radiation survey instruments would be a COL responsibility. Section 14.2.13 in SSAR Amendment 18 was modified to include this as a COL License Information. This is acceptable. This was also DFSE COL Action Item 14.2.12.4-2. GE incorporated this action item into SSAR Section 14.2.13, and the staff finds this acceptable.

- 1.n.(14)(f) Control room habitability systems. Demonstrate proper operation of smoke and toxic chemical detection systems and ventilation shutdown

Initial Test Program

devices, including leak tightness of ducts and flow rates, proper direction of air flows, and proper control of space temperatures

GE revised the test description in SSAR Section 14.2.12.1.34 (Amendment 18) to indicate that the control room habitability function is to be included in the testing specified for the dedicated HVAC system of the main control room. Additionally, GE added a specific requirement to demonstrate the system capability to detect smoke and/or toxic chemicals and to remove and/or prevent in-leakage of smoke or chemicals. The staff finds this acceptable.

- 2.c Final functional testing of the reactor protection system to demonstrate proper trip points, logic, and operability of scram breakers and valves. Demonstrate the operability of manual scram functions

GE stated in its response of May 20, 1991, that final functional testing will have been completed as part of the preoperational testing described in SSAR Section 14.2.12.1.14. Additionally, these tests are part of the plant TS surveillance program that will be required to be instituted before fuel loading, as specified in SSAR Section 14.2.10.1.

GE revised Section 14.2.12.2.3 in SSAR Amendment 18 to specifically require that the final functional testing required by RG 1.68, Position 2.c, be completed as prerequisites to fuel loading. This is acceptable. This was also tracked as DFSER TS Item 14.2.12.4-1.

The staff found that GE's response to TS Item 14.2.12.4-1 dated April 16, 1993, stated: "Final functional surveillance testing will demonstrate operability of scram breakers and valves prior to fuel loading. Such demonstrations will be assured as part of the TS program (LCO 3.3.1.1 and 3.3.1.2) which must be instituted prior to entry into mode 5." The staff found GE's response acceptable; however, GE had not submitted a response to the ABWR STS LCO 3.3.1.1 and LCO 3.3.1.2, which incorporates surveillance testing of scram breakers and valves to demonstrate their operability before fuel loading. The staff concluded that STS SR 3.3.1.2.1 or STS Bases Section 3.3.1.2 should be revised to include testing the operability of scram breakers, scram solenoid valves, and backup scram solenoid valves. The staff revised the bases portion of the STS to include a description of scram valves and backup scram valves that are tested under STS SR 3.3.1.2. The description does not include testing of scram breakers since these breakers do not exist in the ABWR design. The description for testing the solenoid valves is in the final revision to STS Bases Section B 3.3.1.2 which was issued to GE on August 30, 1993. The

acceptability of the ABWR technical specifications is discussed in Chapter 16 of this report. This resolved DFSER TS Item 14.2.12.4-1.

- 2.d Final reactor coolant system leak rate test to verify that system leak rates are within specified limits

GE stated in its response of May 20, 1991, that final reactor coolant leak rate testing will have been completed as part of the preoperational testing described in SSAR Section 14.2.12.1.1, which references the required reactor coolant leak rate tests specified in SSAR Section 5.2.4.6.1.

GE revised Section 14.2.12.2.3 in SSAR Amendment 18 to specifically require that the leak rate tests required by RG 1.68, Position 2.d, be completed as prerequisites to fuel loading. This is acceptable.

- 4.k Steam driven plant auxiliaries and power conversion equipment

The staff verified that Section 14.2.12.2 was revised in draft to add SSAR Section 14.2.12.2.39 to address testing of steam and power conversion systems. This was acceptable subject to incorporation into a future SSAR revision. This was DFSER Confirmatory Item 14.2.12.4-2.

The staff verified that Section 14.2.12.2.39 was revised to address testing of steam and power conversion systems in SSAR Amendment 23. The staff finds the revisions acceptable. This resolved DFSER Confirmatory Item 14.2.12.4-2.

- 4.l Branch steamline valves and bypass valves used for protective isolation functions at rated temperature and pressure conditions

GE stated in its response of May 20, 1991, that the only branch steamline valves used for ABWR protective isolation functions will be those on the reactor core isolation cooling (RCIC) steamline and the common drainline from the main steamlines.

GE revised the description of the RCIC system testing in Section 14.2.12.2.22 in SSAR Amendment 18 to include specific testing of the RCIC steamline isolation valves and revised Section 14.2.12.2.26 in the same amendment to include specific testing of the main steamline branch drain line isolation valves in addition to the main steam isolation valve (MSIV) testing already specified. The staff finds this acceptable.

• 5.i Plant performance is as expected for rod runback and partial scram

GE stated in its response of May 20, 1991, that the ABWR design has no partial scram function. Rod runback will be accomplished by the select control rod run-in (SCRRI) function. GE revised Section 14.2.12.2.5 in SSAR Amendment 18 to ensure that appropriate testing will be performed to demonstrate proper functioning of SCRRI logic and hardware. GE also revised Section 14.2.12.2.30 to ensure that proper plant response is demonstrated during an event that will result in initiation of SCRRI. The staff finds this acceptable.

• 5.n Reactor coolant system loose parts monitoring system

GE added Section 14.2.12.2.36 to address loose parts monitoring system baseline data collection. This was acceptable subject to incorporation into a future SSAR revision. This was DFSEF Confirmatory Item 14.2.12.4-3.

GE added Section 14.2.12.2.36 in SSAR Amendment 23 to address loose parts monitoring system baseline data collection. The staff finds Section 14.2.12.2.36 acceptable. This resolved DFSEF Confirmatory Item 14.2.12.4-3.

• 5.o Reactor coolant leak detection systems

GE stated in its response of May 20, 1991, that testing of reactor coolant leak detection systems will be completed during the preoperational stage. The staff finds this acceptable.

• 5.q Proper operation of failed fuel detection systems

GE stated in its response of May 20, 1991, that the failed fuel detection function is performed by the process radiation monitoring system, the testing of which is described in SSAR Section 14.2.12.2.1.

GE revised Section 14.2.12.2.1 in SSAR Amendment 18 to require the appropriate demonstration of the related failed fuel detection function. (Also see Item 1.j.(12) above.) The staff finds this acceptable.

• 5.u Branch steamline isolation valve operability and response times

GE revised Sections 14.2.12.2.22 and 14.2.12.2.26 in SSAR Amendment 18 to address branch steamline isolation valve operability and response times. The staff finds this acceptable.

• 5.w Demonstration that concrete temperatures surrounding hot penetrations do not exceed design limits with the minimum design capability of cooling system components available

GE agreed to add SSAR Section 14.2.12.2.37 to address concrete penetration temperature surveys. This was acceptable subject to incorporation into a future SSAR revision. This was DFSEF Confirmatory Item 14.2.12.4-4.

GE added Section 14.2.12.2.37 to address concrete penetration temperature surveys in SSAR Amendment 23. The staff finds Section 14.2.12.2.37 acceptable. This resolved DFSEF Confirmatory Item 14.2.12.4-4.

• 5.x Auxiliary systems required to support operation of engineered safety features

GE stated in its response of May 20, 1991, that the auxiliary systems required to support operation of ESFs include the cooling water and HVAC systems for which testing is described in SSAR Sections 14.2.12.2.23 and 14.2.12.2.24, respectively.

GE revised Sections 14.2.12.2.23 and 14.2.12.2.24 in SSAR Amendment 18 to ensure that the testing performed and results obtained will ultimately demonstrate the adequacy of a particular auxiliary system's performance under limiting accident conditions. The staff finds this acceptable.

• 5.z Demonstration that process and effluent radiation monitoring systems are responding correctly by performing independent laboratory or other analyses

GE stated in its response of May 20, 1991, that this testing is part of that described in SSAR Section 14.2.12.2.1(3).

GE revised Section 14.2.12.2.1(3) in SSAR Amendment 18 to specifically address, RG 1.68, Position 5.z. The staff finds this acceptable.

• 5.c.c Demonstration that gaseous and liquid radioactive waste processing, storage, and release systems operate in accordance with design

GE agreed to add Section 14.2.12.2.38 to address radioactive waste system testing. This was acceptable subject to incorporation into a future SSAR revision. This was DFSEF Confirmatory Item 14.2.12.4-5.

GE added Section 14.2.12.2.38 to address radioactive waste system testing in SSAR Amendment 23. The staff

Initial Test Program

finds Section 14.2.12.2.38 acceptable. This resolved DFSER Confirmatory Item 14.2.12.4-5.

- 5.g.g. Demonstration of design features to prevent or mitigate anticipated transients without scram (ATWS)

GE stated in its response of May 20, 1991, that ATWS design features will consist primarily of dedicated logic and some hardware, which will be thoroughly checked as part of the preoperational test program. Most hardware design features perform ATWS-related functions in their normal mode only when initiated by dedicated ATWS logic. Therefore, the functioning of these features has already been adequately verified during the preoperational testing. Thus, no dedicated testing of ATWS-related features will be planned during the power ascension test phase.

GE incorporated in SSAR Amendment 18 specific ATWS-related testing requirements into individual test abstracts (see Item 1.c above). The staff finds this acceptable.

- 5.h.h. Demonstration that the dynamic response of the plant to load swings for the facility, including step and ramp changes, is in accordance with design

GE stated in its response of May 20, 1991, that this testing is intended to be a part of that described in SSAR Section 14.2.12.2.16.

GE revised Section 14.2.12.2.16 in SSAR Amendment 18 to demonstrate the dynamic response of the plant to load swings for the facility, including step and ramp changes, is in accordance with design. The staff finds this acceptable.

14.2.12.5 Three Mile Island (TMI) Items

Staff Evaluation of DSER Items

Appendix A to SSAR states that testing described in Chapter 14 is consistent with the BWR Owner's Group response to Action Item I.G.1 of NUREG-0737 as documented in a letter dated February 4, 1981, from D.B. Waters to D.G. Eisenhut. The staff determined that the test abstracts in SSAR Section 14.2.12 that describe testing outlined in Appendix E of this letter should be identified or modified accordingly. This was DSER Open Item 124.

GE stated in its response of May 20, 1991, that testing outlined in Appendix E of the referenced document would be specified in the applicable test abstracts.

GE revised test abstracts 14.2.12.1.1(3)(a), 14.2.12.1.9(3)(j), and 14.2.12.1.44(3)(a) to include a

reference to 1A.2.4 of Appendix A to SSAR Chapter 1, and GE revised 1A.2.4 to discuss the requirements of Action Item I.G.1 Appendix E applicable to the ITP. The staff found this item acceptable subject to incorporation into a future SSAR revision. This resolved DSER Open Item 124, and was tracked in the DFSER as DFSER Confirmatory Item 14.2.12.5-1.

GE revised the above test abstracts in SSAR Amendment 23. The staff finds the changes acceptable. This resolved DFSER Confirmatory Item 14.2.12.5-1.

14.2.12.6 Conformance With Other Regulatory Guides

Staff Evaluation of DSER Items

In the DSER, the staff determined that GE should address the concerns of RG 1.56, "Maintenance of Water Purity in Boiling Water Reactor," (Rev. 0), in SSAR Sections 14.2.12.1.19, 14.2.12.1.54, and 14.2.12.2.21. This was DSER Open Item 125.

GE stated in its response of May 20, 1991, that RG 1.56 deals mainly with design related issues, specifically the equipment and instrumentation needed to ensure proper BWR reactor water chemistry. SSAR Sections 14.2.12.1.19, 14.2.12.1.54, and 14.2.12.2.21 describe preoperational and power ascension testing that is adequate to demonstrate that acceptable reactor water chemistry will be maintained by the reactor water clean up system and the condensate filter/demineralizer system. Subsection 14.2.12.1.22 describes the preoperational testing intended to demonstrate the proper functioning of the instrumentation required by RG 1.56. Likewise, SSAR Section 14.2.12.2.1 indicates that a proper reactor water chemistry monitoring program will be in place.

GE agreed to revise SSAR Sections 14.2.12.1.22 and 14.2.12.2.1 to more specifically address functioning of conductivity meters, which are a major focus of RG 1.56. In the DFSER, the staff found this item acceptable subject to incorporation into a future SSAR revision. This resolved DSER Open Item 125, and the issue was tracked in the DFSER as DFSER Confirmatory Item 14.2.12.6-1.

GE made the above revisions to Sections 14.2.12.1.22 and 14.2.12.2.1 in SSAR Amendment 23. The staff finds the revisions acceptable. This resolved DFSER Confirmatory Item 14.2.12.6-1.

In the DSER, the staff determined that Section 14.2.12.2.14, "Feedwater Control," should address the following items in accordance with RG 1.68, "Preoperational and Initial Startup Testing of Feedwater

and Condensate Systems for Boiling Water Reactor Power Plants," Revision 1:

- (1) Modify the test description for demonstration of the required functionality of the feedwater system at low reactor power (less than or equal to 15-percent reactor power) (RG 1.68, Position 1.C.2.a). This was DSER Open Item 126.

GE stated in its response of May 20, 1991, that this testing is already specified in the current description, and supplemented by information to be included in a test matrix.

The staff verified that GE's proposed test matrix, to be included in the SSAR, identified feedwater system performance and feedwater control system adjustment/confirmation tests to be performed at the nuclear heat-up and low-power testing plateaus. In the DFSER, the staff found this acceptable subject to incorporation into a future SSAR revision. This resolved DSER Open Item 126, and this issue became DFSER Confirmatory Item 14.2.12.6-2.

GE added the test matrix in SSAR Amendment 23. The test matrix contained feedwater system performance and feedwater control system adjustment/confirmation tests which are performed at the nuclear heat-up and low-power testing plateaus. The staff finds this to be acceptable. This resolved DFSER Confirmatory Item 14.2.12.6-2.

- (2) Modify or clarify the test acceptance criteria to provide assurance that vibration levels for system components and piping are within predetermined limits (RG 1.68, Position 1.C.2.f); piping movement during heat up and steady state and transient operation are within predetermined limits (RG 1.68, Position 1.C.2.g); and adequate margins exist between system variables and set points of instruments monitoring these variables to prevent spurious actuation or loss of system pumps and motor-operated valves (RG 1.68, Position 1.C.2.h). This was DSER Open Item 127.

GE stated in its response of May 20, 1991, that the testing called for by Positions C.2.f and C.2.g is included in the test abstracts of SSAR Sections 14.2.12.1.51, 14.2.12.1.53(3)(b) and (k), 14.2.12.2.10, 14.2.12.2.11, and 14.2.12.2.18. GE further stated that it revised Section 14.2.12.2.18 in SSAR Amendment 18 to more specifically address Position C.2.h. In the DFSER, the staff verified the above, and found it acceptable. This resolved DSER Open Item 127.

In the DSER, the staff determined that, for SSAR Section 14.2.12.1.27, GE should address the following items in accordance with RG 1.68:

- (1) Determine that the total air demand at normal steady-state conditions, including leakage from the system, is in accordance with design (RG 1.68, Position 3.C.5). This was DSER Open Item 127.

GE revised Section 14.2.12.1.27(3)(f) in SSAR Amendment 18 to include testing of the total air demand at normal steady state-conditions, including leakage from the system, in accordance with the design as specified in Position 3.C.5. The staff finds this acceptable. This resolved DSER Open Item 127.

- (2) Demonstrate that the plant equipment designated by design to be supplied by the instrument and control air system is not being supplied by other compressed air supplies (such as service air (SA)) that may have less restrictive air quality requirements (RG 1.68, Position 3.C.9). This was DSER Open Item 128.

For Position 3.C.9, GE stated in its response of May 20, 1991, that the SA system acts as a backup to instrument air upstream of the instrument air filters. Furthermore, although totally separate (except for the manual backup cross-tie), the design of the two systems is essentially identical. Thus, the air supplied to the inlet of the instrument air filters is of the same quality, whether it is sourced from the instrument or SA system; therefore, the outlet air will be of the same quality. Because the design precludes occurrence of the conditions hypothesized, no specific test demonstration is needed beyond the construction verification and preoperational testing already planned. The staff finds this acceptable. This resolved DSER Open Item 128.

- (3) Demonstrate that functional testing of instrument and control air systems important to safety is performed to ensure that credible failures resulting in an increase in the supply system pressure will not cause loss of operability (RG 1.68, Position 3.C.11). This was DSER Open Item 129.

For Position 3.C.11, GE revised Section 14.2.12.1.27 in SSAR Amendment 18 to include testing that will demonstrate continued operability of supplied loads in response to credible failures that result in an increase in the supply

Initial Test Program

system pressure. The staff finds this item acceptable. This resolved DSER Open Item 129.

In the DSER, the staff determined that GE should address the control room habitability concerns of RG 1.95 in SSAR Section 14.2.12.1.34. This was DSER Open Item 130.

GE revised Section 14.2.12.1.34 in SSAR Amendment 18 to indicate that the control room habitability function will be included in the testing specified for the dedicated HVAC system of the main control room. GE also added a specific requirement to demonstrate the system capability to detect smoke and/or toxic chemicals and to remove and/or prevent in-leakage of such. (Also see RG 1.68, Position 1.n.(14)(f), in Section 14.2.12.4 of this report.) The staff finds this acceptable. This resolved DSER Open Item 130.

In the DSER the staff determined that GE should address, in SSAR Section 14.2.12.1.34, HVAC Systems Preoperational Tests, or other appropriate preoperational tests, the concerns for in-place testing of HEPA and charcoal filters of Position C.5, In-Place Testing Criteria, of RG 1.140. This was DSER Open Item 131.

GE revised Section 14.2.12.1.34 in SSAR Amendment 18 to include testing requirements specified by RC 1.140 and by the industry standards referenced therein. The staff finds this acceptable.

GE later determined that Position C.5, of RG 1.140 only discussed testing criteria for HEPA and carbon absorber filters in HVAC systems. The HVAC systems described in Section 14.2.12.1.34 do not contain HEPA or carbon absorber filters, therefore, Position C.5 of RG 1.140 does not apply. RG 1.52 addresses testing of HEPA and carbon absorber filters in ESF systems. GE added RG 1.52 and removed RG 1.140 in Sections 14.2.12.1.34 and 14.2.12.1.36 of a markup dated June 7, 1993, to address testing of ESF HVAC systems (i.e., control room habitability HVAC system and the standby gas treatment system (SGTS)) that require in place testing of HEPA and carbon adsorber filters. This information was incorporated into Amendment 30 of the SSAR. The staff finds this acceptable. This resolved DSER Open Item 131.

Finally, the staff determined in the DSER that GE should address the following items in accordance with RG 1.139 in SSAR Section 14.2.12.1.8.

- (1) Residual heat removal (RHR) system isolation (RG 1.139, Position C.2). This was DSER Open Item 132.

GE stated in its response of May 20, 1991, that the applicable demonstrations were intended to be a part of the testing described in SSAR Section 14.2.12.1.8(3)(i).

Further, GE agreed to revise SSAR Section 14.2.1.2.1.8 (1) to specifically address testing of features designed to ensure isolation of low-pressure portions of the RHR system from the reactor coolant system (RCS) at high pressure. In the DFSER, the staff found this item acceptable subject to incorporation into a future SSAR revision. This was DFSER Confirmatory Item 14.2.12.6-3.

GE revised SSAR Section 14.2.12.1.8 (1) in SSAR Amendment 23 to include testing of features designed to ensure isolation of the low pressure portions of the RHR system at high pressures. The staff finds this acceptable. This resolved DFSER Confirmatory Item 14.2.12.6-3.

- (2) RHR system pressure relief (RG 1, Position 139.C.3). This was DSER Open Item 133.

GE stated that the design of the RHR system will include the relief capacity required by Position C.3 in accordance with the applicable ASME Code. GE indicated that no specific additional preoperational test was needed because verification of the proper setting of relief valves was a vendor bench test required per the same ASME Code.

Section 14.2.12.1.8 was revised to allow for verification of proper set points of system relief valves per ASME code requirements (including those intended to meet the requirements of RG 1.139 using the results of vendor tests and the appropriate documentation of such). In the DFSER, the staff found this item was acceptable subject to incorporation into a future SSAR revision. This was DFSER Confirmatory Item 14.2.12.6-4.

GE added system pressure relief valve verification tests per ASME code and RG 1.139 requirements into Section 14.2.12.1.8 (3) in SSAR Amendment 23. The staff finds the above tests acceptable. This resolved DFSER Confirmatory Item 14.2.12.6-4.

14.2.12.7 Certified Design Material

The staff reviewed the GE certified design material and determined that the ITPs were not included in the Tier 1 or ITAAC material. The staff expected that this Tier 1

material would be at a high level limited to a commitment to an ITP and a description of the program and the major program documents (i.e., a site-specific startup administrative manual, test specifications, and test procedures). This was DFSE Open Item 14.2.12.7-1.

GE provided the Tier 1 ITP information. An evaluation of the ITP information is contained in Section 14.3.3.5 of this report. The staff has found the description acceptable. This resolved DFSE Open Item 14.1.12.7-1.

14.2.13 COL License Information - Initial Test Program

The preceding discussion of preoperational and startup tests were limited to those systems and components within the scope of, or directly related to, the ABWR standard plant. Other testing, with respect to utility specific aspects of the plant will be necessary to satisfy certain ABWR requirements. Testing of such systems and components should be adequate to demonstrate conformance to such requirements as defined throughout the specific chapters of the SSAR. Below are systems that may require such testing

- (1) electrical switchyard and equipment
- (2) the site security plan
- (3) personnel monitors and radiation survey instruments
- (4) the automatic dispatcher control system (if applicable)

Also to be supplied by the applicant referencing the ABWR design is the startup administrative manual described in Section 14.2.4, which will describe, among other things, what specific permissions are required for the approval of test results and the permission to proceed to the next testing plateau.

The staff received a GE response dated May 7, 1993, containing a new section (i.e., Section 14.2.13, COL License Information). Section 14.2.13 lists all the COL Action Items (i.e., DFSE COL Action Items 14.2.3-1 through 14.2.12.4-2) that were identified in the DFSE. The staff found that the new section lists all COL license information that will be provided by the COL applicant. The COL Action Items list was subsequently incorporated into the SSAR. The staff finds the COL license information provided in Section 14.2.13 acceptable.

Conclusion

The staff performed the review of the plant ITPs in accordance with Section 14.2, of NUREG-0800, the SRP, and RG 1.68 (Rev. 2), "Initial Test Programs for Water Cooled Nuclear Power Plants." The staff reviewed

thirteen areas relating to initial plant test programs, described in Chapter 14 of the SSAR, and submitted by GE as part of its design certification application. These areas of review included:

- summary of the test program and objectives
- organization and staffing
- test procedures
- conduct of test program
- review, evaluation, and approval of test results
- test records
- conformance of test program with regulatory guides
- utilization of reactor operating and testing experience in the development of test program
- trial use of plant operating and emergency procedures
- initial fuel loading and initial criticality
- test program schedule
- individual test descriptions
- combined license (COL) information - ITP

The acceptance criteria used by the staff for this review were contained in SRP Section 14.2, Subsection II.

Based on the review of the GE ABWR ITP description in SSAR Chapter 14 and the responses to the RAIs and the DSE open items, DFSE open items and confirmatory items, the staff concluded that the ITP was generally comprehensive and covered all areas of staff concern. The staff also conducted an in-depth review of system-specific testing requirements within each test abstract. The staff concluded that GE provided a sufficient level of detail to adequately describe system-specific test prerequisites and acceptance criteria for design certification. The staff also reviewed cross-references to acceptance criteria information in other parts of the SSAR and found them acceptable. An additional review of COL action items and interfacing systems will be performed at the time a COL applicant referencing the ABWR design submits an application for a license under 10 CFR Part 52.

Initial Test Program

14.3 Certified Design Material

Introduction

The objective of this section of the report is to provide the basis for the staff's approval of the certified design material (CDM) for the ABWR design. This section of the report is based on the staff's review of the GE document titled "ABWR Certified Design Material" and standard safety analysis report (SSAR) Section 14.3, "Certified Design Material." The requirement to submit this information as part of an application for design certification is contained in 10 CFR 52.47.

The GE document titled "ABWR Certified Design Material" contains the information that will be certified by the rule that approves the ABWR design. The CDM consists of an introductory section, design descriptions and corresponding inspections, tests, analyses, and acceptance criteria (ITAAC) for the systems of the design, design material applicable to multiple systems of the design, interface requirements, and site parameters for the ABWR design. This information is also referred to as Tier 1 information. The ABWR SSAR Section 14.3, "Certified Design Material," provides the bases and methods that were used to develop the information for each of these CDM items for the ABWR. This report documents the staff's review of the ABWR CDM, as supported by the design information contained in the ABWR SSAR.

The information in the CDM and SSAR Section 14.3 is derived from the detailed information contained in the SSAR. Further, the purpose of the ITAAC, which are part of the CDM, is to verify that a facility that references the design certification has been built and will operate in accordance with the design certification and the applicable regulations. Consequently, there is no design information presented in the CDM or SSAR Section 14.3 that is not also contained in the various sections of the SSAR. Therefore, the staff did not base its safety evaluations for the design on the information in the CDM.

Basis for Approval of the CDM

The ABWR was the lead design for the initial development of CDM for design certification. Although the staff was able to make its safety determinations for the design based on its review of the information in the SSAR, this was the first time design material had been developed for certification. Therefore, there was no precedent for GE to follow to develop information for the CDM. Furthermore, the staff had no regulatory guidance on which to base its review. Recognizing this, the staff sought and received Commission guidance on several key issues associated with the design certification reviews in the staff requirements

memorandum (SRM) dated February 15, 1991, relating to SECY-90-377, "Requirements for Design Certification Under 10 CFR Part 52." These issues included the development of regulatory guidance, the role of ITAAC, the level of design detail needed for design certification, issue finality, the two-tiered approach to the design certification rule structure, and flexibility in design change process. In its review, the staff evaluated the CDM in the context of Part 52 requirements and the Commission guidance contained in that SRM.

As the lead design, GE submitted the CDM to the staff for review in stages, so that lessons learned at each stage could be incorporated into later submittals. The staff reviewed the material in an iterative manner and provided comments on the CDM to GE at each stage. The staff informed the Commission of the development of the CDM in multiple SECY papers issued in 1991 and 1992 (SECYs 91-178, 91-210, 92-053, 92-196, 92-214, 92-287 and 287A, 92-299, and 92-327). These papers are listed as references in Appendix B of this report.

In SSAR Section 14.3, GE provided the process it used in the development of the CDM, based on the design presented in the appropriate sections of the SSAR. GE provided CDM based on the structures and systems of the ABWR design rather than based on the format of the ABWR SSAR. In addition, GE adopted a graded approach to the level of design detail for the information in the CDM, based on the safety significance of particular structures, systems, and components (SSCs). GE applied various selection criteria to the information in the SSAR to determine the level of design information for a given structure or system in the CDM. The results of this process were illustrated with cross-references from the SSAR information to the CDM for important parameters that were selected for treatment in the CDM. Although many issues and analyses could have been cross-referenced, the listings in Section 14.3 were developed only for selected integrated plant safety analyses for the ABWR design. GE provided additional cross-references of key insights and assumptions from probabilistic risk analyses (PRA) and analyses for severe accidents which are contained in SSAR Chapter 19. GE provided more detailed cross-references to the CDM for these analyses in a letter dated March 31, 1994.

The staff also utilized a graded approach to the level of detail in its review of the CDM based on the safety significance of the SSCs. Thus, consistent with the guidance of Part 52 and the SRM related to SECY-90-377, the staff recognized that although many aspects of the design were important to safety, the level of design detail in the CDM and verification of the key design features and

performance characteristics should be commensurate with the significance of the safety functions to be performed.

The intent of the CDM is to ensure that the key characteristics and performance requirements of safety-significant SSCs are implemented in an as-built facility referencing the certified design. Although all these aspects of the design are described in the CDM, not all can be verified by the ITAAC because Part 52 requires that the ITAAC be satisfied prior to fuel loading. The initial test program (ITP) serves to verify the remaining aspects of the design after fuel load, but prior to operation. Examples of these requirements are the post-fuel load startup and power ascension test program verification of fuel, control rod, and core characteristics, as well as system and integrated plant operating characteristics. The treatment of these issues will be similar to their treatment at facilities licensed under 10 CFR Part 50, in that verification of the satisfactory completion of these requirements will be a condition of the license.

The staff recognized that other programs also ensure the continued safe operation of a facility after fuel load. For example, the continued operability of a facility after the ITAAC are satisfied is ensured through the technical specifications (TS), as well as various programs such as the maintenance program, quality assurance program, and the in-service inspection and testing program. The operator ensures the facility is operated as designed, through the use of appropriate plant operating and emergency procedures. Additionally, a utility referencing the design is required by 10 CFR Part 50, Appendix B, to have a quality assurance (QA) program that ensures these SSCs are appropriately designed, procured, and perform satisfactorily in service.

The above considerations provided an overall framework for GE's development of the CDM and the staff's review. The staff utilized multiple sources of information to determine the safety significance of SSCs in the CDM. These sources included the SRP, applicable rules and regulations, general design criteria (GDC), regulatory guides (RGs), unresolved safety issues (USIs) and generic safety issues (GSIs), NRC generic correspondence, operating experience, NRC inspection programs, facility testing programs, PRA, and insights from ABWR safety and severe accident analyses. For selected portions of the review, the staff also utilized the regulatory guidance from the Commission related to SECY-90-016, "Evolutionary Light Water Reactor Certification Issues and Their Relationship to Current Regulatory Requirements," as modified by the Commission guidance related to SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor Designs."

Nonetheless, because this was the first time CDM had been developed for a design, because there was no precedent for GE to follow and no detailed review guidance for the staff to base its review on, and because of the iterative nature of the ITAAC development process, considerable judgement was inherent in the approval of the final material for the CDM.

Background

Part 52 of Title 10 of the Code of Federal Regulations was issued April 18, 1989. The concept in Part 52 of certifying a design and specifying the required ITAAC in a rule prior to construction of a facility had not been attempted before. Consequently, GE and the Nuclear Management and Resources Council (NUMARC) held extensive discussions with the staff in 1990 and 1991 on the CDM, especially the form and content of the ITAAC. After several senior management meetings in mid-1991, the GE ABWR emerged as the lead plant in the development of the first CDM for a standardized design.

The development of the CDM was an iterative process. As the lead design, GE submitted the CDM to the staff for review in stages, so that lessons learned at each stage could be incorporated into later submittals. The staff reviewed the material as it was submitted and provided comments on the CDM to GE at each stage. Senior management meetings between GE and the staff were held at intervals of approximately six to eight weeks to resolve difficult policy and technical issues associated with the development and review of the material.

GE submitted the first stage of the CDM, consisting of a set of nine "pilot" ITAAC, in a submittal dated September 20, 1991. The staff provided comments on this submittal in a letter of October 23, 1991, and after several meetings with the staff to discuss the comments, general agreement was reached on the pilot ITAAC in January 1992.

GE submitted the second stage of the CDM, consisting of approximately one-half the ITAAC, in a submittal titled "Tier 1 Design Certification Material for the GE ABWR Design - Stage 2 Submittal," dated March 31, 1992. The staff provided preliminary comments to GE on the Stage 2 submittal in a letter dated May 7, 1992, as well as during several senior management meetings.

GE submitted the third stage of the CDM, consisting of the remainder of the Tier 1 material, in its submittal titled "Tier 1 Design Certification Material for the GE ABWR," dated June 1, 1992, as supplemented by a letter dated June 17, 1992. The staff provided detailed comments on the CDM in a letter to GE dated August 12, 1992, including the comments of an NRC review group

Initial Test Program

comprised of senior managers from several headquarters and regional offices of the NRC.

In September 1992, a group comprised of representatives from various vendors, utilities, and industry groups provided industry comments on the CDM to GE. GE revised the CDM to incorporate the comments provided by both the NRC and this review group in late 1992. Subsequently, a team of NRC reviewers met with GE in January, February, and March 1993, to review the revised ITAAC.

GE submitted the next stage of the CDM, consisting of a revised, complete CDM that incorporated all previous lessons learned and review comments, in incremental submittals in April, May, and June of 1993. In May 1993, the staff formed several task groups to perform a multidisciplinary review of the CDM. The task groups provided extensive comments on the submittals to GE in a letter dated July 9, 1993. The task groups met with GE on July 27 through 29, 1993, to discuss the comments, and the disposition of all of the comments were documented in a meeting summary dated August 10, 1993.

GE submitted the next stage of the CDM, consisting of a revised, complete CDM that incorporated the comments from the staff's task group review, in a submittal titled "ABWR Certified Design Material," dated August 31, 1993. GE supplemented this material with CDM Revision 1 in September 1993, and with CDM Revision 2 in December 1993. Changes to the detailed CDM supporting information in the SSAR were contained in various amendments to the SSAR, up to and including Amendment 33.

The staff had identified multiple inconsistencies between the SSAR, the Tier 1 design descriptions, and the related ITAAC during the reviews. Consequently, the staff formed an independent review group to ensure consistency of the CDM and the SSAR in November 1993. The independent review group completed its review in February 1994, and its comments were provided to GE after review by the staff. GE provided satisfactory resolutions to all the comments in CDM Revision 3 and SSAR Amendment 34. In addition to resolving the comments of the independent review group, CDM Revision 3 also included resolutions to comments by the ACRS and the staff. Subsequent revisions to the CDM prior to the start of the administrative review of the ABWR design were minor in nature.

Format of the CDM

GE developed a format for the design certification information to meet the requirements of 10 CFR 52.47,

including bounding parameters for siting of the standard design, design descriptions and corresponding ITAAC, and interface requirements for the design. The CDM is the portion of the design information that is certified by the rule certifying the design, and will be incorporated into the rule as the Tier 1 part of the design control document (DCD). The DCD is the master document that contains the information that must be conformed with by an applicant who references the rule. The format for this material is listed below, and is discussed in more detail in the following sections of this report.

- (1) Introduction - Definitions of terms used in the CDM, and a listing of general provisions that are applicable to all CDM entries.
- (2) System Design Descriptions and ITAAC - System design descriptions and ITAAC are provided for: (a) structures and systems that are fully within the scope of the ABWR design certification, and (b) the in-scope portions of those systems that are only partially within the scope of the ABWR design certification. The system design descriptions are accompanied by the appropriate ITAAC.
- (3) Additional Certified Design Material - Design descriptions and their related ITAAC for design and construction activities that are applicable to more than one system of the design. This additional material was provided because in selected areas of the design, GE did not provide sufficient design detail in the SSAR. GE did not provide complete design information in these areas because they were either areas of rapidly changing technology where GE believed it was unwise to prematurely freeze the design, or because the information was dependent on as-built or as-procured information. For these areas, GE provided the design related processes in the CDM and in the SSAR, with appropriate codes and standards, that a COL applicant or licensee would follow to complete the design.
- (4) Interface Requirements - Requirements that must be met by the site-specific portions of a facility that are not within the scope of the certified design. This section also identifies the scope of the design to be certified. Interface requirements are defined for: (a) systems that are entirely outside the scope of the design, and (b) the out-of-scope portions of those systems that are only partially within the scope of the ABWR design.
- (5) Site Parameters - Bounding parameters of the design to be used in the selection of a suitable site

for a facility referencing the ABWR certified design. The design was evaluated in terms of these parameters. A suitable site must be demonstrated to be within the bounding parameters and characteristics, and a facility must be constructed at the site in accordance with their use in the approved design. If a site cannot meet them, an exemption must be requested and the facility must be reevaluated in terms of these parameters for the actual selected site.

14.3.1 Introduction to CDM

14.3.1.1 Definitions

This section of the CDM provides terms used in the CDM that could be subject to various interpretations. The intent of the terms used in the CDM was to be consistent and as closely aligned as possible with the terminology in the SSAR, in common industry use, industry codes and standards, and NRC rules, regulations, and guidance. Thus, should questions on terminology arise, these references would aid in understanding the intent of the information in the CDM.

14.3.1.2 General Provisions

This section of the CDM provides general provisions that are applicable to the design descriptions, figures, and the ITAAC.

14.3.1.2.1 Verifications for Basic Configuration for Structures and Systems

This section of the CDM includes provisions related to the verification of the ITAAC for basic configuration for systems and structures of the design. This ITAAC is contained in the buildings and many of the systems described in Section 2 of the CDM. The verification consists of an inspection of the system functional arrangement in its final as-built condition at the plant site, and includes the elements of the design descriptions and the system figures in the CDM. This functional arrangement inspection verifies, using as-built system drawings, design documentation, and in-situ plant walkdowns, that the as-built facility is in conformance with the certified design and applicable regulations.

Several other aspects of the design were considered to have significance to the performance of safety functions of SSCs of a facility. The basis for selecting these aspects included its importance to safety as well as its past experience with construction and operating problems. Thus, specific inspections for these aspects are part of the basic configuration ITAAC for systems and structures. The

other inspections to be conducted to satisfy this ITAAC include, and are limited to, verification of the following:

- (1) Verifications of the quality of pressure boundary welds for ASME Code Class 1, 2, and 3 components and systems described in the design descriptions and figures. Detailed supporting information for verification of welding requirements in accordance with ASME Code requirements is contained in SSAR Chapter 3.
- (2) Verifications of the dynamic qualification (e.g., seismic, LOCA, and safety relief valve discharge loads) of seismic Category I mechanical and electrical equipment (including connected instrumentation and controls) described in the design descriptions and figures. Detailed supporting information for dynamic qualification requirements, including qualification records, is contained in SSAR Chapter 3.
- (3) Verifications of the environmental qualification of Class 1E electrical equipment described in the design descriptions and figures. Detailed supporting information for environmental qualification requirements is contained in SSAR Chapter 3.
- (4) Verifications of the design qualification of motor-operated valves (MOVs) described in the design descriptions and figures. Detailed supporting information for design qualification of MOVs is contained in SSAR Chapter 3.

14.3.1.2.2 Treatment of Individual Items

A licensee is not prohibited from utilizing an item not described in the CDM. However, the as-built facility must be consistent with the rule approving the design, including both tiers of information. The change processes for the certified design are described in the design certification rule for the ABWR.

The term "operate" as utilized in the CDM is intended to refer to the actuation and running of equipment. This is not meant to include the term "operable" in the context of the ongoing reliability and availability of equipment. In developing the ITAAC, the staff recognized that other programs ensure the continued safe operation of a facility after fuel load. For example, the continued operability of a facility after the ITAAC are satisfied is ensured through the Technical Specifications, Startup and Power Ascension Test Programs, as well as various programs such as the maintenance program, quality assurance program, and the in-service inspection and in-service testing program. Also,

Initial Test Program

the operator ensures the facility is operated as designed, through the use of appropriate plant operating and emergency procedures.

The term "exists," when used in the Acceptance Criteria, means that the item is present and meets the design description. Detailed supporting information on what must be present to conclude that an item "exists" and meets the design description is contained in the appropriate sections of the SSAR.

14.3.1.2.3 Implementation of ITAAC

GE developed a three column format for the ITAAC. The design commitments in the first column are derived from the design information in the design descriptions. The inspections, tests, and analyses in the middle column provide the intended means of verifying the design commitment. The acceptance criteria in the third column provide the criteria used to determine whether the design commitment is met.

The licensee is required by 10 CFR Part 52 to perform the required inspections, tests, and analyses for the design, and certify to the NRC that the acceptance criteria have been met. A licensee may utilize the efforts of subordinate vendors, contractors, or consultants. However, the licensee referencing the certified design retains responsibility for ensuring that the ITAAC are met. Additionally, the ITAAC can be satisfied using other programs, such as the pre-operational testing portion of the ITP required by CDM Section 3.5, or the QA program required by 10 CFR Part 50, Appendix B.

The ITAAC may be satisfied at any time prior to fuel load, including prior to issuance of a combined license. However, the primary intent of the ITAAC is to verify that the as-built plant on the final site has been constructed and will perform in accordance with the design certification and applicable regulations.

14.3.1.2.4 Discussion of Matters Related to Operations

Descriptions in the CDM may refer to matters of operation, such as normal valve or breaker alignment during normal operational modes. These descriptions are not intended to require operators to take any particular action. The operational matters referred to in the CDM are governed by existing programs to ensure the ongoing safe operation of a facility, such as plant operating and emergency procedures.

14.3.1.2.5 Interpretation of Figures

The design descriptions include the figures in the CDM, where the figures are provided. They are intended to depict the functional arrangement of the significant SSCs of the ABWR design. An as-built facility referencing the certified design must be consistent with the performance characteristics and functions described in the design descriptions and figures. Any changes to the detailed information in the SSAR must be in accordance with the "50.59-like" change process in the design certification rule for the ABWR, which allows the COL applicant or licensee to make design changes, provided the changes do not impact the information in the CDM.

14.3.1.2.6 Rated Reactor Core Thermal Power

The rated reactor core thermal power for the ABWR is 3926 MW(th).

14.3.1.3 Conclusions

As discussed above, the staff reviewed the definitions and general provisions that are contained in CDM Section 1.0, and the supporting material contained in SSAR Section 14.3.1, in accordance with the requirements in Part 52 and the guidance in SRMs related to design certification applications provided by the Commission. Based on this, the staff concludes that the definitions and general provisions in the CDM are appropriate to support the design descriptions and ITAAC, and are acceptable.

14.3.2 Certified Design Material for Structures and Systems

GE developed design descriptions and ITAAC for the structures and systems of the ABWR design, and these are contained in CDM Section 2.0. General provisions that apply to most of the structures and systems are contained in CDM Section 1.2. Additional CDM material for design issues that apply to many of these structures and systems are contained in CDM Section 3.0. Interface requirements for these systems are provided in the system design descriptions for the in-scope portions of the systems. The interface requirements for the out-of-scope portions of the systems of the design are contained in CDM Section 4.0. GE provided an entry in the CDM for every system of the design to define the full scope of the design.

14.3.2.1 Design Descriptions

The design descriptions address the most safety-significant aspects of each of the systems of the design, and were derived from the detailed design information contained in the SSAR. The design descriptions include the figures associated with the systems. GE's selection criteria and methodology for the system design descriptions are specified in SSAR Section 14.3.2.1. In its review of the material, the staff followed the general guidance for the reviews specified in the SRM related to SECY-90-377, as discussed previously in the introduction to Section 14.3 of this report.

The Tier 1 design descriptions will serve as commitments for the lifetime of a facility. Once completion of ITAAC and the supporting design information demonstrate that the facility has been properly constructed, it then becomes the function of existing programs such as the technical specifications, the in-service inspection and in-service testing program, the quality assurance program, and the maintenance program, to demonstrate that the facility continues to operate in accordance with the certified design and the license. Nevertheless, the Tier 1 design descriptions will remain in effect throughout the plant life to assure that the plant does not deviate from the certified design. In general, a COL applicant or licensee may change the information in the SSAR in accordance with the "50.59-like" change process described in the rule certifying the design, provided that the change does not impact the information in the design descriptions.

GE provided the selection criteria for information in the design descriptions in Section 14.3.2.1. Essentially, GE put the top-level design features and performance standards that were most significant to safety in the design descriptions. The criteria GE utilized in determining the safety significance of SSCs in the design descriptions included the NRC's regulations, whether or not the information pertained to safety-related SSCs, the importance in the SRP, the relative importance based on PRA or severe accident analysis, operating experience, or the technical specifications. GE also included other SSCs based on their importance to safety. Non-safety aspects of SSCs were generally not discussed in the design descriptions. Thus, although a Tier 1 entry was provided for every system that is either fully or partially within the scope of the ABWR design certification, the amount of information provided in CDM Section 2.0 for a given system, if any, was based on the safety significance of the system.

GE provided additional certified design material applicable to the systems of the design in CDM Section 3.0. The design descriptions in CDM Section 3.0 describe the scope

and applicability of the additional certified design material to the systems of the design in CDM Section 2.0. Amplifying information on CDM Section 3.0 is provided in SSAR Section 14.3.3, and the staff's review of CDM Section 3.0 is contained in Section 14.3.3 of this report.

The CDM utilizes a system-based structure which is different than the structure of the SSAR. Consequently, developing the CDM design description entries for any one system was based on a review of the multiple SSAR chapters having technical information related to that system. GE illustrated this approach in SSAR Section 14.3.2.1, showing how the many design aspects of any single system in the CDM were derived from multiple chapters of the SSAR.

The staff was particularly interested in ensuring that the assumptions and insights from key safety and integrated plant safety analyses in the SSAR, where plant performance was dependent on contributions from multiple systems of the design, were adequately considered in the CDM. Addressing these assumptions and insights in the CDM ensures that the integrity of the fundamental analyses for the design are preserved in an as-built facility referencing the certified design. These analyses included flooding analyses, overpressure protection, containment analyses, core cooling analyses, fire protection, transient analyses, and radiological analyses.

GE provided information regarding these analyses in the SSAR, and documented the important design information and parameters from the various chapters of the SSAR that are addressed in the CDM in Tables 14.3-1 through 14.3-10. GE provided more detailed cross-references to the CDM for these analyses in a letter dated March 31, 1994. GE also provided cross-references in SSAR Chapter 19 showing how key insights and assumptions from PRA and severe accident analyses are addressed in the CDM. A COL applicant or licensee proposing to change design information in the SSAR that pertained to these analyses via the "50.59-like" change process can use these cross-references when considering whether the proposed change impacts the treatment of these parameters in the CDM.

14.3.2.2 ITAAC

The purpose of the ITAAC is to verify that an as-built facility conforms to the approved plant design and applicable regulations. When coupled in a COL with the ITAAC for site-specific portions of the design, they constitute the verification activities for a facility that must be successfully met prior to fuel load. If the licensee demonstrates that the ITAAC are met and the staff agrees that they are successfully met, then the licensee will be permitted to load fuel.

Initial Test Program

The scope of the ITAAC is consistent with the SSCs that are in the design descriptions. In general, each system has one or more ITAAC that verify the information in the design descriptions. This is not true in all cases. Reasons for not requiring an ITAAC verification for a Tier 1 design commitment include: (1) the information is only included for context, (2) fulfillment of other ITAAC are sufficient to show verification of the design commitment, (3) a single ITAAC can verify more than one design commitment, or (4) verification of the item can only occur after fuel loading. For the last item, the staff reviewed the power ascension testing program described in SSAR Chapter 14 to ensure that all important design features and commitments that could not be verified prior to fuel load were addressed where appropriate.

The staff reviewed the system ITAAC to ensure that the verifications were consistent with the safety significance of the key design characteristics and performance requirements of the SSC verified by that ITAAC. The certified design descriptions for an SSC contain the significant functions and bases for that SSC. Therefore, the ITAAC have been reviewed to ensure they are necessary and sufficient to provide the NRC with reasonable assurance that the facility should be authorized to load fuel. As a result, the ITAAC verify the significant design features from the design descriptions and the applicable requirements that are necessary and sufficient to authorize fuel loading and subsequent operation.

The staff and industry reached agreement on a three-column format for ITAAC, as discussed below.

Column 1 - Design Commitment

This column contains the text for the specific design commitment that was extracted from the design descriptions discussed above. Any differences in text were minimized, unless intentional. Differences in text were generally intended to better conform the commitments in the design description with the ITAAC format.

Column 2 - Inspections, Tests, and Analyses

This column contains the specific method to be used by the licensee to demonstrate that the design commitment in Column 1 has been met. The method is either by inspection, test, or analysis or some combination of inspections, tests, or analyses.

The SSAR contains detailed supporting information for the CDM about various inspections, tests, and analyses that can, and should be, used to verify the Tier 1 design information and satisfy the acceptance criteria. If questions on interpretation should arise, the material in the

SSAR provides the background material and context for the CDM. The SSAR contains information reviewed by the staff which was the basis for the staff's safety determination for the design. Therefore, the information in the SSAR provides an acceptable means of satisfying an ITAAC.

Inspections are defined in CDM Section 1.1, and include visual and physical observations, walkdowns or record reviews. The inspections required for the "Basic Configuration Walkdown" ITAAC invoke the general provisions contained in CDM Section 1.2 for as-built structures and systems.

Tests are defined in CDM Section 1.1, and mean the actuation, operation, or establishment of specified conditions to evaluate the performance or integrity of the as-built SSCs. This includes functional and hydrostatic tests for the systems. The term "as-built" is intended to mean testing in the final as-installed condition at a facility. The term "type tests" is used in this column to mean manufacturer's tests or other tests that are not necessarily intended to be in the final as-installed condition. The results of pre-operational tests can be used to satisfy an ITAAC. In its review, the staff did not rely on the pre-operational tests described in SSAR Section 14.2 or RG 1.68 to substitute for ITAAC. Where testing is specified, appropriate conditions for the test should be established in accordance with the ITP described in CDM Section 3.5, SSAR Section 14.2 and RG 1.68. Conversion of the test results from the test conditions to the design conditions may be required to satisfy the ITAAC.

During its review, the staff emphasized in-situ testing, where possible, of the as-built facility as the preferred means to satisfy the ITAAC. Also, the staff recognized that the results and documentation from facility programs such as the quality assurance program or the ITP may be used to satisfy an ITAAC.

Analyses are defined in CDM Section 1.1, and may refer to detailed supporting information in the SSAR, simple calculations, or comparisons with operating experience or design of similar SSCs. For example, detailed analysis methods of seismic and environmental qualification supporting CDM Section 1.2 are contained in SSAR Chapter 3, and detailed piping design information supporting CDM Section 3.3, are also contained in SSAR Chapter 3.

Column 3 - Acceptance Criteria

This column contains the specific acceptance criteria for the inspections, tests, or analyses described in Column 2

which, if met, demonstrate that the design commitment in Column 1 has been met.

In general, the acceptance criteria were developed to be objective and unambiguous. In some cases, the acceptance criteria were more general because the detailed supporting information in the SSAR did not lend itself to concise verification. For example, the acceptance criteria for the design integrity of piping and structures is that a report "exists" that concludes the design commitments are met. In these cases, the SSAR provides the detailed supporting information on multiple interdependent parameters that must be provided in order to demonstrate that a satisfactory report exists.

Numeric performance values for SSCs were specified as ITAAC acceptance criteria when the design commitment so lent itself, or when failure to meet the stated acceptance criterion would clearly indicate a failure to properly implement the design. The staff did not require that numeric performance values be specified in the design description unless there was a specific reason to include them (e.g., important to be maintained for the life of the facility).

ITAAC Implementation

The ITAAC may be satisfied at any time prior to fuel load, including prior to issuance of a combined license. However, the primary intent of the ITAAC is to verify that the as-built facility on the final site has been constructed and will operate according to the design certification and applicable regulations.

The implementation of a construction verification program, including ITAAC, and other licensee programs, is the responsibility of the licensee. The successful completion of the ITAAC in the combined license will constitute the basis for the NRC's determination to allow fuel loading for the facility.

The licensee will periodically certify to the NRC that the inspections, tests, and analyses have been performed, and that the acceptance criteria have been met. These notifications should document the basis for the successful completion of the ITAAC. In accordance with 10 CFR 52.99, the staff will assure that the required inspections, tests, and analyses have been performed and that the prescribed acceptance criteria have been met. At appropriate intervals, the NRC will publish in the Federal

Register, notices of the successful completion of the inspections, tests, and analyses.

14.3.2.3 Staff Review Approach

GE developed the CDM based on the systems of the design rather than on the format of the SSAR and SRP. In order to ensure that the safety-significant design information in the SSAR was adequately reflected in the CDM, the staff adopted a multidisciplinary review approach, rather than the more traditional review approach based on the individual chapters of the SSAR. The staff formed several task groups comprised of various disciplines to ensure that the CDM would provide reasonable assurance that a facility would be built and operated in accordance with the design certification and applicable regulations.

The task groups were composed of various representatives from the technical branches of the staff, depending on the primary area of review by the task group. The task groups were formed based on the following discipline areas: plant systems, reactor systems, electrical, human factors, radiation protection, structural, and instrumentation and controls.

The task groups had primary review responsibilities for systems that were predominantly in their discipline area, and secondary review responsibilities for systems in other discipline areas where appropriate based on the safety significance of the issues. Thus, the groups had overlapping system review responsibilities. All information in the CDM was reviewed by one or more task groups. For example, the reactor core isolation cooling system was reviewed primarily by the Reactor Systems Task Group, and that task group received technical input and comments from the Instrumentation and Controls Task Group. Specialists were designated to provide input to the task groups for selected design issues. Examples of these issues included severe accident issues, testing issues and the ITP, treatment of alarms, displays, and controls, insights from PRA, and functionality of MOVs.

The task groups with primary review responsibility for systems maintained overall responsibility for the reviews of those systems. Overall continuity and consistency of the reviews was maintained through frequent meetings with all task groups and with the projects branch of the staff. Significant policy and technical issues, or issues of concern to multiple task groups, were identified for discussion at periodic senior management meetings between the staff and GE.

Initial Test Program

The staff developed preliminary draft guidance for use in the reviews of the CDM, and incorporated lessons learned during the course of the reviews into the draft guidance. The draft guidance contained checklists for use in the reviews. The applicability of the issues identified in the checklists to the systems was based on the safety significance of the specific SSCs. The draft guidance also contained standard ITAAC entries that were used to verify selected issues in the appropriate systems of the design. Examples of these standard ITAAC entries are those for the basic configuration of systems, verification of control room and remote shutdown features, and electrical independence. The issues in the checklist and the use of the standard ITAAC entries are discussed in the following sections of this report.

The task groups utilized multiple sources of information to determine the safety-significance of SSCs in the CDM. These sources included the SRP, applicable rules and regulations, GDCs, RGs, USIs and GSIs, NRC generic correspondence, operating experience, NRC inspection programs, facility testing programs, PRA, and insights from ABWR safety and severe accident analyses.

For selected portions of the review, the staff also utilized the regulatory guidance from the Commission related to SECY-90-016, "Evolutionary Light Water Reactor Certification Issues and Their Relationship to Current Regulatory Requirements," as modified by the Commission guidance related to SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor Designs." These issues included staff positions that deviated from or were not embodied in current regulations applicable to the ABWR design. The staff's positions and design-specific requirements are addressed where appropriate in this section of this report, and in the ABWR design certification rule as "applicable regulations."

The staff determined that selected material in the SSAR that supports the CDM, if considered for a change by an applicant or licensee that references the certified ABWR design, would constitute an unreviewed safety question, and therefore, would require NRC review and approval prior to implementation of the change. The material supporting the CDM is discussed where appropriate in this section, in the applicable chapters of this report, and in the ABWR design certification rule.

The task groups utilized a graded approach to the level of detail in its review of the CDM based on the safety-significance of the SSCs. Thus, consistent with the guidance of Part 52 and the SRM related to SECY-90-377, the staff recognized that although many aspects of the design were important to safety, the level of design detail

in the CDM and verification of the key features and performance characteristics should be commensurate with the significance of the safety functions to be performed. In addition, the SSAR was reviewed to ensure that the information was consistent with the design description and that the information supporting the Tier 1 material was comprehensive and technically adequate. Thus, the individual task groups reviewed the CDM based on the safety significance of the material, as discussed in the following paragraphs.

14.3.2.3.1 Plant Systems Task Group Review

The Plant Systems Task Group had primary review responsibility for most of the fluid systems in CDM Section 2.0 that were not part of the core reactor systems. The scope of the plant systems review included new and spent fuel handling systems, power generation systems, air systems, cooling water systems, radioactive waste systems and heating, ventilation and air conditioning systems. The group also reviewed selected interface requirements within those systems. The group reviewed issues which affect multiple SSCs such as equipment qualification and protection from fires, floods and tornado missiles, and had secondary review responsibilities for most of the fluid systems and the structures of the design.

The task group primarily utilized the SRP in its review of the CDM to determine the safety significance of SSCs. Other sources included applicable rules and regulations, GDCs, RGs, USIs and GSIs, NRC generic correspondence, PRA, insights from ABWR safety and severe accident analyses, and operating experience. The task group also used the draft review guidance for the design control document as an aid in its review of the systems. For selected portions of the review, the staff also utilized the regulatory guidance from the Commission related to SECY-90-016, "Evolutionary Light Water Reactor Certification Issues and Their Relationship to Current Regulatory Requirements," as modified by the Commission guidance related to SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor Designs." The task group reviewed the Tier 1 submittals (including the design description, figures, and ITAAC) of the design using the guidelines provided in the draft review guidance for the CDM as an aid for establishing consistency and completeness.

The task group reviewed the CDM for treatment of design information proportional to the safety significance of the SSC for that system. Many items were judged to be important to safety, and were thus included in the CDM. The following issues were identified to ensure comprehensive and consistent treatment in the CDM based on the safety significance of the system being reviewed:

- (1) System purpose and functions
- (2) Location of system
- (3) Key design features of the system
- (4) Seismic and ASME code classifications
- (5) System operation in various modes
- (6) Controls, alarms, and displays
- (7) Logic
- (8) Interlocks
- (9) Class 1E electrical power sources and divisions
- (10) Equipment to be qualified for harsh environments
- (11) Interface requirements
- (12) Numeric performance values
- (13) Accuracy and quality of figures

Additionally, standard ITAAC entries were utilized to verify selected issues, where appropriate. Examples of these included basic configuration, physical separation, and divisional power supplies. In particular, the general provision for environmental qualification aspects of SSCs invoked by the basic configuration ITAAC was reviewed to ensure appropriate treatment in the CDM.

Environmental qualification (EQ) of safe-shutdown equipment is verified as part of the basic configuration ITAAC for safety-related systems. EQ treatment in the ITAAC is discussed in the General Provisions section of the CDM. Verification includes type tests or a combination of type tests and analyses of Class 1E electrical equipment identified in the Design Description or accompanying figures to show that the equipment can withstand the conditions associated with a design basis accident without loss of safety function for the time that the function is needed.

The task group reviewed integrated plant safety analyses such as fires, floods and missile protection to ensure they were adequately addressed in the CDM. The insights from these analyses that were addressed in the CDM are contained in SSAR Section 14.3. The issues of floods, fires, missiles, pipe failures, and environmental protection are verified by the ITAAC on a system-specific basis, rather than generically. Divisional separation (both physical and electrical) is the primary means of ensuring protection of safety-related equipment from these events. Verification of divisional separation is performed as part of both individual system ITAACs and building ITAACs. Physical and electrical separation is verified in each safety-related system ITAAC and divisional barriers are verified in the reactor and control building ITAACs.

The design features in the CDM were selected to ensure that the integrity of the analyses would be preserved in an as-built facility. For example, 3-hour fire boundaries and divisional separation were shown in the building figures. Also, flooding features such as structure elevations were

specified in the site parameters, flood doors were shown on the building figures, and elevations were shown on the buildings to verify that the approximate physical location of components and relative elevations of buildings minimized the effects of flooding. As-built reconciliation reports for fires and floods to ensure consistency with the SSAR analyses are required by the fire protection system ITAAC and selected building ITAAC, respectively.

Other specific issues that were addressed include heat removal capabilities for design-basis accidents and tornado and missile protection. Heat removal capabilities were verified through heat removal requirements for core cooling system heat exchangers and interface requirements for site-specific systems. Tornado and missile protection was provided by inlet and outlet dampers in ventilation systems, and through the structural design of buildings.

The staff was evaluating ACRS comments regarding the need for verification of fires and flooding analyses in the ITAAC for buildings when the advance SER was issued. This was Open Item F14.3.2.3.1-1. GE provided satisfactory resolutions to the ACRS comments in CDM Revision 3 and SSAR Amendment 34. The CDM contained provisions for reconciliation analyses to be conducted for the ABWR design for fires and floods, to ensure that the as-built facility is consistent with the assumptions and analyses for these issues in the design certification. The staff finds this acceptable. This resolved Open Item F14.3.2.3.1-1.

The task group received inputs from other task groups such as the structural, electrical and I&C task groups. The task group also reviewed the ITAAC for consistency with the initial test program described in SSAR Chapter 14. In addition, specialists provided key insights and assumptions from PRA and severe accident analyses, as well as inputs for issues such as treatment of alarms, displays and controls, and functionality of MOVs. A cross-reference from the SSAR to the CDM providing these key insights and assumptions is contained in SSAR Section 19.8.

The issue of containment isolation is addressed by a combination of the system ITAACs and the Leak Detection and Isolation System ITAAC. The containment isolation valves are shown on the system figures. The verification of the design qualification of the motor operated containment isolation valves will be verified by the basic configuration check in the system ITAAC as discussed in the general provisions discussion. In addition, in-situ tests are required for containment isolation MOV and check valves in each system ITAAC. The Leak Detection and Isolation System ITAAC verifies that the containment isolation valves close on receipt of an isolation signal. Actual closure of the containment isolation valves is

Initial Test Program

checked using the manual isolation switches in the main control room (MCR). A separate ITAAC entry verifies that a containment isolation signal is generated for each the process variables that will cause a containment isolation. This precludes multiple cycling of the containment isolation valves during the testing.

The staff decided during the review of the ITAAC that the MCR ITAAC would verify only the minimum inventory as derived from the Emergency Procedure Guidelines, the requirements of RG 1.97, and probabilistic risk assessment insights. Other controls, indications and alarms are identified in the system ITAAC and verified to exist in the MCR. The ability of these controls, indications, and alarms to function will be checked during operation of the system for the functional tests required by the system ITAAC. The operation of the system during the completion of the functional tests required in the system ITAAC will be conducted from the MCR. Therefore, it was decided that the verification that the system can be operated from the MCR need not be a separate ITAAC. The staff also decided that since the operation of the equipment from the control room demonstrates the control function, continuity checks between the remote shutdown panel (RSP) and the equipment demonstrates that the control signal will be received by the component and provides adequate assurance that the equipment can be operated from the RSP. Additionally, the Initial Test Program will adequately cover the technical verifications of the ability to operate plant equipment from the RSP.

14.3.2.3.2 Reactor Systems Task Group Review

The reactor systems task group had primary review responsibility for the reactor systems and core cooling systems in CDM Section 2.0. The group had secondary review responsibilities for those systems that could affect the operation of the reactor and core cooling systems.

The task group primarily utilized the SRP in its review of the CDM to determine the safety significance of SSCs. Other sources included applicable rules and regulations, GDCs, RGs, USIs and GSIs, NRC generic correspondence, PRA, insights from ABWR safety and severe accident analyses, and operating experience. The task group also used guidelines provided in the draft review guidance for the design control document as an aid for establishing consistency and completeness in its review of the systems. For selected portions of the review, the staff also utilized the regulatory guidance from the Commission related to SECY-90-016, "Evolutionary Light Water Reactor Certification Issues and Their Relationship to Current Regulatory Requirements," as modified by the Commission guidance related to SECY-93-087, "Policy,

Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor Designs."

The task group reviewed the CDM systems in a similar manner as the plant systems task group because the reactor and core cooling systems were primarily fluid systems. Thus, the group examined the systems for comprehensive and consistent treatment of the issues listed in Section 14.3.2.3.1 of this report, based on the safety significance of the respective systems being reviewed. The task group found that many of the systems in this area of review were classified as safety related, and thus many of the characteristics and features of these systems were judged to have safety significance. This is reflected in a higher level of detail in the CDM for these systems.

The task group reviewed the CDM to verify that plant safety analyses, such as for core cooling, transients, overpressure protection, and anticipated transient without scram (ATWS), were adequately addressed. The task group used the tables contained in SSAR Section 14.3 to assess that the important input parameters used in the transient and accident analyses were verified by the ITAAC. The task group also interacted with specialists in PRA and severe accident analyses to ensure important insights and design features from these analyses were incorporated into the CDM. For the severe accident analyses in particular, the basis for the staff's review was the Commission guidance related to SECYs 90-016 and 93-087. For both PRA and severe accident analyses, although large uncertainties and unknowns may have been associated with the event phenomena, design features important for severe accident prevention and mitigation resulting from these analyses were selected for treatment in the CDM. The supporting information regarding the detailed design and analyses remained in the SSAR. For many of the design features, it was impractical to test their functionality. Consequently, the existence of the feature on a figure, subject to a basic configuration walkdown, was considered sufficient CDM treatment.

The staff determined that the detailed supporting information in the SSAR for the nuclear fuel, fuel channel, and control rod CDM, if considered for a change by a COL applicant or licensee that references the certified ABWR design, would constitute an unreviewed safety question. Thus, the staff has concluded that the fuel cycle and control rod design criteria in SSAR Sections 4B and 4C, the first cycle fuel, control rod and core design and the methods used to analyze these components may not be changed without prior NRC review and approval. The specific fuel, control rod, and core designs presented in SSAR Chapter 4 will constitute, based on staff review and approval, an approved design that may be used for the COL first cycle core loading, without further NRC staff

review. If any other core design is requested for the first cycle, the COL applicant or licensee will be required to submit for staff review that specific fuel, control rod, and core design analyses as described in SSAR Chapters 6 and 15.

No ITAAC are required for the CDM information in these areas because of the requirement for prior NRC review and approval of any proposed changes to the approved design. Post fuel load testing programs (e.g., startup testing and power ascension testing) verify that the actual core performs in accordance with the analyzed core design.

Examples of the issues that the task group examined for treatment in Tier 1 included net positive suction head for key pumps (standard ITAAC entry specified in the applicable systems), and intersystem LOCA (the design pressure of the piping of the systems that interface with the reactor coolant pressure boundary is specified in the design descriptions of the applicable systems). The task group also reviewed the ITAAC for consistency with the initial test program described in SSAR Chapter 14.

14.3.2.3.3 Electrical Task Group Review

The electrical task group had primary review responsibility for the station electrical systems in CDM Section 2.12. The scope of the ABWR electrical design includes the entire Class 1E portion of the electrical system as well as a major portion of the non-Class 1E electrical system. It also includes portions of the plant lighting system. The group also reviewed selected interface requirements. The group had secondary review responsibilities for some systems using Class 1E power.

In establishing the top level requirements for the electrical design, the staff used the Code of Federal Regulations including the GDC of Appendix A and Parts 50.49, "Environmental Qualification," and 50.63, "Station Blackout," as its main bases. In addition, IEEE nuclear standards were used, as appropriate, to further establish top level requirements. The staff also considered significant lessons learned from operating experience problems and insights gained from the PRA for the ABWR.

GDC 17, in part, requires that an onsite and an offsite electric power system be provided to permit functioning of structures, systems and components important to safety. It further requires that the onsite electric power system have independence and redundancy and the electric power supplied by the offsite system be supplied by two physically independent circuits.

10 CFR 50.49 requires that certain electrical equipment be qualified for accident (referred to as harsh) environments.

10 CFR 50.63 requires that a nuclear power plant be able to withstand and recover from a station blackout event.

IEEE 308 "IEEE Standard Criteria for Class 1E power Systems for Nuclear Power Generating Stations," in conjunction with other related IEEE standards, establish specific design criteria for nuclear power plant electrical systems and equipment.

The staff's review of the ABWR standard plant was conducted to ensure, in part, that the certified design contains top level design, fabrication, testing, and performance requirements for SSCs important to safety. Design descriptions and ITAAC were established to verify that these top level (Tier 1) requirements (or design commitments) are met when the plant is built.

Class 1E Electrical Systems

The ABWR Class 1E electrical systems include: (1) the Class 1E electrical power distribution system, (2) the emergency diesel generators, (3) the Class 1E direct current power supply, and (4) the Class 1E vital ac and Class 1E instrument and control power supplies. Using the above regulations, IEEE standards, operating experience, and PRA as its bases, GE established top-level design commitments for the Class 1E electrical systems of the ABWR to be included in the design descriptions and verified by ITAAC.

The top-level design commitments for the Class 1E electrical systems include design aspects related to:

- (1) Equipment qualification for seismic and harsh environment
- (2) Redundancy and independence
- (3) Capacity and capability
- (4) Electrical protection features
- (5) Displays/controls/alarms

Equipment Qualification

To ensure that the seismic design requirements of GDC 2 and the environmental qualification requirements of 10 CFR 50.49 have been adequately addressed, "basis configuration" ITAAC were established for applicable systems to verify these design aspects of electrical equipment important to safety.

Initial Test Program

The design description identifies that Class 1E equipment is seismic Category 1 and equipment located in a harsh environment is qualified. A "basic configuration" ITAAC was developed to include these areas.

Redundancy and Independence

To ensure that the Class 1E electric systems meet the single failure requirements of GDC 17 (and other GDC), ITAAC were established to verify the redundancy and independence of the Class 1E portion of the electrical design.

For the electrical systems, ITAAC verified the Class 1E divisional assignments and independence of electric power by both inspections and tests. The independence is established by both electrical isolation and physical separation. Identification of the Class 1E divisional equipment is included to aid in demonstrating the separation. (The detailed requirements are specified in the SSAR. For example, separation distances and identification are outlined in the SSAR.) These attributes are verified all the way to the electrically powered loads by a combination of the electrical system ITAAC and the ITAAC of the individual fluid, I&C, and HVAC systems which also cover the electrical independence and divisional power supply requirements.

Capacity and Capability

To ensure that the electrical systems have the capacity and capability to supply the safety-related electrical loads, ITAAC were established to verify the adequate sizing of the electrical system equipment and its ability to respond (e.g., automatically in the times needed to support the accident analyses) to postulated events. This includes the Class 1E portion and the non-Class 1E portion to the extent that it is involved in supporting the Class 1E system.

ITAAC are included to analyze the as-built electrical system and installed equipment (diesel generators, transformers, switchgear, batteries, etc.) to verify its ability to power the loads. In addition, the ITAAC also includes tests to demonstrate the operation of the equipment.

To ensure that the Class 1E portions of the electrical power system have the capability to respond to postulated events including LOCA, loss of normal preferred power, and degraded voltage conditions, ITAAC were established to verify the initiation of the Class 1E equipment necessary to mitigate the event.

ITAAC are included to analyze the as-built electrical power system for its response to a LOCA, loss of voltage, combinations of LOCA and loss of voltage, and degraded voltage. In addition, tests are included to demonstrate the actuation of the electrical equipment in response to postulated events.

Electrical Protection Features

To ensure that the electrical power system is protected against potential electrical faults, ITAAC were established to verify the adequacy of the electrical circuit protection included in the design. Operating experience and NRC Electrical Distribution System Functional Inspections (EDSFI)s have indicated some problems with the short circuit rating of some electrical equipment and breaker and protective device coordination.

ITAAC are included to analyze the as-built electrical system equipment for its ability to withstand and clear electrical faults. ITAAC are also included to analyze the protection feature coordination to verify its ability to limit the loss of equipment due to postulated faults.

Displays, Alarms and Controls

To help ensure that the electrical power system is available when required, ITAAC are included to verify the existence of monitoring and controls for the electrical equipment. The minimum set of displays, alarms, and controls is based on the emergency procedure guidelines. In some cases, additional displays, alarms, and controls are specified based on special considerations in the design and/or operating experience.

ITAAC are included to inspect for the ability to retrieve the information (displays and alarms), and to control the electrical power system in the main control room and/or at locations provided for remote shutdown.

Other Electrical Equipment Important to Safety

In addition to the Class 1E systems addressed above, other aspects of the electrical design were deemed to be important to safety and the top-level design commitments were included in the CDM.

Offsite Power

To ensure that the requirements of GDC 17 for the adequacy and independence of the preferred offsite power sources within the ABWR scope were met, ITAAC were developed to verify the capacity and capability of the offsite sources to feed the Class 1E divisions, and the independence of those sources.

ITAAC are included to inspect the direct connection of the offsite sources to the Class 1E divisions and to inspect for the independence/separation of the offsite sources. Lightning protection and grounding features are inspected as part of the configuration ITAAC.

In addition, the design description includes "interface" requirements for the portions of the offsite power outside of the ABWR scope, however no ITAAC are included for the interfaces. The interfaces define the requirements that the offsite portion of the design (that is out-of-scope) must meet to support and not degrade the in-scope design.

Containment Electrical Penetrations

To ensure the containment electrical penetrations (both those containing Class 1E circuits and those containing Non Class 1E circuits) do not fail due to electrical faults and potentially breach the containment, ITAAC were developed to verify that all electrical containment penetrations are protected against postulated currents greater than their continuous current rating.

ITAAC are included to inspect and analyze the electrical penetrations and their protection.

Combustion Turbine Generator

To ensure the availability of the combustion turbine generator (CTG) as an alternate AC source for station blackout events, ITAAC were developed to verify its inclusion in the design and its independence from other AC sources. In addition, the PRA has indicated that the CTG is relatively important from a risk perspective.

ITAAC are included to inspect and test the CTG and its auxiliaries.

Lighting

To ensure that portions of the plant lighting remain available during power failures, ITAAC were developed to verify the continuity of power sources for the lighting systems.

ITAAC are included to inspect and test the lighting and its power sources.

Electrical Power For Non-Safety Plant Systems

To ensure that electrical power is provided to support the non-safety plant systems, Design Descriptions cover portions of the non-Class 1E electrical systems. A basic configuration ITAAC verifies the functional arrangement and the Tier 1 design commitments for these areas.

14.3.2.3.4 Human Factors Task Group Review

The Human Factors task group had primary review responsibility for the main control panel, remote shutdown panel, and local control panels, described in CDM Section 2.0. The group also reviewed CDM Section 3.1, "Human Factors Engineering." CDM Section 3.1 is discussed further in Section 14.3.3.1 of this report. The task group provided input to other task groups on the minimum inventory of alarms, controls, and indications for the control room and the RSS.

The basis for the task group's review in this area was a human factors engineering (HFE) program review model (PRM) developed by the staff. The staff's certification review in the control room design area was based on a design and implementation process plan. The staff informed the Commission of the development of the DAC in this area in SECY-92-299, "Development of Design Acceptance Criteria (DAC) for the Advanced Boiling Water Reactor (ABWR) in the Areas of Instrumentation and Controls (I&C) and Control Room Design," dated August 27, 1992. In addition, the task group utilized the SRP in its review of the CDM. Other sources included applicable rules and regulations, RGs, USIs and GSIs, and operating experience.

The staff developed the HFE PRM, contained in Appendix J of this report, to serve as a technical basis for the review of the design process and DAC proposed by GE for certification of the ABWR control room and remote shutdown station design. The HFE PRM is (1) based upon currently accepted HFE practices, (2) well-defined, and (3) validated through experience with the development of complex, high-reliability systems in other industrial and military applications. The review model identifies the important HFE elements in a system development, design, and evaluation process that are necessary and sufficient requisites to successful integration of human factors in complex systems. The review model also identifies aspects of each HFE element that are key to a safety review, and describes acceptance criteria by which the HFE elements can be evaluated. The HFE PRM has eight program elements, each of which contain both general and more specific acceptance criteria.

Part 52 requires applicants for design certification to meet the TMI requirements in 10 CFR 50.34(f)(2)(iii) for providing a control room design that reflects state-of-the-art human factors principles. GE did not develop a final control room and RSS design before design certification because this is an area of rapidly changing technology. Instead, GE provided the processes and acceptance criteria in CDM Section 3.1 and the detailed supporting information in SSAR Chapter 18 by which the details of

Initial Test Program

the design in this area would be developed, designed, and evaluated. In lieu of having a completed control room design for review, the staff concluded that it could make its safety determination if GE submitted for certification an acceptable process for the design of the control room. In addition, GE must have submitted a description of a minimum inventory of displays, controls, and alarms necessary to accomplish the emergency procedure guidelines (EPGs) and critical operator actions identified through GE's PRA analysis.

The processes and design acceptance criteria in CDM Section 3.1, "Human Factors Engineering," apply to the human factors design of the control room and the RSS systems of the ABWR design. The detailed supporting information for the human factors aspects of the ABWR control room and RSS design are provided in SSAR Chapter 18, "Human Factors," and together with the associated DAC in CDM Section 3.1, are evaluated in Chapter 18 of this report. GE provided amplifying information regarding the processes and CDM selection criteria in this area in SSAR Section 14.3.3.1. The implementation of the process and the design is the responsibility of the COL applicant or licensee.

The staff requested that the minimum inventory of displays, controls, and alarms be developed through a task analysis of the operator actions necessary to carry out the EPGs and PRA critical actions. The staff's evaluation of the resulting minimum inventory encompassed a multi-disciplinary effort consisting of human factors, I&C, PRA, and plant, reactor, and electrical system engineering. The criteria used to determine acceptability of the inventory included assuring that: (1) the scope of these items in the EPGs and PRA effort were adequately considered, (2) the task analysis was detailed and comprehensive, (3) RG 1.97, category I variables for accident monitoring were included, and (4) important system displays and controls described in the Tier 1 system design descriptions necessary for transient mitigation were included.

The minimum inventory list for the control room was included in the CDM Section 2.7.1, "Main Control Room Panels." The controls and indicators required on systems to remotely shutdown the reactor are contained in CDM Section 2.2.6, "Remote Shutdown System." The items required for operation of the RSS are shown with an "R" on the figures for the individual systems. Detailed supporting information is contained in Chapter 7 of the SSAR. The individual systems that contained the sensors for the displays, controls, and alarms were reviewed to ensure that standard ITAAC entries were used to verify their function. The design processes and acceptance criteria specified in the DAC for I&C equipment contained in CDM Section 3.4, particularly the verification and

validation aspects of the I&C DAC, will verify proper operation of the I&C aspects of the equipment. Similarly, the design processes and DAC for HFE contained in CDM Section 3.1, particularly the verification and validation aspects of the HFE DAC, will verify proper design of the equipment for human factors aspects.

14.3.2.3.5 Radiation Protection Task Group Review

The Radiation Protection Task Group had primary review responsibility for the area radiation monitoring system, containment atmospheric monitoring system, and emergency response facilities in CDM Section 2.0; the additional material in CDM Section 3.2, "Radiation Protection," applicable to multiple systems of the design; and selected site parameters. CDM Section 3.2 is discussed further in Section 14.3.3.2 of this report. The group had secondary review responsibility for all other ITAACs which addressed the plant radiation protection design or systems relied upon in the design-basis accidents (DBAs) dose assessment. These ITAACs included buildings, ventilation and filtration systems, primary containment, drywell bypass, post-accident sampling system, and site parameters (atmospheric dispersion).

The task group primarily utilized the SRP in its review of the CDM to determine the safety significance of SSCs. Other sources included applicable rules and regulations, GDCs, RGs, USIs and GSIs, NRC generic correspondence, and operating experience. The task group also used the draft review guidance for the design control document as an aid for consistency in its review of the systems.

The task group relied heavily on the material in CDM Section 3.2 during its review of the design. The design processes and acceptance criteria in this section were developed because GE did not provide sufficient information to stipulate the source terms needed to verify the design of the shielding, ventilation, and airborne radioactivity monitoring systems. Therefore, GE extracted the most important acceptance criteria for these design features from Chapter 12 of the SRP and put them into the DAC in CDM Section 3.2. A COL applicant or licensee must meet these criteria in the design of the plant, and the staff can audit the facility's design documentation to ensure that the criteria are met. The DAC are general criteria which apply to the design of shielding and ventilation systems throughout the plant. Therefore, there are no references to the DAC in the ITAACs for the buildings and systems.

The group reviewed the ITAAC for the area radiation monitoring system to ensure that the system provides information on radiation dose rates in the plant during normal operation and accidents and provides alarms to

warn plant personnel of changes in those dose rates. The group reviewed the ITAAC for the containment atmospheric monitoring system to ensure that the system provides information on radiation dose rates and gas concentrations during accidents and provides alarms to warn plant personnel of high levels of these parameters. The group reviewed the ITAAC for emergency response facilities to ensure that adequate facilities are provided for the technical support center (TSC) and operational support center including space, data retrieval and communications equipment, and a ventilation system to provide radiation protection.

The group reviewed several ITAAC for which the group had secondary review responsibility. The review of these ITAAC was focused on verifying design features and assumptions upon which the radiological dose consequence assessment of the design basis accidents (DBAs) in this SER is based. The following discussion provides examples of some of the important design features and assumptions that are addressed in the CDM. The maximum MSIV closure time and maximum MSIV leakage rates will be verified by the ITAAC for the nuclear boiler system. The maximum primary containment leakage rate will be verified by the ITAAC for the primary containment system. The minimum radioiodine removal efficiency of the charcoal adsorbers in the standby gas treatment system (SGTS) filter trains and the maximum time for the SGTS to draw a specified negative pressure in the secondary containment will be verified by the ITAAC for the SGTS. The minimum radioiodine removal efficiency of the charcoal adsorbers in the control room and TSC ventilation system filter trains will be verified by the ITAAC for the HVAC systems. Capability of the main steam system to maintain structural integrity in an safe-shutdown earthquake (SSE) will be verified by the ITAAC for the turbine main steam system. Capability of the off-gas system to withstand an internal hydrogen explosion will be verified by the ITAAC for the off-gas system. In addition, the meteorological dispersion values assumed in the accident analyses were identified as bounding parameters for a site in CDM Table 5.0, ABWR Site Parameters. Also, the radiological analysis table in SSAR Section 14.3 was used to ensure that GE had addressed in the CDM the most important, though not necessarily all, of the key parameters in the accident dose analyses.

14.3.2.3.6 Structural Task Group Review

The Structural Task Group had primary review responsibility for building structures, chemical engineering systems, site parameters, piping DAC, reactor pressure vessel (RPV) system, and the legend for figures. The piping DAC contained in CDM Section 3.3 is discussed in Section 14.3.3.3 of this report. The task group had

secondary review responsibilities for other systems as they related to MOVs, check valves, hydrostatic tests, and seismic and safety classification of systems, and for other structural aspects of systems. The task group was composed of reviewers with experience in structural, mechanical, materials, and chemical engineering. In addition, the task group was augmented by a specialist in MOVs, check valves, and pumps, a specialist in seismic and safety classification, and a specialist in chemical engineering.

The task group primarily utilized rules and regulations to review the top level commitments in the CDM. Other sources included RGs, SRP guidelines, and PRA insights from ABWR safety and severe accident analyses and operating experience. For selected portions of the review, the staff also adhered to policy discussions by the Commission in the SRM related to SECY-90-016, "Evolutionary Light Water Reactor Certification Issues and Their Relationship to Current Regulatory Requirements," as modified by the Commission guidance in the SRM related to SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor Designs." In addition, the task group reviewed the Tier 1 submittals (including the design description, figures, and ITAAC) of the design using the guidelines provided in the draft review guidance for the CDM as an aid for establishing consistency and completeness.

The task group reviewed the design description for those assigned systems to ensure that the certified design was consistent with the NRC regulations and policy decisions as discussed in SECY-93-087. The task group reviewed the Tier 1 material to assess whether a conclusion could be reached that the ITAAC were necessary and sufficient to provide reasonable assurance that the facility has been constructed and will be operated in conformity with the license, the provisions of the Atomic Energy Act, and the Commission's rules and regulations.

The task group reviewed the design descriptions, figures, ITAAC, and the SSAR for consistency. The task group reviewed the CDM for all the systems to ensure consistency with the seismic and safety classification described in Section 3.2 of the SSAR. The task group ensured that the seismic classification of the system as described in the design description and the ASME Code Class boundaries of the system as depicted on the figures was consistent with SSAR Section 3.2.

The task group reviewed the ITAAC for consistency with the preoperational tests specified in Chapter 14 of the SSAR. The task group reviewed the tests identified in the ITAAC tables to determine whether those tests have been

Initial Test Program

appropriately included in SSAR Chapter 14 and also whether the preoperational tests have been adequately incorporated into ITAAC. The task group also reviewed tests in Chapter 14 of the SSAR or in the ITAAC that would require an analysis to convert preoperational test conditions to accident conditions to ensure that the methodology for performing the analysis was specified adequately. In addition, the task group reviewed all systems ITAAC to ensure that selected issues were adequately and consistently treated in the CDM through the use of the standard ITAAC entries for basic configuration, hydrostatic test, MOVs, and check valves.

The task group used the following general approach in reviewing the design descriptions, figures, and ITAAC and for establishing what information should reside in each tier. The certified design (design description) should contain top level design, fabrication, testing, and performance requirements for SSCs important to safety. ITAAC are established, in part, to verify that these top-level (Tier 1) design, fabrication, testing and performance requirements are met when the plant is built.

Although the establishment of what specific information was to be included in the design description was essentially a matter of judgement, the draft review guidance provided some guidance for consistency in certain areas regarding what information should be in which tier as well as whether an inspection, test or analysis was required to be performed. The draft review guidance also provided a basis for the staff's judgement in selecting which tier the information should reside and why an ITAAC was deemed necessary. These areas include component welding, equipment seismic qualification, pumps, valves, and piping systems. The basis for selecting these areas included its importance to safety as well as its past experience with construction and operating problems.

Design descriptions and ITAAC were developed and grouped by systems and building structures. These Tier 1 requirements for systems and building structures are typically verified by inspections, tests, and analyses specified in the system ITAAC. For example, system-specific performance tests are typically conducted to demonstrate that the system can perform its intended function. For building structures, the structural capability is typically verified by performing an analysis to reconcile the as-built data with the structural design bases for each safety-related building.

For components, the verification of design, fabrication, testing, and performance requirements are partially addressed in conjunction with the specific system ITAAC. For example, a test is typically performed to verify the ability of a motor-operated valve to close under design-

basis fluid conditions. However, performance tests are not practical for verifying certain component design requirements such as its seismic design or safety classification. Therefore, ITAAC have been developed to verify certain areas where performance tests are not practical. These areas include seismic design qualification and fabrication (i.e., welding) of components. The ITAAC for seismic design qualification and fabrication of components are established on a generic basis in the general provisions for verifying the basic configurations of systems rather than on an individual component basis.

The Tier 1 treatment of the design qualification and fabrication of components was reviewed to ensure that the issues were verified by ITAAC as discussed below:

(1) Fabrication of Components

A basic configuration check (system) is required in each individual system ITAAC. The configuration check includes an inspection of the welding quality for all ASME Code Class 1, 2, and 3 piping systems. A hydrotest is also required in each system ITAAC for ASME Code Class 1, 2, and 3 piping systems to verify that, in the process of fabricating the overall piping system, the welding and bolting requirements for ensuring the pressure integrity have been met. The methods to be used by the COL applicant or licensee to verify the acceptability of the welds are discussed in the SSAR in the sections applicable to the specific component or structure.

(2) Design Qualification of Components

- (a) **Safety Classification** - The safety classification of SSCs are described in each system's design description. The functional drawings identify the boundaries of the ASME Code classification that are applicable to the safety class. The piping DAC includes a verification of the design report to ensure that the appropriate code design requirements for the system's safety class have been implemented.
- (b) **Mechanical and Electrical Equipment (including I&C)** - A basic configuration check (system) is required in each individual system ITAAC. The configuration check includes an inspection of the as-built equipment (including anchorages) and a review of the qualification records to verify that the equipment in its as-built condition is seismically qualified. The material in SSAR

Section 3.10.1 provides detailed supporting information for the CDM regarding the methods to be used by the COL applicant or licensee for the dynamic qualification of equipment. This material, if considered for a change by an applicant or licensee that references the certified ABWR design, would constitute an unreviewed safety question, and therefore, would require NRC review and approval prior to implementation of the change. This material supporting the CDM is discussed further in Section 3.10 of this report.

- (c) **Valves** - The verification of the design qualification of valves is performed in conjunction with the basic configuration check for mechanical equipment as discussed above. Specifically, for MOVs, a special inspection is required as a part of the basic configuration check to verify the records of vendor tests that demonstrate the ability of MOVs to function under design conditions. In addition, in-situ tests are required for MOVs and check valves in each system ITAAC. These tests will be performed during the initial test program. The material in SSAR Section 3.9.6.2.2 provides detailed supporting information for the CDM regarding the methods to be used by the COL applicant or licensee for the design, qualification, and testing of MOVs to demonstrate their design basis capability. This material, if considered for a change by an applicant or licensee that references the certified ABWR design, would constitute an unreviewed safety question, and therefore, would require NRC review and approval prior to implementation of the change. This material supporting the CDM is discussed further in Section 3.9.6 of this report.
- (d) **Piping** - The verification of the overall piping design including the effects of high-energy line breaks and the application of leak-before-break (as applicable) is performed in conjunction with the piping DAC. The as-built piping system is required to be reconciled with the design commitments. The material in SSAR Section 3.12 provides detailed supporting information for the CDM regarding the analysis methods and design criteria to be used by the COL applicant or licensee to complete the piping design. This material,

if considered for a change by an applicant or licensee that references the certified ABWR design, would constitute an unreviewed safety question, and therefore, would require NRC review and approval prior to implementation of the change. This material supporting the CDM is discussed further in Section 3.12 of this report.

Review of the ABWR Structural Design Integrity

The scope of structural design covers the major structural systems in the ABWR plant including the RPV, ASME Code Class 1, 2, and 3 piping systems, and major building structures (primary containment, reactor building, control building, turbine building, service building, and radwaste building). The RPV, piping systems, and primary containment are included because they provide the defense-in-depth principle for nuclear plants. The major building structures house those systems and components that are important to safety.

In establishing the top level requirements for structural design, the staff used the General Design Criteria (GDC) of 10 CFR Part 50, Appendix A, as its basis. The primary general design criteria pertaining to the major structural system design are GDC 1, "Quality Standards and Records," GDC 2, "Design Bases for the Protection Against Natural Phenomena," GDC 4, "Environmental and Dynamic Effects Design Basis," GDC 14, "Reactor Coolant Pressure Boundary," GDC 16, "Containment Design," and GDC 50, "Containment Design Basis."

GDC 1 requires, in part, the need for structures, systems and components important to safety to be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed.

GDC 2 requires, in part, the need to design structures, systems, and components important to safety to withstand the effects of natural phenomena such as earthquakes, tornados, hurricanes, and floods without loss of capability to perform their safety functions, including the appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena.

GDC 4 requires, in part, the need to protect structures, systems, and components important to safety from dynamic effects including the effects of missiles, pipe whipping, and discharging fluids that may result from equipment failures and from events and conditions outside the nuclear power unit.

Initial Test Program

GDC 14 requires, in part, the need for the reactor coolant pressure boundary to be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.

GDC 16 requires, in part, the need for the reactor containment to provide an essentially leak-tight barrier against uncontrolled release of radioactivity to the environment.

GDC 50 requires, in part, the need for the reactor containment structure including access openings and penetrations to be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any loss-of-coolant accident.

Using the above GDC as its basis, the staff established the following top-level attributes to be verified by ITAAC:

- (1) pressure boundary integrity (GDC 14, 16 and 50)
- (2) normal loads (GDC 2)
- (3) seismic loads (GDC 2)
- (4) suppression pool hydrodynamic loads (GDC 4)
- (5) flood, wind, and tornado (GDC 2)
- (6) rain and snow (GDC 2)
- (7) pipe rupture (GDC 4)
- (8) codes and standards (GDC 1)

In addition, to ensure that the final as-built plant conforms to the certified design, GE provided ITAAC to reconcile the as-built plant with the structural design basis. A summary of the top-level structural design requirements for the major structural systems that are verified by the structures and systems in CDM Section 2.0 and the piping design information in CDM Section 3.3 is provided below.

Pressure Boundary Integrity

To ensure that the applicable requirements of GDC 14, 16, and 50 have been adequately addressed, ITAAC were established to verify the pressure boundary integrity of the RPV, piping, and primary containment for the ABWR. GDC 16 and 50 apply to the primary containment and GDC 14 applies to the RPV and the reactor coolant pressure boundary piping systems. The pressure integrity for these major structural systems are needed to ensure the defense-in-depth principle.

For the RPV and piping, hydrostatic tests performed in conjunction with the ASME Boiler and Pressure Vessel

Code, Section III are required by ITAAC. For the primary containment, a structural integrity test is required by ITAAC to be performed on the pressure boundary components of the primary containment in accordance with the ASME Boiler and Pressure Vessel Code, Section III. Because the requirements of GDC 14, 16, and 50 do not apply to the reactor, control, turbine, service, and radwaste buildings, ITAAC were not required to verify the pressure integrity for these other buildings.

Normal Loads

To ensure that the applicable requirements of GDC 2 have been adequately addressed, ITAAC were established to verify that the normal and accident loads have been appropriately combined with the effects of natural phenomena.

For piping systems, ITAAC require an analysis to reconcile the as-built piping design with the design-basis loads (which include the appropriate combination of normal and accident loads). For the RPV, the fabrication is performed primarily in the vendor's shop where adherence to design drawings is tightly controlled. Therefore, ITAAC for the as-built reconciliation of normal loads with accident loads for the RPV were deemed to be inappropriate. Instead, ITAAC verify that the ASME Code-required reports exist to document that the RPV has been designed, fabricated, inspected, and tested to Code requirements to ensure adequate safety margin.

Similarly, for safety-related buildings, ITAAC require an analysis for reconciling the as-built plant with the structural design basis loads (which include the combination of normal and accident loads with the effects of natural phenomena). The analysis results are to be documented in a structural analysis report, the scope and contents of which are described in the SSAR. The staff determined that the design of certain structures did not require verification by ITAAC, based on their safety significance. In particular, these ITAAC apply only to safety-related structures and are not applicable to the service and turbine buildings.

Seismic Loads

To ensure that the applicable requirements of GDC 2 have been adequately addressed, ITAAC were established to verify that the safety-related systems and structures have been designed to seismic loadings. Component qualification for seismic loads is addressed by ITAAC that were established for verifying the basic configuration of systems.

As discussed above for normal loads on piping systems and the RPV, ITAAC require an analysis to reconcile the as-built piping design with the design basis loads (which include seismic loads). For the RPV, ITAAC for the as-built reconciliation of seismic loads for the RPV were deemed to be inappropriate as previously discussed. Instead, ITAAC verify that the ASME Code-required reports exist for the RPV ensuring that the RPV has been designed, fabricated, inspected, and tested to ASME Code requirements.

For safety-related buildings, ITAAC require an analysis for reconciling the as-built plant with the structural design-basis loads (which include seismic loads). The analysis results are to be documented in a structural analysis report, as discussed above. These ITAAC apply only to safety-related structures and are not applicable to the service and turbine buildings. However, because the leakage path for fission products includes components within the turbine building, the turbine building is required to withstand the effects of a safe-shutdown earthquake. Therefore, ITAAC were established to verify that, under seismic loads, the collapse of the turbine building will not impair the safety-related functions of any structures or equipment located adjacent to or within the turbine building.

For non-seismic Category I SSCs, the need for ITAAC to verify that their failure will not impair the ability of nearby safety-related SSCs to perform their safety-related functions was assessed. Because the design detail and as-built and as-procured information for many non-safety-related systems (e.g., field-run piping and balance-of-plant systems) are not required for design certification and the spatial relationship between such systems and seismic Category I SSCs cannot be established until after the as-built design information is available, the non-seismic to seismic (II/I) interaction cannot be evaluated until the plant has been constructed. Accordingly, the design criteria for ensuring acceptable II/I interactions and a commitment for the COL applicant to describe the process for completion of the design of balance-of-plant and non-safety related systems to minimize II/I interactions and proposed procedures for an inspection of the as-built plant for II/I interactions have been specified as a COL action item in the SSAR.

Suppression Pool Hydrodynamic Loads

To ensure that the applicable requirements of GDC 4 have been adequately addressed, ITAAC were established to verify that the safety-related systems and structures have been designed to suppression pool hydrodynamic loadings, which include safety relief valve discharge and loss-of-coolant accident (LOCA) loadings. Component qualification for suppression pool hydrodynamic loads is

addressed by ITAAC established for verifying the basic configuration of systems.

As discussed above for seismic loads on piping systems and the RPV, ITAAC require an analysis to reconcile the as-built piping design with the design-basis loads (which include suppression pool hydrodynamic loads). For the RPV, ITAAC verify that the ASME Code-required reports exist to ensure that the RPV has been designed, fabricated, inspected, and tested to ASME Code requirements.

For the reactor building and primary containment including the internal structures, ITAAC require an analysis for reconciling the building as-built configuration with the structural design basis loads (which include suppression pool hydrodynamic loads). The as-built analysis results are to be documented in a structural analysis report as discussed above. The effects of suppression pool hydrodynamic loads do not extend beyond the reactor building, and, thus, ITAAC are not required to verify these loadings for the other ABWR building structures.

ITAAC also require the verification of the horizontal vent system, water volume, and the safety-relief valve discharge line quencher arrangement to ensure adequacy of the suppression pool hydrodynamic loads used for design.

Flood, Wind, Tornado, Rain, and Snow

To ensure that the applicable requirements of GDC 2 have been adequately addressed, ITAAC were established to verify that the safety-related systems and structures have been designed to withstand the effects of natural phenomena other than those associated with seismic loadings. The effects include those associated with flood, wind, tornado, rain, and snow.

These loadings do not apply to the RPV, the ASME Code Class 1, 2, and 3 piping systems and components, nor the primary containment because they are all housed within the safety-related buildings. For safety-related buildings, ITAAC require an analysis for reconciling the as-built plant with the structural design basis loads (which include the flood, wind, tornado, rain, and snow loads). Based on their safety significance, these ITAAC apply only to safety-related structures and are not applicable to the service and turbine buildings.

For flooding, ITAAC also require inspections to verify that water-tight doors exist, penetrations (except for water-tight doors) in the divisional walls are at least 2.5 m above the floor, and safety-related electrical, instrumentation, and control equipment are located at least 20 cm above the floor surface. In addition, for safety-related buildings, ITAAC require that external walls below flood level are

Initial Test Program

equal to or greater than 0.6 m to protect against water seepage, and penetrations in the external walls below flood level are provided with flood protection features.

Pipe Break

To ensure that the applicable requirements of GDC 4 have been adequately addressed, ITAAC were established to verify that the safety-related SSCs have been designed to the dynamic effects of pipe breaks. Component qualification for the dynamic effects of pipe breaks is addressed by ITAAC established for verifying the basic configuration of systems.

For the RPV, ITAAC that verify the basic configuration of the RPV system require an inspection of the critical locations that establish the bounding loads in the LOCA analyses for the RPV to ensure that the as-built areas not exceed the postulated break areas assumed in the LOCA analyses.

In addition, ITAAC have been established to verify by inspections of as-built, high-energy pipe break mitigation features and of the pipe break analysis report that safety-related SSCs be protected against the dynamic and environmental effects associated with postulated high-energy pipe breaks. ITAAC to verify pipe break loads are not required for the turbine, service, and radwaste buildings either because they are not safety-related structures or there are no high-energy lines located within the structure.

Codes and Standards

To ensure that the applicable requirements of GDC 1 have been adequately addressed, ITAAC were established to verify that appropriate codes and standards were used in the design and construction of safety-related systems and components. In general, the staff considered those codes and standards endorsed by the regulations under 10 CFR 50.55a in determining which codes and standards were appropriate for Tier 1 verification. The ASME Boiler and Pressure Vessel Code, Section III for Code Class 1, 2, and 3 systems and components was established as the code for the design and construction of ABWR piping systems and the RPV. For safety-related building designs, the staff based its safety findings on audits of ABWR design calculations which relied on specific codes and standards. These codes and standards are contained in SSAR Sections- 3.8.1, 3.8.2, 3.8.3, 3.8.4, and 3.8.5, and were identified in Section 3.8 of this report as material that, if considered for a change by an applicant or licensee that references the certified ABWR design, would constitute an unreviewed safety question, and therefore, would require NRC review and approval prior to implementation of the change.

Inspections will be conducted as a part of ITAAC to verify that ASME Code-required documents exist that demonstrate that the RPV, piping systems and containment pressure boundaries have been designed and constructed to their appropriate Code requirements. For other ASME Code components and equipment, the verification of Code compliance will be performed in conjunction with the quality assurance programs and by the authorized inspection agency as required by the ASME Boiler and Pressure Vessel Code.

As-built Reconciliation

To ensure that the final as-built plant structures are built in accordance with the certified design as required by 10 CFR Part 52, structural analyses will be performed which reconcile the as-built configuration of the plant structures with the structural design bases of the certified design. The structural analyses will be documented in structural analysis reports. Structural analysis reports will be verified in conjunction with ITAAC for the primary containment and the reactor, control, radwaste, and turbine buildings. The detailed supporting information on what is required for an acceptable analysis report is contained in SSAR Chapter 3.

Similarly for piping systems, an as-built analysis will be performed using the as-designed and as-built information. ITAAC will verify the existence of acceptable final as-built piping stress reports that conclude the as-built piping systems are adequately designed.

For the RPV, the key dimensions of the RPV system will be verified in conjunction with the basic configuration check of the system. The key dimensions of the RPV system and the acceptable variations of the key dimensions are provided in the certified design description.

For component qualification, tests, analyses, or a combination of tests and analyses will be performed for seismic Category I mechanical and electrical equipment (including connected instrumentation and controls) to demonstrate that the as-built equipment and associated anchorages are qualified to withstand design basis dynamic loads without loss of safety function. These test and analyses will be performed as a part of ITAAC to verify the basic configuration of the system in which the equipment is located.

14.3.2.3.7 Instrumentation and Controls (I&C) Task Group Review

The I&C Task Group's primary review responsibilities included a review of the CDM for I&C systems involving core protection and control, other miscellaneous I&C

systems, the additional I&C material in CDM Section 3.4 applicable to multiple systems of the design, and selected interface requirements. The material in CDM Section 3.4 is discussed further in Section 14.3.3.4 of this report. The group's secondary review responsibilities included ESF systems, reactivity control systems, and other systems using I&C equipment.

The figures in CDM Section 3.4 depict both safety-related and non-safety-related systems of the design. The block concept was used for developing the system control interface diagrams that were needed for depicting the configuration of the I&C system architecture. The I&C design described in the SSAR was to the level of control functional blocks, and therefore, the configuration in the CDM was to the same level.

The CDM entries were reviewed to confirm that the safety-related I&C system met the protection systems requirements of 10 CFR 50.55a(h), as well as the quality standards and records requirements of GDC 1, the protection against natural phenomenon requirements of GDC 2, the environmental and dynamic effects requirements of GDC 4, the instrumentation and control requirements of GDC 13, the control room requirements of GDC 19, the protection system design requirements of GDC 20, the protection system reliability and testability requirements of GDC 21, the protection system independence requirements of GDC 22, the protection system failure modes requirements of GDC 23, the protection system requirements for reactivity control malfunctions of GDC 25, and the protection against anticipated operational occurrences requirements of GDC 29. To meet the criteria of 10 CFR 50.55a(h), "Criteria for Protection Systems for Nuclear Generating Stations," and IEEE Standard 279-1971, the ITAAC entries were reviewed considering the following design issues:

- (1) General functional requirements for the system
- (2) Single failure criterion
- (3) Quality of components and modules (hardware and software)
- (4) Equipment qualification
- (5) Channel integrity and channel independence
- (6) Classification of equipment
- (7) Isolation devices
- (8) Single random failure

- (9) System inputs
- (10) Capability for sensor checks, tests and calibration
- (11) Channel bypasses, operating bypasses, indication of bypasses, and access to means for bypassing
- (12) Completion of protective action once initiated
- (13) Manual initiation
- (14) Information read-out
- (15) Identification

Standard ITAAC entries for several attributes of the I&C system were developed and used for basic configuration, divisionalized power supply, electrical isolation and physical separation (independence), and control room and remote shutdown system configuration. For those systems reviewed that were not safety-related systems, appropriate criteria from the SRP applicable to those systems were used.

For the microprocessor and digital control technology aspects of the I&C system design of the ABWR, GE did not provide complete design information in the SSAR. This was because the technology in this area is rapidly evolving and it is, therefore, important that the certified design description and ITAAC not "lock in" a design which could be obsolete at the time of construction. The process to complete the design, with appropriate acceptance criteria, is specified in CDM Section 3.4, with detailed supporting information in SSAR Chapter 7. The issues discussed in that material include the SSLC system, hardware and software development, electromagnetic compatibility, instrument setpoint methodology, environmental qualification of I&C equipment, and I&C system diversity and defense-in-depth considerations. Since the additional CDM information in CDM Section 3.0 apply to both safety- and non-safety-related I&C systems, the staff relied heavily on the information contained in those references in its reviews of the I&C systems.

The CDM Section 3.0 and SSAR contain criteria which describe the method to develop plans and procedures that will guide the design process throughout the lifecycle stages. The ITAAC provides the acceptance criteria for verifying the design through the stages while the SSAR adds the set of guidelines and standards that will provide more detailed criteria for the development of the design. The CDM has been written to incorporate the most important and general aspects (top-level requirements) from the

Initial Test Program

standards. The set of standards and criteria in the SSAR encompass the guidance for generating the plans that will be used in the computer software and hardware design process for the computer design throughout the lifecycle.

The certified design description and design development process continue for the lifetime of the plant. Any safety-related software that is changed or added after plant startup is required to either be developed using the certified design development process described in the computer CDM, or the licensee must submit a design process (together with the design bases) description that will produce software of the same or higher quality than the original certified design process, consistent with the CDM. The licensee will be required to use the approved software change procedure (SCP) based upon the certified design development process for the operation stage of the lifecycle.

14.3.2.4 Approval of the CDM for Structures and Systems

The staff performed a multidisciplinary review, utilizing several task groups, of the SSCs of the ABWR, in accordance with 10 CFR Part 52 and the guidance provided in SRMs related to design certification applications provided by the Commission. This review included information contained in multiple CDM and SSAR submittals to the staff, as discussed in the background portion of Section 14.3 of this report.

Based on the task group reviews, the staff concludes that the top-level design features and performance characteristics of the SSCs important to safety in the ABWR are appropriately described in the design descriptions of the CDM, and are acceptable.

Further, these top-level design features and performance characteristics can be adequately verified by the ITAAC provided by GE. Therefore, the staff concludes that the ITAAC in the CDM are necessary and sufficient to provide reasonable assurance that if the inspections, tests, and analyses are performed and the acceptance criteria met, the SSCs important to safety in a facility that references the design have been constructed and will operate in accordance with the design certification and applicable regulations.

14.3.2.5 DFSER Issues

Section 2.0, "Tier 1 Material for ABWR Systems," of the "Tier 1 Design Certification Material for the GE ABWR," was under staff review at the time the DFSER was issued. The staff stated that the final evaluation would be provided in the FSER. This was DFSER Open Item 14.1.2-1.

The DFSER contained preliminary comments on the second stage submittal of the CDM, titled "Tier 1 Design Certification Material for the GE ABWR Design - Stage 2 Submittal," dated March 31, 1992. These preliminary comments were documented in many sections of the DFSER based on their relationship to the detailed design information in various SSAR sections. The preliminary comments provided in the DFSER were intended to discuss the philosophy of development of the Tier 1 design certification material, to establish early staff positions on the material, and to provide an indication of the status of development of the material.

GE provided resolutions to all of the staff's comments in various revisions to the Tier 1 CDM, and provided revised supporting design information in various SSAR amendments, as discussed in the background part of Section 14.3 of this report. In these submittals, GE addressed all of the comments of the staff, the ACRS, and an independent review group. Based on the revised material in the CDM and SSAR submittals, the CDM development process, criteria, and methodology described in SSAR Section 14.3, and the review process discussed in this section of this report, the staff concluded that these issues were adequately addressed where appropriate in the CDM. This resolved DFSER Open Item 14.1.2-1.

The following is a list of issues identified in the DFSER that specifically cited the Tier 1 material, and that were resolved as discussed above. This list of issues considered for treatment in the CDM is not all-inclusive, nor were all issues listed necessarily incorporated into the final CDM. The staff considered many issues for treatment in the CDM, using general approach and criteria discussed in this section of this report. Other specific issues considered for treatment in the CDM may also be discussed where applicable in other sections of this report.

ITAAC Open Items In DFSER

<u>Item Number</u>	<u>Description of Item</u>
2.6-1	Additional Site Parameters from Tier 1
3.2.1-3	ITAAC-plant specific walkdown
3.4.1-2	ITAAC-flood protection
3.5.1.1-1	ITAAC-protect SSCs from internally-generated missiles
3.5.1.2-2	ITAAC-protection of safety-related (SR) equipment from missiles

ITAAC Open Items In DFSER

<u>Item Number</u>	<u>Description of Item</u>
3.5.1.4-1	ITAAC-missiles generated from natural events
3.5.2-1	ITAAC-protect S/R SSCs from failure of non-S/R SSCs
3.5.2-2	ITAAC-protect SSCs from externally-generated missiles
3.6.1-1	ITAAC-protection of safety equipment from DBA
3.8.6-1	ITAAC-generic building design concerns (11 items)
3.8.6-2	ITAAC-SW design concerns (2 items)
3.8.6-3	ITAAC-containment design concerns (4 items)
3.8.6-4	ITAAC-containment structures design concerns (2 items)
3.8.6-5	ITAAC-reactor vessel pedestal design concerns (3 items)
3.8.6-6	ITAAC-reactor building design concerns (6 items)
3.8.6-7	ITAAC-control building design concerns (4 items)
3.8.6-8	ITAAC-radwaste building design concerns (2 items)
3.8.6-9	ITAAC-yard structures-stack systems design concerns (3 items)
3.9.6.4-1	ITAAC-generic MOV sizing
3.11.3-1	ITAAC-equipment qualification (EQ) radiation concern
4.4-2	ITAAC-LPMS consistent with RG 1.133
4.6-1	ITAAC-control rod drive system
5.2.2-1	ITAAC-safety relief valve and fuel
5.2.5-1	ITAAC-reactor coolant system leakage detection
5.4.1-1	ITAAC-recirculation flow control
5.4.6-1	ITAAC-reactor core isolation cooling

ITAAC Open Items In DFSER

<u>Item Number</u>	<u>Description of Item</u>
5.4.7-1	ITAAC-residual heat removal
6.2.1.7-1	ITAAC-containment design
6.2.3.1-1	ITAAC-functional of secondary containment
6.2.4-1	ITAAC-standby gas treatment system
6.2.4.1-4	ITAAC-containment isolation system
6.2.6-9	ITAAC-containment leak testing
6.3.6-1	ITAAC-high pressure core flooders
6.4-1	ITAAC-control room habitability
6.4-2	ITAAC-control room environmental design
7.1.3.3-1	ITAAC/DAC-instrument setpoints, safety system logic and control, EQ, computer development
7.1.4-1	ITAAC/DAC-neutron monitoring systems
7.2.2.3-1	ITAAC-timeout predetermined safe states
7.2.8-1	ITAAC-safety hazards, sneak circuit, timing analyses
7.4.1-1	ITAAC-systems required for safe shutdown
7.4.2-1	ITAAC-use of remote shutdown panel
7.6.1-1	ITAAC-interlock systems
7.7.1-1	ITAAC-key features of the control system
8.2.1.4-1	ITAAC-interfaces
8.2.2.1-1	ITAAC-physical separation (circuits and transformers)
8.2.2.1-2	ITAAC-circuit separation
8.2.2.2-1	ITAAC-physical separation (power, instrumentation, etc.)
8.2.2.3-1	ITAAC-electrical independence
8.2.2.4-1	ITAAC-testing of the offsite power system

Initial Test Program

ITAAC Open Items In DFSER

<u>Item Number</u>	<u>Description of Item</u>
8.2.2.6-1	ITAAC-capacity and capability of the offsite power system
8.2.2.7-1	ITAAC-electrical grounding
8.2.3.1-1	ITAAC-independence between offsite circuits and Class 1E
8.2.3.2-1	ITAAC-independence during parallel operations
8.2.3.3-1	ITAAC-LOCA during parallel operations
8.2.3.3-2	ITAAC-LOOP during parallel operations
8.2.3.3-3	ITAAC-diesel generator (DG) protective relaying when DG is operating
8.2.3.3-4	ITAAC-synchronizing interlocks
8.2.3.4-1	ITAAC-independence during ops/fail of non-safety load
8.2.3.4-2	ITAAC-configuration offsite connection
8.2.3.4-3	ITAAC-separation of offsite and onsite Class 1E
8.3-1	ITAAC-onsite Class 1E design
8.3.1.2-1	ITAAC-safe shutdown with one division
8.3.2.1-1	ITAAC-conduits to open tray separation
8.3.2.2-1	ITAAC-separation of neutron monitoring raceways
8.3.2.3-1	ITAAC-separation of dc emergency lighting raceways
8.3.2.4-1	ITAAC-separation between Class 1E penetrations
8.3.2.4-2	ITAAC-separation between Class 1E & non-Class 1E
8.3.2.4-3	ITAAC-separation of non-Class 1E from Class 1E
8.3.2.5-1	ITAAC-separation/protection of cables outside cabinets and panels

ITAAC Open Items In DFSER

<u>Item Number</u>	<u>Description of Item</u>
8.3.2.6-1	ITAAC-separation of cables inside cabinets/panels
8.3.2.7-1	ITAAC-separation of cables approaching/exiting cabinets
8.3.2.8-1	ITAAC-independence/physical separation of equipment
8.3.2.9-1	ITAAC-ID power, instrumentation, control equipment
8.3.2.9-3	ITAAC-ID neutron monitoring, scram solenoid
8.3.3.1-1	ITAAC-protection of electrical penetrations
8.3.3.2-1	ITAAC-design and qualification of electrical equipment
8.3.3.3-1	ITAAC-seismic qualification of light bulbs
8.3.3.4-1	ITAAC-submergence
8.3.3.5-2	ITAAC-protection of redundant Class 1E (environment)
8.3.3.6-1	ITAAC-associate circuits
8.3.3.7-1	ITAAC-diesel generator protective relaying bypass
8.3.3.8-1	ITAAC-thermal overload
8.3.3.10-1	ITAAC-protective relay
8.3.3.11-1	ITAAC-fault interrupting capacity
8.3.3.12-1	ITAAC-control of design parameters for MOV
8.3.3.13-1	ITAAC-separation of raceways
8.3.3.14-1	ITAAC-electrical protection for scram and MSIV
8.3.3.15-1	ITAAC-safety buses grounding
8.3.3.16-1	ITAAC-control of access to Class 1E equipment
8.3.4-1	ITAAC-electrical independence
8.3.4.1-1	ITAAC-interconnections

Initial Test Program

ITAAC Open Items In DFSER

<u>Item Number</u>	<u>Description of Item</u>
8.3.4.2-1	ITAAC-CVCF power supplies
8.3.4.4-1	ITAAC-isolation between safety and non-safety buses
8.3.5-1	ITAAC-lighting system under design basis accident
8.3.5-2	ITAAC-lighting requirements
8.3.6.1-1	ITAAC-control of the electrical design process
8.3.7-2	ITAAC-testing surveillance
8.3.8.1-1	ITAAC-non-safety dc power system
8.3.8.2-1	ITAAC-capacity of the Class 1E 125v dc battery supply
8.3.8.4-1	ITAAC-class 1E ac standby power system
8.3.8.5-1	ITAAC-constant voltage/constant frequency (CVCF) capacity
8.3.8.6-1	ITAAC-battery charger
8.3.8.7-1	ITAAC-distribution system
8.3.9.1-1	ITAAC-station blackout (SBO) compliance
8.3.9.2-1	ITAAC-SBO coping capability
8.3.9.3-1	ITAAC/DAC
9.1.1-1	ITAAC-change new fuel storage interface to ITAAC
9.1.2-1	ITAAC-change spent fuel storage interface to ITAAC
9.1.2-2	ITAAC-review fuel storage facility ITAAC and T1
9.1.3-2	ITAAC-fuel pool cooling and cleanup system
9.1.4-1	ITAAC-light load handling system
9.1.5-1	ITAAC-reactor building crane capability under safe shutdown earthquake
9.2.4-2	ITAAC-sanitary and potable water system

ITAAC Open Items In DFSER

<u>Item Number</u>	<u>Description of Item</u>
9.2.5-1	ITAAC-ultimate heat sink
9.2.8-1	ITAAC-makeup water system (preparation)
9.2.9-1	ITAAC-makeup water condensate system
9.2.10-2	ITAAC-makeup water (purified) system
9.2.11-2	ITAAC-reactor building cooling water system
9.2.12-1	ITAAC-heating, ventilation, and air conditioning (HVAC) normal cooling water
9.2.13-1	ITAAC-HVAC emergency cooling water system
9.2.14-1	ITAAC-turbine building cooling system
9.2.15-2	ITAAC-reactor service water system
9.2.16-2	ITAAC-turbine service water system
9.3.1-3	ITAAC-compressed air systems
9.3.2.2-3	ITAAC-post accident sampling system (PASS)
9.3.5-1	ITAAC-standby liquid control system (SLCS)
9.3.8-1	ITAAC-radioactive drain transfer system
9.3.9-1	ITAAC-hydrogen water chemistry
9.3.10-1	ITAAC-oxygen injection
9.3.11-1	ITAAC-zinc injection system
9.4-1	ITAAC-HVAC systems
9.5.1.3-1	ITAAC-fire protection system
9.5.4.1-2	ITAAC-DG and auxiliary system, fuel oil storage and transfer
9.5.5-1	ITAAC-put interfaces into ITAAC
10.2.1-1	ITAAC-turbine generator
10.2.2-1	ITAAC-turbine disk test data

Initial Test Program

ITAAC Open Items In DFSER

<u>Item Number</u>	<u>Description of Item</u>
10.2.2-2	ITAAC-design bases for turbine design overspeed
10.3.1-1	ITAAC-main steam (7 items)
10.4.1-1	ITAAC-main condenser
10.4.2-1	ITAAC-main condenser evacuation system (7 items)
10.4.3-1	ITAAC-turbine gland seal system (7 items)
10.4.4-1	ITAAC-turbine bypass (7 items)
10.4.5-1	ITAAC-circulating water system (10 items)
10.4.7-1	ITAAC-condenser/feedwater (7 items)
11.0-1	ITAAC-radwaste system
12.3.5.3-1	ITAAC and Tier 1 radiation design submittal
13.3-3	ITAAC and Tier 1 for technical support center and operations support center
13.6.3.6-1	ITAAC-verify HVAC bulletproof features
14.1.1.5.2-1	ITAAC-roadmap of key analyses
14.1.1.5.3-1	ITAAC-certain systems may not have ITAAC
14.1.2-1	ITAAC-staff evaluation of system ITAAC
14.1.3.1-1	ITAAC-staff evaluation of generic ITAAC
14.1.3.3.3.6-1	ITAAC-structural design of small bore piping
14.1.3.3.3.9-1	ITAAC-buried piping design
14.1.3.3.4.1-1	ITAAC-confirmatory analysis on computer model adequacy
14.1.3.3.4.3-1	ITAAC-piping benchmark program
14.1.3.3.4.4-1	ITAAC-small bore piping decoupling criteria

ITAAC Open Items In DFSER

<u>Item Number</u>	<u>Description of Item</u>
14.1.3.3.5.2-1	ITAAC-60 year life cycle factor of 1.5
14.1.3.3.5.7-1	ITAAC-environmental effects in fatigue design, Cl. 1
14.1.3.3.5.7-2	ITAAC-method of including environmental effects of fatigue
14.1.3.3.5.8-1	ITAAC-environmental effect in fatigue design, Cl. 2
14.1.3.3.5.10-1	ITAAC-methodology to address thermal striping
14.1.3.3.5.13-1	ITAAC-inertial and seismic motion effects
14.1.3.3.5.17-1	ITAAC-modal damping for composite structures
14.1.3.3.5.18-1	ITAAC-minimum temperature for thermal analyses
14.1.3.3.6-1	ITAAC-pipe support criteria (8 items)
14.1.3.3.7-1	ITAAC-high energy line break criteria
14.1.3.3.9.1-1	ITAAC-fatigue cumulative usage factor of 1.0
14.1.3.3.9.13-1	ITAAC-Tier 1 piping design description
14.1.3.8-1	ITAAC-reliability assurance program
14.1.3.9-1	ITAAC-welding
14.1.4-1	ITAAC-interface requirements
14.1.5-1	ITAAC-site parameters
14.2.12.7-1	ITAAC-design certification material initial test program
18.9.1-1	ITAAC-design description
19.1.2.2.2-2	ITAAC-fire barriers/separation
19.1.2.2.2-3	ITAAC-interface design for fires
19.1.2.4.2-1	ITAAC-PRA for internal, external events
19.1.5.2-3	ITAAC-interfacing piping

ITAAC Open Items In DFSER**Item Number Description of Item**

- 19.1.5.4-3 ITAAC-reliability of interfacing systems
- 19.1.5.6.3-1 ITAAC-human reliability analyses

ITAAC Confirmatory Items

- 19.1.6.3.2-3 ITAAC-seismic capacity of equipment
- 19.1.6.4-4 ITAAC-fire barrier installation, smoke mitigation
- 19.1.6.4-5 ITAAC-fire for COL design
- 5.2.4-1 ITAAC-add discussion of PSE and 89 code
- 7.2.2.1-1 ITAAC-verify maximum transmission distance
- 7.2.2.1-2 ITAAC-include accuracy in setpoint methods
- 7.2.2.5-2 ITAAC-Tier 1-eliminate test jumpers and lifted leads
- 7.2.3-1 ITAAC-verify bypasses are annunciated
- 7.2.8-1 ITAAC-Tier 1-software metrics to track error rate
- 7.2.8-4 ITAAC-equipment to be tested for low range EMI
- 7.7.1.5-1 ITAAC-reactor protection system trip ID in computer
- 7.10.2-1 ITAAC-follow EPRI for operating experience
- 8.2.3.4-1 ITAAC-independence safety vs non-safety systems
- 12.2.3-1 ITAAC/DAC include former interfaces
- 14.1.3.3.3.8-2 ITAAC-verification of seismic/non-seismic interactions
- 18.9.2.2.8-4 ITAAC-tests and analyses in CDM Table 3.6

The following issues were incorrectly classified in the DFSER as ITAAC COL Action Items, but were also resolved based on the revised CDM material, supporting SSAR information, the discussion in SSAR Section 14.3, and the discussion in this section of this report.

ITAAC COL Action Items**Item Number Description of Action Item**

- 9.5.7-1 ITAAC-DG lube oil system design criteria
- 9.5.8-1 ITAAC-DG combustion air system flow capacity

14.3.3 Additional Certified Design Material

This section of the ABWR CDM provides additional certified design material for design and construction activities that are applicable to more than one system. There are five entries in this CDM section, and these are discussed in the appropriate paragraphs that follow. The first four entries describe design related processes and associated DAC for the ABWR, and the fifth entry describes the ITP for a facility referencing the certified design. The design description for each entry describes its scope and applicability to the ABWR design. Amplifying information on CDM Section 3.0 is contained in SSAR Section 14.3.3. The material in this section of the CDM applies to the individual systems of the ABWR design contained in CDM Section 2.0, and the staff's review of the material in CDM Section 2.0 is contained in Section 14.3.2 of this report. The staff's safety evaluation for each design area where the DAC are used is contained in the section of this report applicable to the area.

Design Acceptance Criteria (DAC)

Design and engineering information for some areas of the design was not provided by GE at a level of detail customarily reviewed by the staff in making a final safety determination. GE provided less detailed information in these areas because GE believed they were either areas of rapidly changing technology and it would have been detrimental to freeze the details of the design many years before an actual plant was ready to be constructed, or because GE believed they were areas for which GE did not have sufficient as-built or as-procured information to complete the final design. Areas of rapidly changing technology included control room and RSS design (human factors) and advanced instrumentation and controls. Areas dependent on as-built or as-procured information included piping design and radiation shielding, ventilation, and airborne monitoring design. The staff provided its views

Initial Test Program

on the DAC to the Commission in SECY-92-053, "Use of Design Acceptance Criteria During 10 CFR Part 52 Design Certification Reviews," dated February 19, 1992.

The design information and appropriate design methodologies, codes, and standards provided in the SSAR, together with the design descriptions and DAC, are sufficiently detailed to provide an adequate basis for the staff to make a final safety determination regarding the design, subject only to satisfactory design implementation and verification of the DAC by the COL applicant or licensee. The DAC are a set of prescribed limits, parameters, procedures, and attributes upon which the NRC relies, in a limited number of technical areas, in making a final safety determination in support of the ABWR design certification. The acceptance criteria for the DAC are objective; that is, they are inspectable, testable, or subject to analysis using pre-approved methods, and must be verified as a part of the ITAAC performed to demonstrate that the as-built facility conforms to the certified design. Thus, the acceptance criteria for DAC are specified together with the related ITAAC in the Tier 1 material, and both are part of the design certification. The DAC and the ITAAC, when met, ensure that the completed design and as-constructed plant conforms to the design certification. The material in the SSAR for each of the DAC areas includes, as appropriate, sample calculations or other supporting information to illustrate methods that are acceptable to the staff for meeting Tier 1 DAC commitments.

The structure of each area where DAC are used is the same as for the other areas of the design that are verified by ITAAC. The structure consists of three parts: the Tier 1 design description, the corresponding DAC, and the Tier 2 supporting information in the SSAR for the DAC. The staff has based its safety findings for the areas where DAC are used on the Tier 2 information specified in the SSAR, including applicable design methodologies, codes and standards, contingent on verification that the design has been properly implemented according to the Tier 1 design descriptions and the corresponding DAC.

For the two areas of rapidly changing technology, control room and RSS design (human factors) and instrumentation and controls design, the design descriptions and DAC delineate the process and requirements that a COL applicant or licensee must implement to develop the design information required in each area. Acceptance criteria are specified in the CDM for the development process at various stages of detailed design and subsequent construction and testing. The COL applicant or licensee is required to develop the procedures and test programs necessary to demonstrate that the DAC requirements are

met at each stage. Similar to ITAAC, the COL applicant or licensee will certify to the NRC that the design through that stage is in compliance with the certified design. The NRC will review and inspect the work to confirm that the COL applicant or licensee has adequately implemented the commitments of the DAC at these stages. The process is referred to as a phased DAC because it consists of a set of sequential steps or stages that require successful completion. A COL applicant or licensee is not required to certify that each phase is completed sequentially. However, if the staff determines that a DAC was not successfully met, the design process may be required to be repeated to meet the DAC, possibly requiring a change to the as-built system design.

14.3.3.1 Human Factors Engineering DAC

The human factors aspects of the ABWR control room and remote shutdown system (RSS) design are provided in SSAR Chapter 18, "Human Factors," and together with the associated DAC in CDM Section 3.1, "Human Factors Engineering," are evaluated in Chapter 18 of this report. GE did not develop a final control room and RSS design before design certification because this is an area of rapidly changing technology. Instead, GE provided the processes and acceptance criteria by which the details of the design in this area would be developed, designed, and evaluated. GE provided amplifying information regarding the processes in this area in SSAR Section 14.3.3.1. The material in CDM Section 3.1 applies to the human factors design of the control room and the RSS. The implementation of the process and the design is the responsibility of the COL applicant or licensee.

Complete detailed human-system interface (HSI) design information was not available for staff review. The basis for the staff's review in this area was a HFE Program Review Model (PRM) developed by the staff. The staff's certification review in the control room design area was based on a design and implementation process plan. The staff informed the Commission of the development of the DAC in this area in SECY-92-299, "Development of Design Acceptance Criteria (DAC) for the Advanced Boiling Water Reactor (ABWR) in the Areas of Instrumentation and Controls (I&C) and Control Room Design," dated August 27, 1992.

The staff developed the HFE PRM, contained in Appendix J of this report, to serve as a technical basis for the review of the design process and DAC proposed by GE for certification of the ABWR control room and remote shutdown station design. The HFE PRM is (1) based upon currently accepted HFE practices, (2) well-defined, and (3) validated through experience with the development of complex, high-reliability systems in other industrial and

military applications. The review model identifies the important HFE elements in a system development, design, and evaluation process that are necessary and sufficient requisites to successful integration of human factors in complex systems. The review model also identifies aspects of each HFE element that are key to a safety review, and describes acceptance criteria by which the HFE elements can be evaluated. The HFE PRM has eight program elements, each of which contain both general and more specific acceptance criteria.

The CDM describes the process to develop the HSI design information for the control room and RSS based on human factors systems analyses and human factors principles. The design effort will be directed by a multi-disciplinary HFE design team comprised of personnel with expertise in HFE and other technical areas relevant to the HSI design, evaluation and operations. The HSI design team shall develop a program plan to establish methods for implementing the HSI design through a process of human factors system analyses as discussed in CDM Figure 3.1, "Human-System Interface Design Implementation Process." The details of implementation of each stage of the development process are described in CDM Section 3.1, together with the related acceptance criteria. Detailed supporting information is contained primarily in SSAR Chapter 18, Appendix 18E.

The staff conducted a complete and thorough review of the ABWR CDM to ensure that the general criteria of the eight program elements in the HFE PRM were appropriately addressed in the Tier 1 CDM. The Tier 2 SSAR material contains more detailed guidelines and applicable guidance documents. The staff also conducted a review of the SSAR material to ensure that the specific acceptance criteria in the HFE PRM were appropriately addressed. The staff reviewed the CDM and SSAR Section 14.3.3.1 in accordance with the requirements in Part 52 and the guidance provided in SRMs related to design certification applications provided by the Commission. This review included information contained in multiple submittals to the staff as listed in the background part of Section 14.3 of this report.

The material in SSAR Chapter 18 provides design information and defines design processes that are acceptable for use in meeting the acceptance criteria in the CDM. However, the SSAR information may be changed by a COL applicant or licensee referencing the certified design in accordance with a "50.59-like" process. The staff's evaluation of the ABWR design for the control room is based on the design processes and acceptance criteria material in the DAC and the SSAR, especially those defined in SSAR Section 18E. Consequently, the staff indicated in Section 18 of this report that any

proposed changes to SSAR Section 18E constitutes an unreviewed safety question and, therefore, must be submitted to the NRC for review and approval prior to implementation.

Conclusions

On the basis of the above, the staff concludes that the top-level design processes, features and performance characteristics of the human factors aspects of SSCs important to safety in the ABWR are appropriately described in the design descriptions of the CDM, and are acceptable.

Further, these top-level design processes, features and performance characteristics can be adequately verified by the DAC provided by GE. Therefore, the staff concludes that the DAC in the CDM are necessary and sufficient to provide reasonable assurance that if the inspections, tests, and analyses are performed and the acceptance criteria met, the human factors aspects of SSCs important to safety in a facility that references the design have been designed, constructed and will operate in accordance with the design certification and applicable regulations.

14.3.3.2 Radiation Protection DAC

The radiation protection aspects of the ABWR design are provided in SSAR Chapter 12, "Radiation Protection," and together with the associated DAC in CDM Section 3.2, "Radiation Protection," are evaluated in Chapter 12 of this report. GE did not provide the complete design information in this design area before design certification because the radiation shielding design and the calculated concentrations of airborne radioactive material were dependent upon as-built and as-procured information of plant systems and components. Therefore, GE was not able to describe the ABWR radiation source terms (i.e., the quantity and concentration of radioactive materials contained in, or leaking from plant systems) in sufficient detail to allow the staff to verify the adequacy of the shielding design, ventilation system designs, or the design and placement of the airborne radioactivity monitors. Instead, GE provided the processes and acceptance criteria by which the details of the design in this area would be developed, designed, and evaluated. GE provided amplifying information regarding the processes in this area in SSAR Section 14.3.3.2. This material in CDM Section 3.2 applies to the radiological shielding and ventilation design of the reactor building, turbine building, control building, service building, and radwaste building. The implementation of the process and the design is the responsibility of the COL applicant or licensee.

Initial Test Program

The acceptance criteria in the DAC are taken from the acceptance criteria in the applicable section of Chapter 12 of the SRP. The analysis methods and source term assumptions specified in the DAC are consistent with approved methods and assumptions listed in the SRP. The SRP is the basis for the staff's safety review of the ABWR design. Therefore, demonstrating that the final design meets these DAC with the methods and assumptions specified in Tier 1 ensures that the as-built ABWR design meets the applicable acceptance criteria of the SRP and the associated regulations and staff technical positions. The staff informed the Commission of the development of the DAC in this area in SECY-92-196, "Development of Design Acceptance Criteria (DAC) for the Advanced Boiling Water Reactor (ABWR)," dated May 28, 1992.

The DAC in the Tier 1 information address the verification of the plant radiation shielding design and the plant airborne concentrations of radioactive materials (e.g., the ventilation system and airborne monitoring system designs). The DAC require the COL applicant to calculate radiation levels and airborne radioactivity levels within the plant rooms and areas to verify the adequacy of these design features during plant construction (concurrently with the verification of the ITAAC). The plant rooms and areas to which the DAC apply are given in the figures in CDM Section 3.2. Detailed supporting information is contained in SSAR Chapter 12.

The criteria in CDM Table 3.2a, Items 1 and 2, ensure that the radiation shielding design (either that provided for by the plant structures, or design permanent or temporary shielding) is adequate to ensure that the maximum radiation levels in plant areas are commensurate with the area's access requirements so radiation exposures to plant personnel can be maintained as low as reasonably achievable (ALARA) during normal plant operations and maintenance. Item 4 in Table 3.2a ensures that adequate shielding is provided for those areas of the plant that may require occupancy to permit an operator to aid in the mitigation of or the recovery from an accident. Item 4 of Table 3.2a ensures that the contribution to the radiation dose from gamma shine (particularly from the turbine building) to a member of the public (off site) will be a small fraction of the EPA dose limit in 40 CFR Part 190.

The criteria in CDM Table 3.2b, Item 1, ensures that the plant provides adequate containment and ventilation flow rates to control the concentrations of airborne radioactivity to levels commensurate with the access requirements of areas in the plant. Item 2 in Table 3.2b ensures that once the concentrations of airborne radioactivity are determined per Item 1 above, the required airborne monitors are provided in the appropriate locations in the plant.

The staff conducted a complete and thorough review of the GE ABWR CDM material to ensure that the SRP guidelines for radiation protection design were appropriately addressed in both the Tier 1 CDM and the SSAR. The staff's evaluation included the analysis methods, design procedures, acceptance criteria, and related ITAAC that are to be used for the completion and verification of the ABWR radiation protection design. The SSAR information contains more detailed guidelines and applicable documents. The staff reviewed the CDM and SSAR Section 14.3.3.2 in accordance with the requirements in Part 52 and the guidance provided in SRMs related to design certification applications provided by the Commission.

Conclusions

On the basis of the above, the staff concludes that the top-level design processes, features and performance characteristics of the radiation protection aspects of SSCs important to safety in the ABWR are appropriately described in the design descriptions of the CDM, and are acceptable.

Further, these top-level design processes, features and performance characteristics can be adequately verified by the DAC provided by GE. Therefore, the staff concludes that the DAC in the CDM are necessary and sufficient to provide reasonable assurance that if the inspections, tests, and analyses are performed and the acceptance criteria met, the radiation protection aspects of SSCs important to safety in a facility that references the design have been designed, constructed and will operate in accordance with the design certification and applicable regulations.

14.3.3.3 Piping Design DAC

The piping design aspects of the ABWR design are provided in SSAR Chapter 3, "Structures, Components, Equipment, and Systems," and together with the associated DAC in CDM Section 3.3, "Piping Design," are evaluated in Section 3.12 of this report. GE did not provide the complete design information in this design area before design certification because the piping design was dependent upon as-built and as-procured information. Instead, GE provided the processes and acceptance criteria by which the details of the design in this area would be developed, designed, and evaluated. GE provided amplifying information regarding the processes in this area in SSAR Section 14.3.3.3. The material in CDM Section 3.3 applies to ABWR piping systems classified as nuclear safety-related, and to non-nuclear safety systems as specified in the Tier 1 material for the individual systems in CDM Section 2.0. The implementation of the process

and the design is the responsibility of the COL applicant or licensee.

The staff used the SRP guidelines to evaluate the piping design information in the ABWR CDM and SSAR and performed a detailed audit of the piping design criteria, including sample calculations. The staff evaluated the adequacy of the structural integrity and functional capability of safety-related piping systems. The review was not limited to the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Class 1, 2, and 3 piping and supports, but included buried piping, instrumentation lines, the interaction of non-seismic Category I piping with seismic Category I piping, and any safety-related piping designed to industry standards other than the ASME Code. The staff's evaluation included the analysis methods, design procedures, acceptance criteria, and related ITAAC that are to be used for the completion and verification of the ABWR piping design. The staff's evaluation included both CDM and SSAR information regarding the applicable codes and standards, analysis methods to be used for completing the piping design, modeling techniques, pipe stress analyses criteria, pipe support design criteria, high-energy line break criteria, and leak-before-break (LBB) approach applicable to the ABWR. The staff informed the Commission of the development of the DAC in this area in SECY-92-196, "Development of Design Acceptance Criteria (DAC) for the Advanced Boiling Water Reactor (ABWR)," dated May 28, 1992.

The material in CDM Section 3.3 provides the design process to develop the piping for the nuclear safety-related (seismic Category I) systems of the ABWR design. Piping systems that must remain functional during and following an SSE are designated as seismic Category I and are further classified as ASME Code Class 1, 2, or 3. The piping systems and their components are designed and constructed in accordance with the ASME Code requirements identified in the individual systems of the ABWR design. The CDM ensures that the piping systems will be designed to perform their safety-related functions under all postulated combinations of normal operating conditions, system operating transients, postulated pipe breaks, and seismic events. The material in the CDM section also addresses the consequential effects of pipe ruptures such as jet impingement, potential missile generation, and pressure and temperature effects.

GE specified three ITAAC in the CDM to ensure the design process for piping systems was as described in the design description. The first ITAAC specified in the CDM requires that an ASME Code certified stress report exists to ensure that the ASME Code Class 1, 2, or 3 piping systems are designed to retain their pressure integrity and

functional capability under internal design and operating pressures and design basis loads. The specific contents and requirements of the certified stress report are contained in the ASME Code. As used in this report, an ASME Code certified stress report is the design document required by ASME Code, Section III, Subarticle NCA-3550. A certified piping stress report provides assurance that requirements of the ASME Code, Section III for design, fabrication, installation, examination, and testing have been met and that the design complies with the design specifications.

The second ITAAC requires that a pipe break analysis report exists that documents that SSCs that are required to be functional during and following an SSE have adequate high-energy pipe break mitigation features, or alternatively, that a leak-before-break report exists for those sections of piping systems qualified for leak-before-break design. As discussed in the design description, the pipe break analysis report specifies the criteria used to postulate pipe breaks and the analytical methods used to perform pipe breaks and confirms the adequacy of the results of the pipe break analyses. This verification provides assurance that the high-energy line break analyses have been completed and meet the following certified design commitments. For postulated pipe breaks, the Pipe Break Analysis Report shall confirm that: (1) piping stresses in the containment penetration area shall be within their allowable stress limits, (2) pipe whip restraints and jet shield designs shall be capable of mitigating pipe break loads, (3) loads on safety-related SSCs shall be within their design load limits, and (4) SSCs are protected or are qualified to withstand the environmental effects of postulated failures. The Pipe Break Analysis Report shall conclude that, for each postulated piping failure, the reactor can be shut down safely and maintained in a safe, cold shutdown condition without offsite power. Detailed information that supports this ITAAC is contained in SSAR Chapter 3.

The third ITAAC requires that an as-built piping stress report exists that documents the results of an as-built reconciliation analysis confirming that the final piping system has been built in accordance with the ASME Code certified stress report. The report provides an overall verification that the as-constructed piping system is consistent with the certified design commitments. Although similar to the first ITAAC, this verification also provides assurance that modification of any document used for construction from the corresponding document used for design analysis has been reconciled with the certified stress report discussed above. This documentation may become part of the certified stress report.

Initial Test Program

As discussed in the advance SER, GE stated that it intended to provide, in a future SSAR amendment, amplifying information in the SSAR to support the piping DAC. This was Confirmatory Item F14.3.3.3.1-1. GE provided this amplifying information in Amendment 34. The staff finds this acceptable. This resolved Confirmatory Item F14.3.3.3-1.

The staff conducted a complete and thorough review of the GE ABWR CDM material to ensure that the SRP guidelines for piping design were appropriately addressed in both the Tier 1 CDM and the SSAR. The staff's evaluation included the analysis methods, design procedures, acceptance criteria, and related ITAAC that are to be used for the completion and verification of the ABWR piping design. The Tier 2 SSAR material contains more detailed guidelines and applicable documents. The staff reviewed the CDM and SSAR Section 14.3.3.3 in accordance with the requirements in Part 52 and the guidance provided in SRMs related to design certification applications provided by the Commission. This review included information contained in multiple submittals to the staff as listed in the introductory part of Section 14.3 of this report.

Selected material in SSAR Chapter 3 provides design information and defines design processes that are acceptable for use in meeting the piping DAC in the CDM. However, the SSAR information may be changed by a COL applicant or licensee referencing the certified design in accordance with a "50.59-like" process. The staff's evaluation of the ABWR design for piping systems is based on the design processes and acceptance criteria material in the DAC and the SSAR. Consequently, the staff indicated in Section 3.12 of this report that any proposed changes to selected aspects of these piping design processes described in the appropriate SSAR sections constitutes an unreviewed safety question and, therefore, must be submitted to the NRC for review and approval prior to implementation.

Conclusions

On the basis of the above, the staff concludes that the top-level design processes, features and performance characteristics of the piping design aspects of SSCs important to safety in the ABWR are appropriately described in the design descriptions of the CDM, and are acceptable.

Further, these top-level design processes, features and performance characteristics can be adequately verified by the DAC provided by GE. Therefore, the staff concludes that the DAC in the CDM are necessary and sufficient to provide reasonable assurance that if the inspections, tests, and analyses are performed and the acceptance criteria

met, the piping design aspects of SSCs important to safety in a facility that references the design have been designed, constructed and will operate in accordance with the design certification and applicable regulations.

14.3.3.4 Instrumentation and Controls (I&C) DAC and Other I&C ITAAC

The I&C aspects of the ABWR design are provided in SSAR Chapter 7, "Instrumentation and Control Systems," and together with the associated DAC and other multi-system I&C related ITAAC in CDM Section 3.4, "Instrumentation and Control," are evaluated in Chapter 7 of this report. GE did not develop a final design for I&C before design certification because this is an area of rapidly changing technology. Instead, GE provided the processes and acceptance criteria by which the details of the design in this area would be developed, designed, and evaluated. GE provided amplifying information regarding the processes in this area in SSAR Section 14.3.3.4. The material in CDM Section 3.4 applies to the design of both safety related and non-safety related I&C systems of the ABWR. These I&C systems are described in CDM Section 2.0, and the staff's review of these systems is described in Section 14.3.2 of this report. The implementation of the process and the design is the responsibility of the COL applicant or licensee.

The staff used the SRP guidelines to review the I&C design information in the ABWR CDM and SSAR to confirm that both the safety-related and non-safety-related I&C systems met the appropriate acceptance criteria of the SRP. The staff also used the Commission guidance contained in a SRM of July 15, 1993, related to SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light Water Reactor (ALWR) Designs," dated April 2, 1993. The staff informed the Commission of the development of the DAC in this area in SECY-92-299, "Development of Design Acceptance Criteria (DAC) for the Advanced Boiling Water Reactor (ABWR) in the Areas of Instrumentation and Controls (I&C) and Control Room Design," dated August 27, 1992.

CDM Section 3.4 has multiple entries addressing three key issues associated with the I&C design. These issues include the design of the SSLC system, the development and qualification processes for I&C systems, and design features that provide diverse backup as protection against common-mode failures in the SSLC. These issues and their relationships to other systems of the design are illustrated in the figures in CDM Section 3.4, which contain a block diagram showing the SSLC logic and control, a depiction of the integrated hardware and

AIM

Association for Information and Image Management

1100 Wayne Avenue, Suite 1100

Silver Spring, Maryland 20910

301/587-8202

Centimeter

Inches

MANUFACTURED TO AIM STANDARDS
BY APPLIED IMAGE, INC.

7 of 10

software development process for I&C systems, and a diagram showing the interfaces of the SSLC system with other I&C systems in the design. Detailed supporting information for the CDM is contained in SSAR Chapter 7.

CDM Section 3.4A, "Safety System Logic and Control," contains material for the SSLC. The SSLC integrates the automatic decision-making and trip logic functions, and manual initiation functions associated with the safety actions of the safety-related systems. Safety-related trip logic and monitoring of plant protection system resides in SSLC equipment. SSLC equipment comprises microprocessor-based, software-controlled signal processors that perform signal conditioning, setpoint comparison, trip logic, system initiation and reset, self-test, calibration, and bypass functions. The signal processors associated with a particular safety-related system are an integral part of that system and do not belong to SSLC.

CDM Section 3.4B, "I&C Development and Qualification Processes," contains the DAC for the I&C area of the design. The DAC are contained in four subsections that describe (1) design processes and acceptance criteria to be used for safety-related systems using programmable microprocessor-based control equipment, (2) a program to assess and mitigate the effects of electromagnetic interference on I&C equipment, (3) a program to establish setpoint for safety-related instrument channels, and (4) a program to qualify safety-related I&C equipment for in-service environmental conditions.

The subsection of CDM Section 3.4B titled, "Hardware and Software Development Process," describes hardware and software development processes to be used in the design, testing, and installation of I&C equipment. The following discussion addresses the considerations made in reviewing the entries for this subsection of the CDM.

The primary function of this development process is to implement the functional instrumentation and control requirements described in the CDM and the SSAR for the systems which comprise the ABWR. The decomposition of the functional system (SSLC, RPS, ARI, etc.) requirements to specific computer hardware and software components to perform the various tasks is accomplished using the structured design process described below.

The CDM includes the description of the design process to be followed for hardware and software development, design commitments, the inspections, tests, and analysis to be performed to verify that the design is consistent with the commitments, and the appropriate acceptance criteria against which the design will be judged. This ITAAC describes attributes of the process to be used to develop the

software as well as attributes of the final software product. The ITAAC for software and hardware describes the following design stages within the design process:

- (1) Planning
- (2) Design definition
- (3) Software design
- (4) Software coding
- (5) Integration
- (6) Validation
- (7) Change control

The CDM and SSAR contain criteria which describe the method to develop plans and procedures that will guide the design process throughout the lifecycle stages. The ITAAC provides the acceptance criteria for verifying the design through the stages listed above, while the SSAR adds the set of guidelines and standards that will provide more detailed criteria for the development of the design. The CDM has been written to incorporate the most important and general aspects (top-level requirements) from the standards. The set of standards and criteria in the SSAR encompass the guidance for generating the plans that will be used in the computer software and hardware design process for the computer design throughout the lifecycle. These plans are described below.

The software QA (SQA) plan describes the software-specific activities that are to be performed and controlled in addition to the approved QA plan (in accordance with 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants") for the total ABWR design. The SQA plan establishes the criteria under which the other software development plans will be generated. The software management plan (SMP) establishes the organization and authority structure for the design, the procedures to be used, and the interrelationships between major activities. The software configuration management plan (CMP) provides the means to identify software products, control and implement changes, and record and report change implementation status. The software development plan (SDP) describes a development process, tools documentation, and products developed according to the software lifecycle. The verification and validation plan (V&VP) describes the method to ensure that the requirements of each phase or stage of the design process (lifecycle) are fully and accurately implemented into the next phase. The software safety plan (SSP) describes the safety and hazards analyses that will be performed. The software operation and maintenance plan (SOMP) includes the procedures required to ensure that the software will be operated correctly and that the quality of the software is maintained. GE has combined these plans into a software management plan, a

Initial Test Program

configuration management plan, and a verification and validation plan.

The ITAAC activities completed by the COL applicant will be inspected by the NRC to verify conformance with the requirements at several stages during the digital control system design process or stage of the lifecycle. The documents which demonstrate satisfactory implementation of the ITAAC will be available for inspection during the NRC audit at the completion of each of the above stages. The stages or phases described by GE are shown in Figure 3.4d. The NRC audit and the COL applicant conformance review points are shown in Figure 7.1-2 of the ABWR FSER. These stages correspond with the phases described by GE in the CDM. The actual stages, including the conformance review and audit points, will be determined for each of the software products to be developed when design implementation is scheduled to begin.

At each stage, the design development must be verified by the COL applicant to be in accordance with the certified design process and the detailed design developed (through that stage) to be in conformance with the certified design. Upon completion of ITAAC activities for each stage, the COL applicant will certify to the NRC that the stage has been completed and the design and construction completed up through that stage is in compliance with the certified design. Although not required, the COL applicant should satisfactorily complete ITAAC activities at each stage prior to proceeding to the next stage of the design development process. Failure to successfully complete the ITAAC at a stage, as determined by the conformance review or the NRC audit, may require repeating an earlier stage ITAAC or changing the system design. The NRC staff will identify any open issues which require resolution for each stage of the ITAAC. Significant open issues which are not resolved could result in the NRC staff concluding that the ITAAC had not been satisfactorily completed.

The subsection of Section 3.4B titled "Electromagnetic Compatibility" describes the process to ensure that I&C equipment is able to function properly when subjected to an electromagnetic environment. An electromagnetic compatibility (EMC) compliance plan to confirm the level of immunity to electrical noise is included in the design, installation, and testing of I&C equipment. The plan is structured on the basis that EMC of I&C equipment is verified by factory testing and site testing of both individual components and interconnected systems to meet electromagnetic compatibility requirements.

The subsection of Section 3.4B titled "Instrument Setpoint Methodology" describes the process to ensure that setpoints for initiation of safety-related functions are determined,

documented, installed, and maintained. The process (the instrument setpoint methodology) establishes a program for specifying requirements for documenting the bases for selection of trip setpoints, accounting for instrument inaccuracies, response testing, and replacement of instrumentation.

The subsection of Section 3.4B titled "Equipment Qualification" describes the process to ensure that qualification of safety-related I&C equipment is able to complete its safety-related function under the environmental conditions that exist up to and including the time the equipment has finished performing that function. An equipment qualification program is established that ensures qualification specifications consider conditions that exist during normal, abnormal, and design-basis accident events in terms of their cumulative effect on equipment performance for the period up to the end of equipment life.

CDM Section 3.4C, "Diversity and Defense-In-Depth Considerations," addresses the concern that software design faults or other initiating events common to redundant, multidivisional logic channels of I&C protection systems could disable significant portions of the plant's safety functions at the moment when these functions are needed to mitigate an accident, and addresses the diverse backup features that are provided for the primary automatic logic. Diversity is provided in the form of hard-wired backup for reactor trip, diverse display of important process parameters, defense-in-depth arrangement of equipment, and other equipment diversity.

The staff conducted a complete and thorough review of the GE ABWR CDM material to ensure that the SRP guidelines and Commission guidance for I&C design were appropriately addressed in both the CDM and the SSAR. The staff's evaluation included the analysis methods, design procedures, acceptance criteria, and related ITAAC that are to be used for the completion and verification of the ABWR I&C design. The SSAR material contains more detailed guidelines and applicable guidance documents. The staff reviewed the CDM and SSAR Section 14.3.3.4 in accordance with the requirements in Part 52 and the guidance provided in SRMs related to design certification applications provided by the Commission. This review included information contained in multiple submittals to the staff as listed in the background part of Section 14.3 of this report.

Selected material in SSAR Section 7.2 provides detailed design information and defines design processes that are acceptable for use in meeting the I&C DAC in the CDM. However, the SSAR information may be changed by a COL applicant or licensee referencing the certified design

in accordance with a "50.59-like" process. The staff's evaluation of the ABWR design for I&C systems is based on the design processes and acceptance criteria material in the DAC and the SSAR. Consequently, the staff indicated in Section 7.2 of this report that any proposed changes to the appropriate SSAR sections constitutes an unreviewed safety question and, therefore, must be submitted to the NRC for review and approval prior to implementation.

Conclusions

On the basis of the above, the staff concludes that the top-level design processes, features and performance characteristics of the I&C aspects of SSCs important to safety in the ABWR are appropriately described in the design descriptions of the CDM, and are acceptable.

Further, these top-level design processes, features and performance characteristics can be adequately verified by the DAC provided by GE. Therefore, the staff concludes that the DAC in the CDM are necessary and sufficient to provide reasonable assurance that if the inspections, tests, and analyses are performed and the acceptance criteria met, the I&C aspects of SSCs important to safety in a facility that references the design have been designed, constructed and will operate in accordance with the design certification and applicable regulations.

14.3.3.5 Initial Test Program (ITP)

This section of the CDM consists of a high level commitment to an ITP and a description of the program and major program documents (i.e., a site-specific startup administrative manual, test specifications, and test procedures). The ABWR SSAR Chapter 14.2 contains a complete description of the ITP, and the staff's evaluation of the ITP is contained in Section 14.2 of this report.

The staff reviewed this CDM Section for consistency with the guidelines contained in the SRP and RG 1.68, "Initial Test Program for Water-Cooled Nuclear Power Plants." RG 1.68 describes the general scope and depth of testing that is acceptable to the staff for conduct of preoperational and startup testing as part of the ITP.

The key facets of the ITP are described in the Tier 1 CDM to ensure that subsequent changes in the conduct of the ITP cannot be initiated unilaterally by the COL applicant. This ITP is described in Tier 1 because of the essential role of a test program in the verification that SSCs have been constructed and will perform satisfactorily in service. The Tier 1 description requires that the ITP be performed under suitably controlled conditions and processes. The development of test procedures, conduct of the tests, and safe execution of the test program, are important

considerations in ensuring that as-built facility is in accordance with the design certification and applicable regulations. Thus, the staff will have the confidence that the ITP will be implemented effectively, so that the appropriate testing methodologies, and associated programmatic controls for testing plant systems will be ensured.

A corresponding ITAAC for this design description is not required for several reasons:

- (1) The Tier 1 certified design material consists of a high level commitment to an ITP, and a description of the program and major program documents that constitute an acceptable ITP (i.e., a site-specific startup administrative manual, test specifications, and test procedures). The specific testing necessary to verify design features and performance aspects of the design is delineated in the system-specific ITAAC.
- (2) The ITP covers a broader spectrum of time than the ITAAC. While ITP pre-operational testing shall be completed prior to fuel load, the ITP startup and power ascension testing will be conducted after fuel load. As the ITP involves testing post-fuel load, it is not appropriate to define associated ITAAC entries as Part 52 specifies that the ITAAC will be completed prior to fuel load.

In summary, the top-level ITP commitments in the CDM ensure that suitable controls are imposed over the pre-operational and start-up testing programs, which provide reasonable assurance that the facility can be operated without undue risk to the public. The staff concludes that the ITP information in the CDM is acceptable.

14.3.4 Interface Requirements

The requirements for interfaces for a design are contained in 10 CFR 52.47(a)(1)(vii-ix). An applicant for design certification is required to provide (1) the interface requirements to be met by those portions of the plant for which the application does not seek certification, (2) justification that compliance with the interface requirements is verifiable through inspection, testing, or analysis, and the method to be used for verification of interface requirements, and (3) a representative conceptual design for those portions of the plant for which the application does not seek certification. The staff evaluated these interface requirements and the ABWR design in the appropriate sections of this report.

GE defined the interface between the systems of the design and the site-specific systems to be at the walls of the turbine building, reactor building, and control building, as

Initial Test Program

depicted in Figure 1.2-1 of this report. This section of the CDM specifies interface requirements for those portions of the certified design that interface with site-specific portions of the design, and specifies the systems that are completely or partially out of scope of the certified design. The interface requirements define the design attributes and performance characteristics that must be met by the site-specific, out-of-scope portion of the plant so that the site-specific portion of the design is in conformance with the certified design. The site-specific portions of the design are those portions of the design that are dependent on characteristics of the site, such as the design of the ultimate heat sink.

The review of the appropriate inspections, tests, and analyses to demonstrate compliance with the interface requirements for the site-specific portion of the design is accomplished in the review of an application and for a combined license under Subpart C of 10 CFR Part 52.

GE provided information discussing the interface requirements in the CDM and in SSAR Section 14.3.4. GE provided acceptable interface requirements in CDM Section 4.0, and in the appropriate systems in CDM Section 2.0. This information was based on the information in the various sections of the SSAR, and is evaluated by the staff in the appropriate sections of this report. In CDM Section 4.0, GE stated that the development of ITAAC for the interface requirements will be similar in nature to the development of ITAAC in CDM Section 2.0. The staff concludes that this is an acceptable justification that compliance with the interfaces is verifiable through ITAAC, and the process described in SSAR Section 14.3 provides an acceptable methodology for verification of the interface requirements. GE provided acceptable representative conceptual designs in the SSAR that enabled the staff to complete its review of the design, as discussed in the appropriate sections of this report.

Therefore, based on the above discussion, the staff concludes that the interface information provided by GE meets the requirements contained in 10 CFR 52.47(a)(1)(vii - ix), and is acceptable.

14.3.5 Site Parameters

The requirements for site parameters for a design are contained in 10 CFR 52.47(a)(1)(iii). An applicant for design certification is required to provide the site parameters used in the design, and an analysis and evaluation of the design in terms of these parameters. The site parameters are specified in both the CDM Section 5.0 and Chapter 2 of the SSAR, and the analysis and evaluation of the design is contained in the various sections

of the SSAR. The staff evaluated these parameters and the design in the appropriate sections of this report.

Site parameters are specified in this section of the CDM for establishing the bounding parameters to be used in the selection of a suitable site for a facility referencing the ABWR certified design. Because they were used in bounding evaluations of the certified design, they define the requirements for the design that must be met by a site to ensure that a facility built on the site remains in conformance with the design certification. The demonstration that the site parameters are met at a given site is accomplished in conjunction with an application and issuance of a combined license under Subpart C of 10 CFR Part 52.

GE provided information discussing the site parameters in the CDM and in SSAR Section 14.3.5. GE provided acceptable site parameters postulated for the certified design in CDM Section 5.0 and in the appropriate sections of the SSAR. The appropriate sections of the SSAR information also provided an acceptable analysis and evaluation of the design in terms of these parameters, and the staff found the design acceptable in the related sections of this report. Therefore, the staff concludes that the site parameter information provided by GE meets the requirements of 10 CFR 52.47(a)(1)(iii), and is acceptable.

14.3.6 Summary

The staff reviewed the GE ABWR CDM and SSAR Section 14.3 in accordance with the requirements in Part 52 and the guidance provided in SRMs related to design certification applications provided by the Commission. This review included information contained in multiple submittals to the staff, as listed in the background portion of Section 14.3 of this report.

Based on the staff's review of the material in the CDM, and a review of the selection methodology and criteria for the development of the CDM contained in SSAR Section 14.3, the staff concludes that the top-level design features and performance characteristics of the ABWR SSCs important to safety are appropriately described in the CDM, and are acceptable.

Further, these top-level commitments can be adequately verified by the ITAAC and additional certified design material provided by GE. Therefore, in the appropriate parts of Section 14.3 of this report, the staff concludes that the CDM are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed, and the acceptance criteria met, a facility referencing the certified design will be constructed and will operate in conformity with the design certification and applicable regulations.

The staff also concludes in the appropriate parts of Section 14.3 of this report that the interface requirements and site parameters provided by GE for the ABWR meet the requirements for design certification applications in 10 CFR 52.47, and are acceptable.

14.3.7 DFSE Issues

14.3.7.1 Treatment of Non-Traditional Items in ITAAC

GE incorporated into the SSAR any insights into the design that were obtained from non-traditional items such as PRA and severe accident issue resolutions. Additionally, the staff followed the Commission's guidance in its review of the evolutionary designs for the resolution of non-traditional issues such as those discussed in SECY-90-016, "Evolutionary Light Water Reactor Certification Issues and Their Relationships to Current Regulatory Requirements," and SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs." In 10 CFR 52.47, the Commission specified that the ITAAC must provide reasonable assurance that "a plant which references the design is built and will operate in accordance with the design certification." Therefore, by verifying key aspects and features of the design, the ITAAC implicitly confirm the implementation of these non-traditional items and the safety findings contained in the safety evaluation report.

The staff requested GE to develop a cross-reference of key aspects, analyses, and features of the design from the SSAR to the CDM in order to document how these issues had been incorporated into the CDM. Specifically, the cross-references were to show how key aspects of the accident analyses, PRA, and severe accident issue resolutions were included in the CDM. This was DFSE Open Item 14.1.1.5.2-1.

GE provided the cross-references in SSAR Amendment 33 and updated the cross-references in Amendment 34. In those submittals, GE provided cross-references for key safety and integrated plant analyses in SSAR Section 14.3, and cross-references for PRA and severe accident analyses in SSAR Section 19.8. GE also provided more detailed cross-references for these analyses in a letter dated March 31, 1994.

The treatment of non-traditional items in the CDM for the ABWR design is discussed further in Section 14.3.2 of this report. Based on the discussion in Section 14.3.2, the staff found the cross-references, and the treatment of non-traditional items in the CDM acceptable. This resolved DFSE Open Item 14.1.1.5.2-1.

14.3.7.2 Relationship of the Design Description to the ITAAC

GE proposed that certain systems could have Tier 1 design descriptions, but may not require any corresponding ITAAC to verify the design for those systems. Examples of these systems were the fuel service equipment, the internal pump maintenance facility, and the fuel cask cleaning facility. The staff was reviewing this proposal at the time the DFSE was issued. This was DFSE Open Item 14.1.1.5.3-1.

GE adopted a graded approach to the level of detail in the development of the CDM, based on the safety significance of the ABWR structures, systems, and components. In SSAR Section 14.3, GE provided the process it used in the development of the CDM, based on the design presented in the appropriate sections of the SSAR. GE applied various selection criteria to the information in multiple chapters of the SSAR to determine the level of design information for a given system in the CDM.

GE provided its selection criteria and methodology for the ITAAC in SSAR Section 14.3.2.2. In general, each ABWR system with information in the design description has one or more ITAAC, based on the its safety significance. A single ITAAC may verify one or more provisions in the design description. Other aspects of systems may be satisfied by ITAAC contained in other systems or other sections of the CDM. For example, the piping design information in CDM Section 3.3 provides acceptance criteria for seismic Category I and ASME Code Class 1, 2, and 3 components. Additionally, since Part 52 requires that the ITAAC be satisfied prior to fuel loading, there are no ITAAC to verify any information dependent on post-fuel load conditions (e.g., nuclear fuel, fuel channels, and control rods). This information will be verified by the ITP as part of start-up and power ascension testing.

In Section 14.3 of this report, "Basis for Approval of the CDM," the staff discussed its graded approach to the review of the CDM based on the safety significance of the SSCs. Thus, consistent with the guidance of Part 52 and the SRM related to SECY-90-377, the staff recognized that although many aspects of the design were important to safety, the level of design detail in the CDM and verification of the key features and performance characteristics should be commensurate with the importance of the safety functions to be performed.

The relationship of design descriptions and ITAAC for the ABWR design is discussed further in Section 14.3.2 of this report. Based on the discussion in Section 14.3.2, the staff found the treatment of design descriptions and ITAAC in

Initial Test Program

the CDM acceptable. This resolved DFSER Open Item 14.1.1.5.3-1.

14.3.7.3 System Design Descriptions and ITAAC

Section 2.0, "Tier 1 Material for ABWR Systems," of the "Tier 1 Design Certification Material for the GE ABWR," was under staff review at the time the DFSER was issued. The staff stated that the evaluation would be provided in the FSER. This was DFSER Open Item 14.1.2-1.

GE provided revised CDM information and supporting information in various SSAR submittals to the staff as discussed in the background part of this report. In these submittals, GE addressed all of the comments of the staff, the ACRS, and an independent review group. As discussed in Section 14.3 of this report, "Basis for Approval of the CDM," the staff reviewed the GE ABWR CDM and selection, methodology and criteria for the development of the CDM contained in SSAR Section 14.3 in accordance with the requirements in Part 52 and the guidance provided in SRMs related to design certification applications provided by the Commission.

System design descriptions and ITAAC for the ABWR design are discussed further in Section 14.3.2 of this report. Based on the discussion in Section 14.3.2, the staff found the treatment of system design descriptions and ITAAC in the CDM acceptable. This resolved DFSER Open Item 14.1.2-1.

14.3.7.4 Equipment Qualification (EQ)

Section 3.1, "Equipment Qualification," of the "Tier 1 Design Certification Material for the GE ABWR," was under staff review at the time the DFSER was issued, and the staff stated that the evaluation would be provided in the FSER. This was DFSER Open Item 14.1.3.1-1.

GE provided revised CDM information in a submittal dated August 31, 1993, and supporting information in Amendment 32. GE eliminated the proposed CDM Section 3.1, and put the required equipment qualification information in CDM Section 1.2, General Provisions. GE provided supporting information on equipment qualification in the SSAR. The equipment qualification of safety-significant portions of the design will be verified as part of the basic configuration walkdown of individual SSCs.

The basic configuration walkdown for equipment qualification is discussed in greater detail in Sections 14.3.1.2 and 14.3.2.3.1 of this report. Based on the discussion in Sections 14.3.1.2, and 14.3.2.3.1 the staff found the treatment of equipment qualification in the

CDM acceptable. This resolved DFSER Open Item 14.1.3.1-1.

14.3.7.5 Reliability Assurance Program

Section 3.8, "Reliability Assurance Program," of the "Tier 1 Design Certification Material for the GE ABWR," was under staff review at the time the DFSER was issued. The staff stated that the evaluation would be provided in the FSER. This was DFSER Open Item 14.1.3.8-1.

When the DFSER was written the staff's position on a reliability assurance program (RAP) was that a high-level commitment to a RAP applicable to design certification (D-RAP) was required. GE committed to provide the required D-RAP commitments in the CDM in a letter dated July 12, 1994. The staff finds GE's commitments to D-RAP acceptable, subject to incorporation in the CDM. DFSER Open Item 14.1.3.8-1 becomes Confirmatory Item F14.3.7.5-1.

14.3.7.6 Welding

Section 3.9, "Welding," of the "Tier 1 Design Certification Material for the GE ABWR," was under staff review at the time the DFSER was issued. The staff stated that the evaluation would be provided in the FSER. This was DFSER Open Item 14.1.3.9-1.

GE provided revised CDM information in a submittal dated August 31, 1993, and supporting information in Amendment 32. GE eliminated the proposed CDM Section 3.9, and put the required welding information in CDM Section 1.2, General Provisions. GE provided supporting information on welding in the SSAR. The welding aspects of safety-significant portions of the design will be verified as part of the basic configuration walkdown of individual SSCs.

The basic configuration walkdown for welding is discussed in greater detail in Section 14.3.1.2 of this report. Based on the discussion in Section 14.3.1.2, the staff found the treatment of welding in the CDM acceptable. This resolved DFSER Open Item 14.1.3.9-1.

14.3.7.7 Interface Requirements

Section 4.0, "Interface Tier 1 Material," of the "Tier 1 Design Certification Material for the GE ABWR," was under staff review at the time the DFSER was issued. The staff stated that the evaluation would be provided in the FSER. This was DFSER Open Item 14.1.4-1.

The staff provided comments on interfaces during a meeting with GE on July 27, through 29, 1993. These comments are documented in a meeting summary dated August 10, 1993. GE provided information to resolve these comments in revised Tier 1 material dated August 31, 1993, and provided supporting information in SSAR Amendment 32. In those submittals, GE modified the CDM Section 4.0 to provide consistent treatment of interface requirements for systems, clarify the in-scope and out-of-scope portions of the design, and modify the SSAR listing of systems so that it was consistent with the CDM.

Interfaces are discussed further in Section 14.3.4 of this report. Based on the discussion in Section 14.3.4, the staff found the treatment of interface requirements in the CDM acceptable. This resolved DFSER Open Item 14.1.4-1.

14.3.7.8 Site Parameters

Section 5.0, "Site Parameters," of the "Tier 1 Design Certification Material for the GE ABWR," was under staff

review at the time the DFSER was issued. The staff stated that the evaluation would be provided in the FSER. This was DFSER Open Item 14.1.5-1.

The staff provided comments on site parameters in a letter to GE dated July 9, 1993. These comments were discussed during a meeting with GE on July 27 through 29, 1993, and were documented in a meeting summary dated August 10, 1993. GE provided information to resolve these comments in revised Tier 1 material dated August 31, 1993, and provided supporting information in SSAR Amendment 32. In those submittals, GE modified the CDM Section 5.0 to provide consistent treatment of site parameters with the site parameters and design basis analyses contained in the SSAR.

Site parameters are discussed further in Section 14.3.5 of this report. Based on the discussion in Section 14.3.5, the staff found the treatment of site parameters in the CDM acceptable. This resolved DFSER Open Item 14.1.5-1.

15 TRANSIENT AND ACCIDENT ANALYSIS

The staff reviewed the transient and accident analysis for the advanced boiling water reactor (ABWR) in accordance with Standard Review Plan (SRP) Section 15. Conformance with the acceptance criteria, except as noted for each of the sections, formed the basis for deciding if the design of the facility for each of the areas reviewed was acceptable.

Three groups of design-basis events are evaluated in this section: anticipated operational occurrences, anticipated operational occurrences involving common-mode software failure, and accidents. A conservative model of the reactor is used for the analysis of events in each group and all appropriate systems whose operations (or postulated misoperations) would affect the event are included. Anticipated operational occurrences (AOOs) are expected to occur during the life of the plant and are analyzed to ensure that they will not cause damage to either the fuel or to the reactor coolant pressure boundary. AOOs involving common-mode software failures have a lower probability of occurrence and are discussed in Section 15.2 of this report. Design-basis accidents (DBAs) are not expected to occur but are postulated because their consequences would include the potential for the release of significant amounts of radioactive material. These postulated accidents are analyzed to determine the extent of fuel damage expected and to ensure that reactor coolant pressure boundary damage, beyond that assumed initially to be the DBA, will not occur and that the radiological dose is maintained within 10 CFR Part 100 guidelines.

For loss-of-coolant accidents (LOCAs), the acceptance criteria for the emergency core cooling system (ECCS) specified in 10 CFR 50.46 are as follows:

- The peak cladding temperature must remain below 1204 °C (2200 °F).
- Maximum cladding oxidation must nowhere exceed 17 percent of the total cladding thickness before oxidation.
- Total hydrogen generation must not exceed 1 percent of the hypothetical amount that would be generated if all the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.
- The core must be maintained in a coolable geometry.
- Calculated core temperatures after successful initial operation of the ECCS must be maintained acceptably low, and decay heat must be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

The staff evaluation of GE Nuclear Energy (GE) LOCA analysis is given in Sections 15.4.4 and 6.3 of this report.

To demonstrate the adequacy of the plant's engineered safety features (ESFs), GE calculated the offsite consequences that could result from the occurrence of each of several DBAs and presented the results of these computations in the standard safety analysis report (SSAR).

15.1 Anticipated Operational Occurrences

AOOs which include infrequent and moderate frequency events are those transients expected to occur during normal or planned modes of plant operation. The acceptance criteria for these transients are based on GDC 10, 15, and 20. GDC 10 specifies that the reactor core and associated control and instrumentation systems shall be designed with appropriate margin to ensure that specified acceptable fuel-design limits are not exceeded during any condition of normal operation, including the effects of AOOs. GDC 15 specifies that sufficient margin shall be included to ensure that design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including AOOs. GDC 20 specifies that a protection system be provided that automatically initiates appropriate systems to ensure specified acceptable fuel design limits are not exceeded during any condition of normal operation including AOOs.

Acceptance criteria contained in Chapter 15 of the SRP for AOOs are as follows:

- Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design values according to American Society of Mechanical Engineers (ASME) Code, Section III, Article NB-7000. For ABWR, which has a design pressure of 8722 kPa (1250 psig), the pressure should not exceed 9584 kPa (1375 psig) during any AOO.
- Fuel-cladding integrity should be maintained by ensuring that the reactor core is designed with appropriate margin during any conditions of normal operation, including the effects of AOOs. For BWRs, the minimum value of the critical power ratio reached during the transient should be such that 99.9 percent of the fuel rods in the core would not be expected to experience boiling transition during core-wide transients. This limiting value of the minimum critical power ratio (MCPR), called the safety limit for the ABWR, is 1.07.
- An incident that occurs with moderate frequency should not generate a more serious plant condition unless other faults occur independently.

Transient and Accident Analysis

- An incident that occurs with moderate frequency in combination with any single active component failure, or operator error, should not result in loss of function of any barrier other than the fuel cladding. A limited number of fuel-rod-cladding perforations is acceptable. (See the discussion of Three Mile Island (TMI) Item II.K-3.44, NUREG-0737, in Chapter 20 of this report.)

For ABWR transient analysis, GE used the ODYNA computer code (proprietary) to simulate pressurization events and the REDYA computer code (proprietary) for other transient events.

The ODYNA code (proprietary) is designed to simulate selected transient conditions for the ABWR. Major features of this code are one dimensional description of kinetics and thermal-hydraulics of the core; reactor internal pumps (RIPs); option for two- and eight-node steamline model; and system models consisting of the core, bypass, and upper plenum. The core thermal-hydraulics is calculated using a five-equation formulation consisting of mass and energy balances for the vapor and liquid phases and momentum balance for the mixture. The code also contains models for ECCS, boron injection, and detailed control system for turbine control and pressure relief. The code includes modifications to the upper plenum model to allow for subcooled liquid.

The REDYA code (proprietary) is designed to simulate selected transient conditions for the ABWR. Major features of this code are point kinetics description of the core; lumped pressure calculation from a volume consisting of the core, bypass, and upper plenum; and multiple RIPs. Additionally, the code has models for internal separators; options for two- and eight-node steamline models; wide range of turbine control, pressure relief, and rod motion; reactor protection system (RPS) options; and boron injection models.

Both ODYNA and REDYA codes are similar to previous models, ODYN (NEDO-24154) and REDY (NEDO-10802), which the staff approved for transient analysis of operating BWRs. The ODYNA and REDYA versions have been revised to reflect the recirculation model and ECCS unique for the ABWR. The REDYA code also incorporated some model improvements already included in the ODYN code (e.g., safety/relief valve model and steamline model).

The recirculation model was the major modification in both ODYNA and REDYA codes. This new recirculation model consists of three groups of RIPs. RIP characteristics, initial conditions, and pump trip and runback functions are assumed to be identical for all RIPs

in the same group. The performance of this recirculation model has been qualified against plant startup data obtained from European plants with similar RIPs. The events simulated for qualification purposes included pump coastdown during a trip of all pumps (data from two plants), during a trip of one pump, and during momentary voltage drop. GE compared the calculated results and plant data to verify the adequacy of the codes. GE also conducted model-to-model comparison and comparison with other test data (e.g., Peach Bottom turbine trip test) to ensure that the models perform correctly.

The staff, with the technical assistance of Brookhaven National Laboratory, audited ODYNA and REDYA during January 1992 at GE offices in San Jose, California. The staff audited three major areas: (1) formulation and models, (2) quality assurance (QA) procedures, and (3) verification and validation. The staff concluded that the modifications to the ODYN and REDY codes for the ABWR were adequately justified. It found the changes to ODYN and REDY to be acceptable. However, there was no documentation to verify that the coding changes to implement new models were independently checked. As stated in the DFSER, GE was to inform the staff, in writing, to confirm that the implementation of the code modifications had been independently verified as correct. This was DFSER Confirmatory Item 15.1-1. (Previously DSER SECY-91-355 Open Item 134.)

In the letter dated February 26, 1993, GE submitted the necessary information describing its verification process. The staff reviewed the documents governing GE QA requirements, and found that the code modifications were performed in accordance with the applicable design control provisions contained within the GE QA program and the GE design control procedures. Therefore, the staff has concluded that GE properly controlled the coding changes for ODYNA and REDYA. Thus, DFSER Confirmatory Item 15.1-1 is resolved.

The staff concludes that the ODYNA and REDYA computer codes are acceptable for design analysis of the ABWR.

In conducting its analyses of anticipated transients, GE used conservative assumptions with regard to initial power, scram reactivity, reactivity coefficients, and power profiles. It used conservative time delays to trip for each scram signal in the analyses. GE analyzed the following transients:

- decrease in core coolant temperature
- increase in reactor pressure
- decrease in reactor coolant system flow rate

- reactivity and power distribution anomalies
- increase in reactor coolant inventory
- decrease in reactor coolant inventory
- anticipated transient without scram (ATWS) (See Section 15.5 of this report.)

(1) Decrease in Core Coolant Temperature

Transients analyzed in this group included loss of feedwater heaters, feedwater control failure, runout of one feedwater pump, feedwater controller failure during maximum demand, opening of turbine control and bypass valves, pressure regulator failure in the open direction, inadvertent opening of a safety/relief valve, and inadvertent residual heat removal (RHR) shutdown cooling operation.

For transients categorized under Decrease in Core Coolant Temperature, the most severe transient is feedwater controller failure during maximum demand (runout of two feedwater pumps). The resultant minimum critical-power ratio (MCPR) reached is 1.07, and the peak vessel pressure is 1262 kPa (168 psi) below the ASME Code limit.

For the loss of feedwater heating transient, GE assumed a drop of 38 °C (68 °F) in feedwater temperature, although a drop of 66 °C (119 °F) occurred at a domestic BWR following an electrical component failure. In a letter dated January 10, 1992, GE stated that the drop of 66 °C (119 °F) was a unique condition for that particular BWR design. The feedwater temperature will not drop as far in the ABWR design during this transient because the ABWR is designed so that no single failure of equipment can cause a temperature drop of more than 38 °C (68 °F). The staff agrees with GE that the 66 °C (119 °F) drop was caused by a plant-specific design feature not present in the ABWR. Therefore, the 38 °C (68 °F) drop analyses is acceptable.

Inadvertent opening of a safety relief valve will cause a decrease in reactor coolant inventory and result in a mild depressurization event that will have only a slight effect on fuel thermal margins. Changes in surface heat flux are calculated to be negligible indicating an insignificant change in the MCPR. Thus, the transient response is acceptable and is bounded by the more severe feedwater controller failure transient.

GE initially inappropriately categorized the inadvertent RHR shutdown cooling operation event as an accident rather than an AOO. This was

identified as DSER (SECY-91-355) Open Item 135, which was a significant deviation from the SRP. GE recategorized this event as a moderate frequency event (an AOO) and applied the appropriate acceptance criteria in the SSAR. The reanalysis of the event shows that the AOO acceptance criteria are met. This is acceptable and resolved DSER SECY-91-355 Open Item 135.

(2) Increase in Reactor Pressure

Transients in this group included generator load rejection and turbine trip with and without turbine bypass, inadvertent main steam isolation valves (MSIV) closure, loss of condenser vacuum, loss of auxiliary power transformer, loss of all grid connections, loss of all feedwater flow, and failure of RHR shutdown cooling.

The transient resulting in the highest system pressure was a generator load rejection without turbine bypass, which would result in a peak system pressure of about 1138 kPa (151 psi) below the allowable maximum pressure of 9584 kPa (1375 psig). In the overpressure protection report, the most limiting transient is an inadvertent MSIV closure with failure of the position switch scram (see Section 5.2.2 of this report). The reactor pressure increase for the transient turbine trip without bypass is bounded by the MSIV closure transient and is acceptable.

(3) Decrease in Reactor Coolant System Flow Rate

Transients in this group included trip of three RIPs, trip of all RIPs, runback of RIPs, and failure of the recirculation flow control to decrease flow.

Traditionally, loss of all forced circulation has been classified as a moderate frequency event and subject to the associated acceptance criteria. The ABWR will use motor-generator (MG) sets to power six of the ten RIPs. On loss of offsite power, the inertia of the MG provides a longer flow coastdown period. Analysis of this event shows that AOO acceptance criteria are met. However, the staff asked GE to consider the possibility of failures of other systems or components that could result in the loss of forced circulation without the availability of MG set coastdown. GE identified loss-of-flow transients both with and without offsite power available. The case with offsite power available was found more limiting because it did not result in an immediate reactor scram.

Transient and Accident Analysis

The most severe transient in this group is the simultaneous trip of all RIPs with offsite power available which is discussed in Section 15.2 of this report.

(4) Reactivity and Power Distribution Anomalies

Transients in this group included rod withdrawal error, abnormal startup of one RIP, fast runout of RIPs, and control rod misoperations.

The startup of an idle RIP is categorized under reactivity anomalies. This event is not a limiting transient and neither primary pressure boundary nor fuel damage criteria are exceeded.

(a) Rod Withdrawal Error at Low Power

GE examined the design of the rod control system to ascertain if a single failure can lead to the uncontrolled withdrawal of a control rod during refueling and during startup and low-power operation. During refueling operations, interlocks ensure that all control rods are inserted while fuel is being handled over the core. When no fuel is being handled, a maximum of one rod may be withdrawn. However, the control system is designed (see Section 4.3.2 of this report) so that the core is subcritical with the highest worth rod withdrawn. Finally, a control rod cannot physically be removed (from the top) without removing the four fuel assemblies that surround the rod. Therefore, GE has not provided an analysis of control rod removal error during refueling. This is in accord with approvals for current BWRs and is acceptable.

GE claims that the uncontrolled withdrawal of a rod during reactor startup is prevented by the rod block control system function of the rod control and information system (RCIS). This system enforces the banked position withdrawal sequence and precludes rod withdrawals other than those permitted in normal operation SRP Section 15.4.1 states that this transient need not be considered if single failures cannot cause the sequence. The single failure evaluation of the RCIS is given in Chapter 7 of this report. However, in accordance with staff requirements on current BWRs, GE also analyzed the erroneous withdrawal of a high worth control rod and found that the results fall well within the MCPR and other fuel criteria limits. This analysis is acceptable for the postulated event.

(b) Rod Withdrawal Error at Power

The rod withdrawal error transient can result from either a procedural error by the operator so that a gang of control rods is withdrawn continuously or from a malfunction of the automated rod withdrawal control logic during automated operation in which a gang of control rods is withdrawn continuously.

In the ABWR, the automated thermal limit monitor (ATLM) and the multi-channel rod block monitor (RBM) subsystem logic issues a rod block signal used in the RCIS logic to enforce rod blocks. This feature acts to prevent fuel damage by ensuring that the MCPR and maximum average planar linear heat generation rate (MAPLHGR) do not violate the fuel thermal operating and safety limits. The operating thermal limits rod block function will block rod withdrawal when the operating thermal limit is reached.

The rod block algorithms and set point are based on online core information (e.g., core flow and local power range monitor (LPRM) readings used to calculate the fuel status relative to limits). In response to staff questions and DSER (SECY-91-135) Open Item 137, GE presented (enclosures to its letter dated January 10, 1992, "GE Response to Agenda Item 12 Discussed During . . . Meeting on November 20 and 21, 1991") a description of the algorithms, their development and bases, and examples of the functions (correlating MCPR or MAPLHGR with LPRM set points) programmed into the microprocessor-based ATLM for a given fuel and core design.

The functions for both MCPR and MAPLHGR are developed statistically with a database developed by analyzing control rod withdrawals for a wide range of initial power and flow conditions for a core design. These cases are used to provide a statistically derived bounding (95/95 probability and confidence) function for the rod block set point. The method is similar to the approach previously approved for BWR/5 rod block set point analysis. The functions will be updated for each fuel and core design used in the ABWR. This resolved DSER SECY-91-355 Open Item 137.

The RBM algorithms are simple relationships between power (LPRM readings) and operating limits (MCPR and MAPLHGR) and are reasonable and acceptable. The statistical approach is similar to previous staff-approved rod block methodology

and is appropriate and conservative for use as proposed for the ABWR.

(5) Increase in Reactor Coolant Inventory

GE analyzed inadvertent startup of the high-pressure core flooders (HPCF) pump. (Feedwater flow control failure to maximum demand is covered in Category (1).)

The transient which could cause unplanned addition to coolant inventory is the inadvertent actuation of the HPCF system. The HPCF system actuation has little effect because its flow is small compared to the recirculation flow. Because the HPCF full flow is a small contributor to total core flow, the increase in total coolant inventory is also small. GE's analysis shows that the consequences of this small inventory increase has little effect on fuel thermal margins and reactor system pressure. In accordance with SRP Section 15.5.1, this is acceptable.

(6) Decrease in Reactor Coolant Inventory

The anticipated operational occurrence of the inadvertent opening of a safety relief valve is covered in Category (1) above.

GE indicated that non-safety-grade equipment is credited for the high water level 8 trip, use of turbine bypass valves, and recirculation pump trip on load/turbine trip.

The staff questioned the appropriateness of GE taking credit for equipment that is not safety grade in the transient analysis as stated in DSER (SECY-91-355) Open Item 138. GDC 1 through 4 require that components important to safety be designed to be commensurate with the quality standards, and GDC 21 requires that the protection system be designed for high functional reliability. GE listed in a table the redundancy, isolation, environmental, seismic, periodic testing, and QA requirements for the equipment.

Even though the equipment discussed above will not be categorized as safety grade, it is of high quality and has sufficient redundancy to ensure its operability. To ensure an acceptable level of performance for the ABWR, GE committed to identify the above equipment in the ABWR technical specifications (TS) with regard to availability, set points and surveillance testing. This was DSER TS Item 15.1-1. GE included the level 8

trip, the RIP trip and the turbine bypass in the proposed ABWR TSs. This is acceptable and TS Item 15.1-1 is resolved.

By letter dated August 23, 1989, GE informed Gulf States Utilities Company of a condition that could be reportable under 10 CFR Part 21, applicable to the River Bend Station. This condition involves a slow closure of one main turbine control valve. This low probability event, which was not previously considered, results from a turbine control valve that GE assumes to close as a result of an unspecified failure in the turbine control circuit or in the servo-mechanism hardware. According to GE, if the valve closes in less than 2.3 seconds, a reactor scram will be initiated as a result of high neutron flux and no safety limits will be exceeded. However, if the valve closes in more than 2.3 seconds, the reactor scram will be initiated by high reactor pressure. During this slow-closure case, the MCPR safety limit may be exceeded if the maximum combined flow limiter is set for less than 113 percent of rated steam flow. GE based the consequences of this postulated event on its assessment of a generic BWR/6 analysis. This was identified as DSER (SECY-91-355) Open Item 139. The staff, however, requested that GE address the event for ABWR applicability. In response, GE performed an ABWR-specific analysis for the slow closure of one turbine control valve with the remaining three control valves remaining open. In this case, the neutron flux increase will not reach the high neutron flux scram set point. Since the available turbine bypass capacity will be high enough to bypass all steam flow not passing through the remaining three turbine control valves, the reactor power settles back to its steady state. (The total steam flow through three control valves will increase to about 85 percent, and the remaining 15 percent of flow will pass through the slow-opening control valve and the bypass valves.) During this transient, the peak fuel surface heat flux will not exceed 104 percent of its initial value. The MCPR remains above the safety limit and is acceptable. Therefore, DSER Open Item 139 was resolved.

15.2 Trip of All Reactor Internal Pumps and Pressure Regulator Down-Scale Failure

For the postulated trip of all of the RIPs with offsite power available, GE postulated a common-mode failure of the adjustable speed drives. GE estimated that a fraction of low burnup fuel rods will achieve boiling transition during this event although test results indicate no fuel failures

would occur. The staff classified this postulated event in the special category of anticipated transients involving a common-mode software failure and established a special acceptance criterion for the radiological dose calculation. The staff will not require that fuel failure be assumed in dose calculations for fuel rods that are under approximately 600 °C (1111 °F) for less than 60 seconds. This time and temperature criterion is based on test data for fuel that has achieved up to 20 gigawatts days per metric ton (GWD/MtU) (18 GWD/t) burnup; thus, it may be applied only to fuel with burnup of less than 20 GWD/MtU (18 GWD/t). For fuel with greater burnup, the dose calculations must assume fuel failure for all fuel rods that achieve transition boiling. In the equilibrium cycle, the higher burnup fuel accounts for about 45 percent of the total fuel bundles. The power generated by these bundles is usually 20 percent less than that of the hottest bundles, and less than 0.2 percent of these rods are expected to enter transition boiling. Because none of the hottest fuel rods exceed the time and temperature failure criterion, the radiological dose requirements limit of 10 percent of 10 CFR Part 100 are satisfied.

For the pressure regulator down-scale failure to occur, all three channels would have to suffer a common-mode failure before the pressure regulator would go either up or down the scale. If the pressure regulator failed down-scale, the steam control valves would close causing the reactor pressure and reactivity to increase. When reanalyzing this postulated event, GE proposed to assume that any fuel rods that achieve transition boiling fail for the purposes of the radiological dose calculation. The staff includes this postulated event in the special category of anticipated transients involving a common-mode software failure because of the uncertainty that such an event will occur during the plant lifetime. GE originally categorized this event as an accident. The staff believes that it is more appropriate to apply a special classification for such an event. The staff required that GE demonstrate that this special event will not exceed the limits of 10 percent of 10 CFR Part 100, which the staff considers appropriate for an event of such postulated frequency.

According to GE analysis, during this event, it is estimated that less than 0.2 percent of fuel rods enter transition boiling and the requirement that the limit of 10 percent of 10 CFR Part 100 not be exceeded is met.

The staff will treat the above two postulated events as special cases, applicable only for the ABWR. This is due to the unique design features of the ABWR instrumentation and control systems, which reduce the frequency of such events; therefore, allowing these events to be recategorized as special cases. This resolved DSER (SECY-91-355) Open Item 136.

15.3 Accidents

GE analyzed RIP seizure and shaft break accidents. In the unlikely event of the pump motor shaft of 1 of the 10 RIPs stops instantaneously, a very rapid decrease of pump flow will result from the large hydraulic resistance introduced by the stopped rotor or shaft and cause pump seizure or shaft break. Consequently, core inlet flow and core cooling capability decreases. However, GE's analysis shows that with only 1 out of 10 RIPs seized, the core flow decrease is small (< 10 percent), so the event is mild. The RIP seizure and shaft break do not result in any fuel failure. This satisfies the dose limit criteria of 10 CFR Part 100 and is acceptable.

GE's analyses of the mislocated fuel bundle accident, misoriented fuel bundle accident, rod ejection accident, and control rod drop accident are discussed below.

(1) Mislocated Fuel Bundle Accident

Three errors must occur for this event to take place: (a) a bundle must be misloaded into a wrong location in the core; (b) the bundle, which was supposed to be loaded where the mislocation occurred, also is put in an incorrect location or discharged; and (c) the misplaced bundles are overlooked during the core verification process after core loading. A fuel loading error not detected by in-core instruments after fueling operations may result in an undetected reduction in thermal margin during power operations. However, GE evaluated the probability and consequences of a misplaced fuel bundle accident in equilibrium, first cycle, and subsequent cycle cores for current operating reactors and concluded that no fuel failure will occur and no radioactive material will be released from the fuel. The staff approved this analysis for operating plants and it is also applicable for the ABWR. This satisfies the criteria of 10 CFR Part 100 as required by the SRP for this event and is acceptable.

(2) Misoriented Fuel Bundle Accident

GE notified the staff by a 10 CFR Part 21 report (GE letter dated June 19, 1992, from S.J. Stark, "10 CFR Part 21, Reportable Condition, Rotated C or S-Lattice Fuel Assembly") that a fuel misorientation event may lead to fuel damage in BWR/6 designs. The staff required GE to evaluate the applicability of the issue for the ABWR and discuss its evaluation in the SSAR. This was DSER Open Item 15.3-1.

For this event the fuel bundle is located correctly, but is rotated by 90 ° or 180 °, resulting in non-uniform adjacent water gaps. While this does not result in exceeding specified acceptable fuel design limits for some fuel bundle configurations during normal operation, for other configurations the limit might be exceeded. GE prepared a generic probabilistic risk model (GE letter dated September 30, 1992, from J.F. Klapproth, "Rotated Bundle Event Licensing Basis Change"), based on experience from recommended procedures for verifying fuel loading instituted in 1981, this model applies to current operating reactors and the ABWR, to demonstrate that the event may be classified as an accident. Since the staff had not completed the review of the GE probabilistic study, GE chose to reanalyze the misoriented fuel bundle event for a core loading with a bundle very similar to the reference fuel bundle design. GE made only slight modifications to the radial enrichment distribution to reduce the delta R factor. The analysis reflected that the energy capabilities were equivalent to the reference bundle design and the 15 percent thermal margin requirement was maintained. The infinite lattice void coefficient was also unchanged. This assures that the fuel safety limits are not exceeded for a misorientation event and is acceptable to the staff. Therefore, DFSE Open Item 15.3-1 is resolved.

(3) Rod Ejection Accident

The rod ejection accident is caused by a major break on the fine motion control rod drive (FMC RD) housing, outer tube, or associated control rod drive (CRD) pipe lines. The consequences of a rod ejection accident are similar to those of the rod drop accident in that the fuel enthalpy criteria may be violated if the speed of the ejected rod and/or the reactivity added are large enough. The same criterion of 280 cal/gm (1172 E+3 joules/kg) is applied to the rod ejection accident.

A redundant brake mechanism is installed in the FMC RD system (two brakes in each FMC RD) to prevent severe consequences resulting from this accident. Even if this accident does happen, the brake effectively terminates this event and prevents any severe consequences.

The radiological consequences are bounded by the analysis of the control rod drop accident. Therefore, the plant response is acceptable.

(4) Rod Drop Accident

The locking piston CRD mechanism in current BWRs cannot detect separation of the control rod from the drive mechanism during normal rod movements. Therefore, a latch mechanism is provided on the control rod to restrict the control rod free-fall distance to acceptable limits to prevent damage to the nuclear system process barrier by the rapid reactivity increase that would result from a free fall of a control rod (rod drop accident) from its fully inserted position to the position where the drive mechanism is withdrawn.

In contrast to the locking piston CRD, the FMC RD is designed to detect the separation of the control rod from the drive mechanism. Two redundant and separate Class 1E switches are provided to detect the separation of either the control rod from the hollow piston or the hollow piston from the ball nut. Actuation of either of these switches cause an immediate rod block and initiate an alarm in the control room, thereby reducing the probability of a rod drop accident from occurring. The radiological consequences of the control rod drop accident are provided in Section 15.4.1 of this report.

15.4 Radiological Consequences of Accidents

(Throughout this chapter there are statements indicating that 10 CFR Part 100 guidelines and criteria have been met. This means that the design basis accident being evaluated would result in a fission product release that would result in an exposure to an individual located in the plant's exclusion area for a period of two hours immediately after the release, of no more than 250 mSv (25 rem), or a whole body dose of no more than 3000 mSv (300 rem), or a total thyroid exposure from iodine of no more than 3000 mSv (300 rem). Further, a person located in the plant's low population zone for the duration of the release would receive no more than a whole body dose of 250 mSv (25 rem) or a total iodine dose to the thyroid of 3000 mSv (300 rem). These dose limits are listed in 10 CFR 100.11.)

In Chapter 15 of the SSAR, GE assessed the radiological consequence of the following six reactor DBAs (1) failure of small lines carrying primary coolant outside containment, (2) main steamline failure outside containment, (3) LOCA, (4) fuel handling accident, (5) spent fuel cask drop accident, and (6) reactor water cleanup system failure outside containment. GE concluded that the ABWR design using (1) reactor accident source term assumptions given in Technical Information Document (TID)-14844, "Calculation of Distance Fac-

Transient and Accident Analysis

tors for Power and Test Reactor Sites," (2) certain ESF systems in the ABWR design, and (3) the bounding sets of atmospheric relative concentration values (χ/Q) which determine certain distances to the exclusion area boundary (EAB) and the low-population zone (LPZ) for a given site, are sufficient to provide reasonable assurance that the radiological consequences of such DBAs will be within the dose reference values established in 10 CFR Part 100 and the dose limits given in GDC 19 of Appendix A to 10 CFR Part 50. (A maximum whole body dose of 50 mSv (5 rem) or its equivalent to any part of the body for the duration of an accident).

To verify GE's conclusion, the staff independently assessed the radiological consequences of the above six DBAs and reactor control rod drop accident. In its assessments, the staff used assumptions and methods described in the SRP Section 15 and in Regulatory Guide (RG) 1.3 (Rev. 2), "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Boiling Water Reactors," and the bounding atmospheric relative concentration values for EAB and LPZ proposed by GE. The major parameters and assumptions used in the staff's radiological consequence assessments are listed in Tables 15.2 through 15.9 of this report.

In its evaluation of the radiological consequences in the DFSE, the staff used an EAB of 800 m (0.5 mi) and an LPZ ranging from 1500 (0.9 mi) to 4800 m (3 mi), with Pasquill F stability and persistent (greater than 95 percent of time) and a wind velocity of 1 m/sec (3 ft/sec). The staff also stated that the median values of EAB and LPZ of current operating reactor sites are 800 m (0.5 mi) and 4800 m (3 mi), respectively. In the DFSE, the staff accepted and used, in its radiological consequence assessments, the atmospheric relative concentration values corresponding to these distances as proposed by GE. GE calculated these values using the acceptable regulatory methodology given in RG 1.3.

Subsequently, as agreed to by the staff, GE proposed in Amendment 32 to the SSAR the use of the bounding atmospheric relative concentrations for EAB and LPZ for the ABWR design rather than median EAB and LPZ distances used by the staff in the DFSE. Table 15.5 of this report and in Table 15.6-13 of the SSAR list the bounding atmospheric relative concentration values proposed by GE and accepted by the staff in its radiological consequence assessments. The staff's recalculated offsite doses resulting from DBAs using the bounding atmospheric relative concentration values are listed in revised Table 15.1 of this report.

The atmospheric relative concentration for the 2-hour EAB was determined by the most limiting DBA (fuel-handling accident) not to exceed the dose acceptance criteria specified in the SRP Section 15.7.4 (750 mSv (75 rem) for the thyroid and 60 mSv (6 rem) for the whole-body doses).

The bounding atmospheric relative concentration values for LPZ (the 8-hour time period from 0 to 8 hours, the 16-hour period from 8 to 24 hours, the 3-day period from 1 to 4 days, and the 26-day period from 4 to 30 days) were also determined by the most limiting DBA (LOCA) not to exceed the dose reference values given in 10 CFR Part 100 (3000 mSv (300 rem) for the thyroid and 250 mSv (25 rem) for the whole-body). Two-hour LPZ atmospheric relative concentration and an annual average (8760 hours) concentration values were obtained by logarithmic interpolation of the calculated LPZ bounding atmospheric relative concentrations. In determination of these χ/Q values, GE followed the guidelines provided in RG 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," (Rev. 1). The bounding atmospheric relative concentration values for the EAB, 2-hour LPZ, and annual average (8760-hours) are specified in Table 5.0, "ABWR Site Parameters," of the ABWR Certified Design Material (Tier 1 Design Control Document) and in Table 2.0-1, "Envelope of ABWR Standard Plant Site Design Parameters," of the SSAR.

The staff will independently assess χ/Q values for potential accident consequence assessments on a site-specific basis using onsite meteorological data (hourly cumulative frequency distributions) submitted by a COL applicant in accordance with RG 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants." In its evaluation, the staff will use the guidance provided in (1) RG 1.23, "Onsite Meteorological Programs," (2) RG 1.145, (3) RG 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluent in Routine Releases from Light-Water-Cooled Reactors," and (4) SRP Section 2.3.4, Short-Term Dispersion Estimates. As discussed in Section 2.3.4 of this report, this was DFSE COL Action Item 2.3.4-1.

If site-specific atmospheric relative concentration values are greater than the bounding values (e.g., less dispersion) used in this report, a COL applicant may have to augment ABWR ESF systems to meet the relevant requirements of 10 CFR Part 100 and GDC 19. However, this condition is not expected to arise frequently since the bounding values for atmospheric relative concentrations envelope most of the current operating reactor site meteorological characteristics.

Table 15.1 Revised radiological consequences of design-basis accidents

Postulated Accident	<u>EAB, mSv (rem)</u>		<u>LPZ, mSv (rem)</u>	
	Thyroid	Whole Body	Thyroid	Whole Body
Loss of coolant				
Containment leakage				
00- 02 hours	331 (33)	40 (4)	38 (4)	3 (0.3)
02- 08 hours			198 (20)	11 (1)
08- 24 hours			244 (24)	15 (1.5)
24- 96 hours			417 (42)	6 (0.6)
96-720 hours			248 (25)	2 (0.2)
Exfiltration			172 (17)	6 (0.6)
<hr/>				
Total containment leakage	331 (33)	40 (4)	1317 (132)	43 (4)
Emergency core cooling systems component leakage	20 (2)	10 (1)	72 (7)	< 10 (1)
Main steam isolation valve leakage	125 (13)	10 (1)	1302 (130)	< 10 (1)
<hr/>				
Total	476 (48)	60 (6)	2691 (269)	63 (6)
Main steamline failure outside containment				
With concomitant iodine spike	110 (11)	< 10 (1)	13 (1)	< 10 (1)
With preaccident iodine spike	270 (27)	< 10 (1)	30 (3)	< 10 (1)
Rod drop accident	6 (0.6)	< 10 (1)	20 (2)	< 10 (1)
Fuel-handling accident	750 (75)	< 10 (1)	20 (2)	< 10 (1)
Small line break accident	46 (5)	< 10 (1)	< 10 (1)	< 10 (1)
Reactor water cleanup line break				
With concomitant iodine spike	92 (9)	< 10 (1)	13 (1)	< 10 (1)
With preaccident iodine spike	225 (23)	< 10 (1)	30 (3)	< 10 (1)

Note: EAB = exclusion area boundary
LPZ = low population zone.

Transient and Accident Analysis

Table 15.2 Assumptions used to compute rod drop accident doses

Parameter	Value
Power level	4005 MWt (4.2 E+6 BTU/sec)
Peaking factor	1.55
Number of fuel rods perforated	770
Number of fuel rods melted	6
Condenser leak rate	1.0%/day
Fraction of fission product inventory release to coolant from melted fuel rods	
Iodines	50%
Noble gases	100%
Fraction of fission product inventory released to coolant from perforated fuel rods	
Iodines	10%
Noble gases	100%
Iodine fraction released to condenser	10%
Iodine fraction available for release from condenser after plate-out and partitioning	10%
Atmospheric diffusion values	
0-2 hour, exclusion area boundary	1.37E-3 sec/m ³ (3.88 E-5 sec/ft ³)
0-8 hour, low population zone	1.56E-4 sec/m ³ (4.42 E-6 sec/ft ³)

Table 15.3 Assumptions used to evaluate the main steamline and small line break accidents outside the containment

Parameter	Value
Mass of primary coolant released before main steam isolation valve closure	
Steam mass released	1.29E+4 kg (2.8E+4 lb)
Water mass released	2.2E+4 kg (4.84E+4 lb)
Mass of primary coolant released through small line	5.5E+3 kg (1.2E+4 lb)
Mass of primary coolant flashed	2.3E+3 kg (5E+3 lb)
Fraction of iodine in the primary coolant released	100 %
Fraction of noble gases in the primary coolant released	100 %
Primary coolant concentration (dose equivalent I-131)	
Technical specification limit, normal long-term operation	7.4E-3 mBq (0.2 µCi/gm)
Technical specification limit, normal short-term operation	1.5E-1 mBq (4.0 µCi/gm)
Atmospheric diffusion values	
0-2 hour, exclusion area boundary	1.37E-3 sec/m ³ (3.88 E-5 sec/ft ³)
0-8 hour, low population zone	1.56E-4 sec/m ³ (4.42 E-6 sec/ft ³)

Transient and Accident Analysis

Table 15.4 Assumptions used to evaluate the loss-of-coolant accident

Parameter	Value
Power level	4005 MWt (4.2E+6 BTU/sec)
Fraction of core inventory released	
Noble gases	100%
Iodines	50%
Iodines initial plate-out fraction	50%
Iodine chemical species	
Elemental	91%
Particulate	5%
Organic	4%
Suppression pool decontamination factor	
Noble gas	1
Organic iodine	1
Elemental iodine	2
Particulate	2
Pool bypass	50%
Iodine dose conversion factors	ICRP-30*
Primary containment leakage	0.5%/day
Main steam isolation valve leakage (total)	3.96 m ³ /hr (140 ft ³ /hr)
Standby gas treatment system	
Filter efficiency	97%
Flow rate	113 m ³ /min (4000 ft ³ /min)
Drawdown time	20 minutes
Emergency core cooling system (ECCS) system leakage	3.785 L/min (1.0 gpm)
Iodine core inventory in suppression pool	0.5 fraction
Suppression pool water volume	3.785E+6 m ³ (1E+6 gal)
Iodine partition factor for ECCS leak	0.1
Primary containment free volume	1.33E+4 m ³ (4.7E+5 ft ³)
Secondary containment free volume	8.5E+4 m ³ (3E+6 ft ³)
Secondary containment mixing efficiency	50%

* ICRP-30 International Commission on Radiation Protection Publication 30.

Table 15.5 Atmospheric dispersion (χ/Q) values used in accident evaluations

Time Period	χ/Q Value (sec/m^3) (Sec/ ft^3)
0-02 hour EAB	1.37E-3 (3.88 E-5)
0-08 hour LPZ	1.56E-4 (4.42 E-6)
8-24 hour LPZ	9.61E-5 (2.72 E-6)
1-04 day LPZ	3.36E-5 (9.51 E-7)
4-30 day LPZ	7.42E-6 (2.10 E-7)

Note: EAB = exclusion area boundary
LPZ = low population zone.

Table 15.6 Method to evaluate iodine removal in main steamlines and main condenser

Parameter	Value				
Source term	Regulatory Guide 1.3				
Main steam isolation valve leakage rate	3.96 m^3/hr (140 ft^3/hr)				
Leakage duration	30 days				
Condenser volume	9.85E+8 cc (3.47 E+4 ft^3)				
	Deposition Temperature (K) (F)	Velocities (cm/sec)	Resuspension Rate (in./sec)	Fixation Rate	
				(L/sec)	(L/sec)
Elemental and particulate iodine	300 (81)	3.2E-2	1.2 E-2	3.14E-6	4E-6
	400 (261)	5.0E-3	2.0 E-3	7.05E-6	8E-6
	500 (441)	1.0E-3	3.9 E-4	8.10E-6	1E-5
	560 (549)	6.2E-4	2.4 E-4	9.20E-6	2E-5
Organic iodine	300 (81)	1.4E-3	5.5 E-4	9.5E-8	4E-6
	400 (261)	3.5E-4	1.4 E-4	2.0E-7	8E-6
	500 (441)	1.0E-5	3.9 E-6	3.0E-7	1E-5
	560 (549)	1.3E-6	5.2 E-7	3.6E-7	2E-5
Component	Inside Diameter (cm) (in.)	Length (cm) (ft)	Thickness (cm) (in.)		
Main steamline	64 (25)	4773 (157)	2.5 (1.0)		
Drain line	6.7 (2.6)	610 (240)	1.1 (.4)		

Transient and Accident Analysis

Table 15.7 Iodine releases* GBq (curies)

Hours	Inlet to Main Steamlines	Inlet to Main Condenser	Outlet From Main Condenser
00 - 002	4.8E+05 (1.3E+04)	1.6E+04 (4.2E+03)	8.9E+02 (2.4E+01)
02 - 008	1.3E+06 (3.5E+04)	1.7E+04 (4.6E+03)	3.5E+03 (9.5E+01)
08 - 024	1.8E+06 (4.9E+04)	1.7E+04 (4.6E+03)	1.8E+04 (4.8E+02)
24 - 096	6.7E+06 (1.8E+05)	5.9E+05 (1.6E+04)	2.0E+05 (5.4E+03)
96 - 720	2.1E+07 (5.6E+05)	1.7E+06 (4.6E+05)	1.1E+06 (2.9E+04)

*Based on total MSIV leakage rate.

Table 15.8 Assumptions used in computing fuel handling accident doses

Parameter	Value
Power level	4,005 Mwt
Peaking factor	1.55
Number of fuel rods damaged	124 ⁽¹⁾ 5,120 ⁽²⁾
Number of fuel rods in cask	1,116
Filter iodine removal efficiencies:	
Organic	99%
Elemental	99%
Shutdown Times	24 hours ⁽¹⁾ 7 days ⁽²⁾ 120 days ⁽³⁾
Inventory released from damaged rods:	
Iodine and noble gases	10%
Kr-85	35%
Iodine fraction:	
Organic	0.25
Elemental	0.75
Atmospheric diffusion values	(sec/m ³) (sec/ft ³):
0-2 hour, exclusion boundary	1.37E-3 (3.88 E-5)
0-8 hour, low population zone	1.56E-4 (4.42 E-6)
(1) single fuel assembly dropping	
(2) heavy object dropping	
(3) fuel cask dropping	

Table 15.9 Assumptions and estimates of the radiological consequences to control room operators after a loss-of-coolant accident

Parameter	Value	
Control room free volume	7,000 m ³ (2E+5 ft ³)	
Recirculation rates		
Filtered intake	1.8 m ³ /sec	
Unfiltered intake	0.0	
Filtered recirculation	0.8 m ³ /sec	
Filter efficacy	95%	
Unfiltered control room infiltration rate (assumed)	4.7E-3 m ³ /sec (1.0 E-1 ft ³ /min)	
Duration of accident	30 days	
Breathing rate of operators in control room for course of accident	3.47 E-4 m ³ /sec (1.22 E-2 ft ³ /sec)	
Meteorology (wind speeds for all sectors)		
00 - 008 hours	4.0 E-3 sec/m ³ (1.1 E-4 sec/ft ³)	
08 - 024 hours	2.4 E-3 sec/m ³ (6.8 E-5 sec/ft ³)	
24 - 096 hours	1.5 E-3 sec/m ³ (4.3 E-5 sec/ft ³)	
96 - 720 hours	6.6 E-4 sec/m ³ (1.9 E-5 sec/ft ³)	
Iodine protection factor	27	
Iodine dose conversion factors*	ICRP-30**	
Iodine reduction factor for dual air intake	4	
Control room operator occupational factors		
00 - 008 hours	1	
08 - 024 hours	1	
24 - 096 hours	0.6	
96 - 720 hours	0.4	
Doses to control room operators	Thyroid dose*	Whole-body dose***
	mSv (rem)	mSv (rem)
00 - 008 hours	10 (1)	10 (1)
08 - 024 hours	30 (3)	3 (0.3)
24 - 096 hours	100 (10)	1 (0.1)
96 - 720 hours	<u>130 (13)</u>	<u>1 (0.1)</u>
Total	270 (27)	15 (1.5)

* Unweighted dose equivalent.

** ICRP-30 International Commission on Radiation Protection Publication 30.

*** Unweighted dose equivalent - red bone marrow.

Transient and Accident Analysis

The bounding atmospheric relative concentrations eliminate the need for either the COL applicant with the certified ABWR design or the staff to assess the offsite radiological consequence assessments for DBAs for the reactor site proposed if a COL applicant can demonstrate that (1) its atmospheric relative concentrations at the proposed ABWR reactor site are less than the bounding values used in this report, and (2) its design characteristics of main steamlines, steam drain lines, and condenser can remove equal or greater amounts of iodine than that claimed in the staff assessment in this report.

15.4.1 Control Rod Drop Accidents

In SSAR Section 15.4.9.6, GE states that the radiological consequences of a control rod drop accident need not be considered because such an accident is extremely unlikely with the improved design of the ABWR. The ABWR design employs the FMCRD system, which has several new features not found in current BWR locking piston CRDs.

In SSAR Section 15.4.9.2, GE states that for the rod drop accident to occur, it is necessary for such highly unlikely events as failures of both Class 1E separation-detection devices or the failure of the rod block interlock, and the failure of the latch mechanism to occur simultaneously with the occurrence of a stuck rod on the same FMCRD. GE further states that, therefore, there is no basis to postulate the occurrence of this event because of the low probability of such simultaneous occurrence of these multiple independent events.

The staff considered past licensing reviews, such as Clinton, Perry, and River Bend, and concluded that a control rod drop accident for the ABWR design results in radiological consequences less than a small fraction of the dose reference values specified in 10 CFR 100.11 even with conservative assumptions. SRP Section 15.4.9 (III) states that a specific calculation of the radiological consequences for this accident is not necessary unless unusual plant or site features are present, or the applicant's calculation shows an unusually large amount of fuel damage. However, the staff specifically evaluated this accident because it is the first application involving the ABWR standard design with hypothetical site boundaries. This evaluation should establish a reference for comparison of future applications incorporating the ABWR design.

To evaluate the radiological consequence of this accident, the staff postulated that the highest worth control rod becomes decoupled from its drive mechanism at a fully inserted position in the core. The drive mechanism is withdrawn, but the decoupled control rod is assumed to be stuck in place. At a later moment, the control rod

suddenly falls free and drops out of the core. This results in the insertion of a large positive reactivity into the core, causing a localized power excursion. An automatic safety feature would terminate this excursion with required operator action. The rod pattern control function of the RCIS limits the worth of any control rod by regulating the withdrawal sequence.

The staff estimated that such a rod drop would cause no more than 770 fuel rods to reach the threshold for cladding damage, with 6 fuel rods melting. Table 15.2 of this report lists the assumptions used for estimating this fuel damage which are consistent with those given in Appendices A (Physics and Thermal-Hydraulics) and B (Radiological Assumption) of RG 1.77, "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors," (Rev. 0). The computed doses are listed in Table 15.1 of this report and are well within the dose reference values of 10 CFR 100.11. Therefore, the staff concludes that the ABWR design is adequate to control the release of fission products following a postulated control rod drop accident.

15.4.2 Failure of Small Lines Carrying Primary Coolant Outside Containment

GDC 55 contains provisions to ensure isolation of all pipes carrying reactor coolant that penetrate the containment building. Exempted from these specifications are small-diameter pipes (instrument lines) that must be continuously connected to the primary coolant system in order to perform necessary functions. The design must include methods of mitigating the consequences of a rupture of an instrument line because the lines cannot be automatically isolated. GE submitted a radiological analysis for an instrument line failure in SSAR Section 15.6.2.5, and the results of the analysis are provided in SSAR Table 15.6.3.

GE postulated that a small steam or liquid line breaks inside or outside the primary containment and that a small instrument line, instantaneously and circumferentially, breaks at a location where it may not be able to be automatically isolated and where detection is not automatic or apparent.

GE estimated that 5,448 kg (12,000 lbs) of primary coolant would be released through the break before it is isolated. GE estimated that 2,270 kg (5,000 lbs) of the 5,448 kg released would flash to steam and be available for release.

While conducting past licensing reviews, such as Clinton, Perry, and River Bend, the staff determined that a small line break accident is expected to result in radiological consequences less than a small fraction of the dose reference values specified in 10 CFR 100.11. Further-

more, the staff believes that these postulated breaks are subsumed by the design-basis LOCA radiological consequences, as stated in SRP Section 15.6.2. However, the staff did perform a specific evaluation of this accident because this application is the first involving the ABWR standard design with hypothetical site boundaries. This evaluation should establish a reference for comparison of future applications incorporating the ABWR design.

The assumptions used for the evaluation are listed in Table 15.3 of this report. The computed doses are listed in Table 15.1 and are well within the dose reference values of 10 CFR 100.11. Therefore, the staff concludes that the ABWR design is adequate to control the release of fission products following a postulated small line break accident.

15.4.3 Main Steamline Failure Outside Containment

GE postulated one of the four main steamlines to rupture between the outer isolation valve and the turbine control valves. GE analyzed this hypothetical accident in SSAR Section 15.6.4.5 and concluded that no more than 34,800 kg (76,770 lb) of reactor coolant would be lost through the break before automatic isolation and less than 12,870 kg (28,373 lb) of that would be lost as steam. The results of the GE analysis are provided in SSAR Table 15.6.7.

The staff accepted GE's estimated value and calculated the potential radiological consequences assuming 34,000 kg (74,957 lb) of reactor coolant are released directly to the environment. The staff assumed that 100 percent of the iodine and noble gases present in the released coolant are released to the atmosphere within 2 hours as stated in RG 1.5, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Steamline Break Accident for Boiling Water Reactors," (Rev. 0). Other assumptions are listed in Table 15.3 of this report.

In Section 3.6 of this report, the staff evaluated the ABWR design for protection against postulated piping failures in fluid systems outside the containment. The staff's evaluation of the protection provided against adverse environmental effects (excluding radiation effects) resulting from postulated piping failure is discussed in Section 3.11 of this report.

The staff assumed two reactor coolant conditions for the evaluation. In Case 1, the lost coolant was contaminated with radioactive iodine at the limits included in the standard technical specifications (STS) (i.e., $7.4\text{E-}3$ mBq ($0.2 \mu\text{Ci}$) of dose-equivalent I-131 per gram) for BWRs at continued full-power operation. In Case 2, a concentration of $1.5\text{E-}1$ mBq ($4.0 \mu\text{Ci}$) of dose-equivalent I-131 per gram was assumed (the limits in the STSs above which the

reactor is required to be shut down). The SRP acceptance criteria are the dose reference values of 10 CFR 100.11 for Case 2, and less than 10 percent of these values for Case 1. Dose-equivalent I-131 is defined as any mixture of iodine isotopes yielding the same inhalation thyroid dose as the stated amount of pure I-131. The staff also considered the amounts of 13 noble gas isotopes that also would be released.

The major parameters and assumptions used in the staff's radiological consequence assessment are given in Table 15.3 of this report. The calculated doses are listed in Table 15.1 and are within the acceptance criteria of SRP Section 15.6.4. Therefore, the staff concludes that the ABWR design is adequate to control the release of fission products following a postulated steamline break accident.

15.4.4 Loss-of-Coolant Accidents

In SSAR Section 15.6.5, GE selected and analyzed a hypothetical design-basis LOCA and concluded that certain bounding sets of atmospheric relative concentration values in conjunction with the use of ESF systems provided in the ABWR design are sufficient to provide reasonable assurance that the radiological consequences of such an accident will be within the dose reference values established in 10 CFR Part 100.

To verify GE's conclusion, the staff independently assessed the radiological consequences of a hypothetical LOCA. In its assessment, the staff used assumptions and methods described in the appendices to SRP Section 15.6.5 and in RG 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Boiling Water Reactors," (Rev. 2). The major parameters and assumptions used in the staff's evaluation are listed in Table 15.4 of this report.

Since no specific site is associated with the ABWR plant, GE previously defined these two boundaries only in terms of various hypothetical atmospheric relative concentrations at fixed EAB and LPZ distances. Subsequently, GE proposed and the staff accepted the bounding χ/Q values as stated in Section 15.4 of this report. Using these bounding χ/Q values, the recalculated offsite doses resulting from a hypothetical LOCA are listed in Table 15.1. The computed doses in Table 15.1 are expressed as thyroid and whole-body exposure for 2 hours at the EAB and for 30 days at the boundary of the LPZ to allow direct comparison with the dose reference values established in 10 CFR 100.11.

The staff evaluated offsite radiological consequences using the current TID-14844 source term assumptions that are consistent with the guidelines in the applicable SRP

Transient and Accident Analysis

sections and regulatory guides, except for the following two deviations: The staff (1) provided a credit for radioactive iodine removal in the main steamlines and in the main condensers by holdup for decay and deposition and (2) accepted the ABWR design without an MSIV leakage control system (LCS) (see Section 15.4.4.2 for technical bases).

The staff postulated a hypothetical LOCA to determine the adequacy of the ESF systems designed to prevent release of fission products into the environment, using bounding meteorological conditions. The passive ESF systems for this purpose in the ABWR design are the primary containment and a secondary containment (reactor building). The staff considered these in conjunction with the standby gas treatment system (SGTS) and the pressure suppression pool scrubbing, both of which are active ESF systems.

The primary containment spray system, which is designed as a containment heat removal feature, has not been given credit in any drywell airborne fission product removal calculations. The spray system is designed as a safety-related system but is not automatically initiated as required in the SRP for iodine removal credit. GE has not requested any iodine removal credit for the spray system. In its assessment, the staff included containment leakage, main steam isolation valve leakage, and post-LOCA leakage from ESF systems outside containment as sources and radioactivity transport paths to the environment following a LOCA.

15.4.4.1 Containment Leakage Contribution

The ABWR primary containment design consists of a drywell, a wetwell, and supporting systems to limit fission product leakage during and following a LOCA with rapid isolation of all pipes or ducts that penetrate the containment boundary. It is designed to prevent the uncontrolled release of airborne radioactivity to the environment. GE proposed that the primary containment will be built and tested periodically to have a leak rate at design pressure of less than 0.5 percent by weight per day at the calculated peak containment pressure associated with a LOCA. The staff used this leak rate in its radiological assessment.

The secondary containment structure and supporting systems will collect and process radioactive material that may leak from the primary containment following a LOCA or that may result from a fuel-handling accident. The secondary containment is a seismic Category I reinforced-concrete structure with a volume of approximately $8.5E+4 \text{ m}^3$ ($3E+6 \text{ ft}^3$).

In SSAR Section 15.6.5.5.1, GE assumed that the primary containment leak rate into the reactor building through penetrations and ESF system components will not be greater than an equivalent release of 0.5 percent by weight per day of the primary containment free air weight for the first 24 hours after a LOCA and half of that value (0.25 percent/day) after 24 hours. GE based its assumption of a reduced leak rate on the finding that the primary containment pressure is reduced by more than a factor of 1/2 during the 12 hours following initiation of a LOCA; therefore, the driving force for leakage through the pathway is correspondingly reduced.

RG 1.3 (Rev. 2) assumes a constant containment leak rate for the duration of a LOCA, although it permits a reduced leak rate with supporting justification. Two rationales support a constant containment leak rate: (1) the pressure profile for a BWR does remain high for a long period of time and (2) for most plants, the leakage is only measured at the maximum value in accordance with Appendix J to 10 CFR Part 50. Therefore, the primary containment leak rate was identified as Open Item 143 in the DSER (SECY-91-355). Subsequently, GE revised the primary containment leak rate of 0.5 percent by weight per day for the entire duration of a LOCA, accepting the staff's position in a draft revision to the SSAR. This was identified as DFSER Confirmatory Item 15.4.4.1-1. In Amendment 24 to the SSAR Section 15.6.5.5.1, GE revised the leak rate to 0.5 percent by weight per day for the duration of a LOCA and this resolved DFSER Confirmatory Item 15.4.4.1-1.

The pressure within the reactor building is maintained slightly negative during normal operation by exhausting the reactor building air through the normal reactor building ventilation system. On receipt of an ESF actuation signal, the normal ventilation system is automatically switched off and the SGTS actuated. Following a postulated LOCA, the pressure in the secondary containment could increase from its initially negative pressure to a slightly positive value as a result of inleakage, air expansion because of heat, and the time required for the startup of the SGTS.

The SGTS is designed to achieve a negative pressure of 0.635 cm (0.25 in.) water gauge in the secondary containment within 20 minutes (drawdown time). Following the drawdown time, the staff assumed an iodine removal efficiency of 97 percent by the SGTS charcoal absorber with a depth of 15.24 cm (6 in.). In accordance with SRP Section 6.5.3, the staff assumed reactor building air mixing efficiency of 50 percent for the primary containment leakage before the release to the environment.

The staff estimates that wind speed in excess of 10 m/s (33 ft/sec) may create potential exfiltration of the reactor

building due to the low atmospheric pressure created outside the reactor building. The loss of the SGTS for the entire period of an accident (720 hours) will increase the offsite radiological consequences by a factor of 100, while the increased wind speed will improve atmospheric dispersion parameters by a factor of 33 based on the meteorological data submitted by GE on November 17, 1992. In the submittal, GE also stated that wind speed will exceed 10 m/s (33 ft/sec) no more than 5 percent of the time, based on meteorological data obtained from 28 BWR sites in the United States. Therefore, the loss of the SGTS for 36 hours (5 percent of 720 hours) could increase the offsite radiological consequences by 15 percent. This increase is reflected in revised Table 15.1 of this report.

Because the secondary containment (reactor building) is designed to completely enclose the primary containment, the staff assumed no bypass leakage to the environment from the primary containment except that directed through the main steamlines. In Section 6.2.3.1 of the DSER (SECY-91-355), the staff concluded that GE adequately addressed the criteria in Branch Technical Position CSB 6-3 of the SRP and that the design of barriers to preclude the secondary containment bypass flow (excluding that through main steamlines) is acceptable.

The fundamental characteristic of an BWR pressure-suppression containment is that steam released from the reactor coolant system will be condensed and scrubbed of radionuclides in a pool of water (the suppression pool) and the pressure rise in the containment will thereby be limited. This is accomplished by directing the steam from the reactor coolant system to the suppression pool through a vent system. However, leakage paths could exist in the pathway between the drywell and the wetwell airspace that could allow steam to bypass the suppression pool, potentially overpressurizing the containment. Potential sources of steam bypass include leakage through the vacuum relief valves, cracking of the drywell structure, and penetrations through the drywell structure.

In SSAR Section 15.6.5.5.1, GE assumed that any elemental and particulate iodine species purged to the suppression pool would be subject to a decontamination factor (DF) of 10. In the DSER (SECY-91-355), the staff found that credit may be given for the removal of fission products by the suppression pool if suppression pool DFs are evaluated in accordance with the methodology prescribed in revised SRP Section 6.5.5, "Pressure Suppression Pools or Fission Product Cleanup Systems" (issued in December 1988), which states that suppression pools are capable of scrubbing airborne fission products and that it is unduly conservative to ignore this capability.

However, the staff also stated in the DSER (SECY-91-355) that GE should provide the drywell leakage capability to justify the suppression pool DF of 10. The suppression pool DF was identified as Open Item 141 in the DSER (SECY-91-355).

In the DFSE, the staff conservatively assumed, in its radiological consequence analysis, that a DF of 2 is provided by the ABWR pressure suppression pool (equivalent to suppression pool steam bypass of 50 percent) for airborne radioactive iodine in elemental and particulate forms. This assumption is further supported by safety-related drywell and wetwell containment sprays in the drywell or wetwell, or both, which also would reduce the effect of suppression pool bypass leakage on containment performance; therefore, the staff stated in the DFSE that this DSER Open Item 141 was resolved. In Amendment 31 to the SSAR, GE accepted the staff position amending Section 15.6.5.5.1.1 of SSAR to reflect a DF of 2 by the suppression pool for airborne radioactive iodine in elemental and particulate forms. This item remains resolved.

The staff recently accepted the drywell leakage value of $(.05 \text{ m}^2) (.06 \text{ yd}^2)$ of effective leakage pathway) proposed by GE for the ABWR design. GE submitted a letter dated August 31, 1993, enclosing a revised set of design description and ITAAC on the suppression pool bypass issue. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.3 of this report.

While containment sprays in the drywell and/or wetwell also would reduce the effect of suppression pool bypass leakage on containment performance lowering its temperature and pressure, the sprays also scrub the containment atmosphere of fission products (even though the staff has not given a credit for scrubbing) and mitigate the effects of bypass on fission product distribution.

The staff will review, for each COL applicant, plant-specific TS, which require periodic inspections to confirm suppression pool depth, and surveillance tests to confirm drywell leak tightness. In DFSE, the staff identified this as TS Item 15.4.4.1-1. GE included this inspection requirement in the ABWR STS for such surveillance tests. Therefore, this item is resolved. The calculated doses for the ABWR resulting from the containment leak of 0.5 percent by weight per day are shown in Table 15.1 of this report.

Transient and Accident Analysis

15.4.4.2 Main Steam Isolation Valve Leakage

The main steamlines in BWR plants contain dual quick-closing main steam isolation valves (MSIVs) which isolate the reactor system in the event of a break in a steamline outside the primary containment, a design-basis LOCA, or other events requiring containment isolation. Although the MSIVs are designed to provide a leaktight barrier, it is recognized that they allow some leakage. The current BWR TS limit for MSIV leakage is typically $0.325 \text{ m}^3/\text{hr}$ ($11.5 \text{ ft}^3/\text{hr}$) per valve. Operating experience has indicated that degradation has occasionally occurred in the leak-tightness of MSIVs and that the valves may exceed their specified leakage limit.

Because of recurring problems with excessive leakage of MSIVs, the staff recommended in RG 1.96, "Design of Main Steam Isolation Valve Leakage Control Systems for Boiling Water Reactor Nuclear Power Plants," (Rev. 1), installation of a supplemental leakage control system (LCS) to ensure that the isolation function of the MSIVs complies with the specified limits. Most of the currently operating BWRs have an LCS.

In response to the MSIV leakage concerns, the BWR Owners Group (BWROG) commissioned a program of studies to determine the causes of high leak rates and the means to eliminate them. The results of these studies were submitted to the Nuclear Regulatory Commission (NRC) in General Electric proprietary reports, NEDO-31643P (November 1988) and NEDO-31858P (February 1991), both entitled, "Increasing Main Steam Isolation Valve Leakage Rate Limits and Elimination of Leakage Control Systems." GE referenced these reports as the basis for not including an LCS and requesting a substantially higher ($0.99 \text{ m}^3/\text{hr}$ ($35 \text{ ft}^3/\text{hr}$) per valve) MSIV leakage limit.

The MSIVs generally have not provided a leak-tight containment pressure boundary to the extent intended in the plant design. Although substantial progress has been made in recent years to identify the causes of the leakage and reduce the amount of leakage, the current typical TS limit of $0.325 \text{ m}^3/\text{hr}$ ($11.5 \text{ ft}^3/\text{hr}$) per valve is still difficult to achieve when the valve is rapidly closed against a full-flow condition at reactor operating pressure and temperature.

The staff conservatively assumes for operating plants that the TS leakage limit of $0.325 \text{ m}^3/\text{hr}$ ($11.5 \text{ ft}^3/\text{hr}$) per MSIV is released directly into the environment in calculating offsite radiological consequences of potential accidents (10 CFR Part 100). No credit is currently given for the integrity and leaktightness of the main steam piping and condenser to provide holdup and plate-out of fission products.

The BWROG proposed an approach adopted by GE which would allow higher leakage limits of $3.96 \text{ m}^3/\text{hr}$ ($140 \text{ ft}^3/\text{hr}$) total for four steamlines and would not include an LCS for the ABWR design. In this alternative approach, GE used the main steam piping (and its associated piping) and the condenser to mitigate the radiological consequences of an accident that could result in offsite exposures comparable to the dose reference values specified in 10 CFR Part 100.

In the DSER (SECY-91-355), the staff stated that it was evaluating whether a credit for the fission product attenuation in the main steamlines and for the condenser is appropriate and reasonable for BWRs even though the main steamlines downstream from the MSIV and its condensers are not designed to withstand the safe shutdown earthquake (SSE), as defined in Appendix A, Section III(c), to 10 CFR Part 100. This was identified as DSER (SECY-91-355) Open Item 142. The staff completed its evaluation, and has accepted the BWROG and GE proposals for the ABWR design based on the following radiological and seismic analyses.

15.4.4.2.1 Radiological Analysis

• MSIV Leakage Pathways

Following a LOCA, three potential release pathways exist for main steam leakage through the MSIVs:

- (1) main steam drain lines (typically 5.08 cm (2 in.) in diameter) to the condenser with delayed release to the environment through the low-pressure turbine seals
- (2) turbine bypass lines (typically 15.2 cm (6 in.) in diameter) to the condenser with delayed release to the environment through the low-pressure turbine seals
- (3) main steamline (typically 61 cm (24 in.) in diameter) turbine stop and control valves through and high-pressure turbine seals to the environment bypassing the condenser

The consequences of leakage from pathways 1 and 2 will be essentially the same because the condenser can be used to process MSIV leakage. The condenser iodine removal efficiency will vary depending on the inlet location of the bypass or drain line piping, but in either case, iodine removal will occur. However, for pathway 3, MSIV leakage through the closed turbine stop and control valves will not be processed via the condenser. For this case, iodine will be removed in the high-pressure turbine, which has a large internal

surface area associated with the turbine blades for plate-out of the radioactive material.

The staff assumed that turbine bypass, stop, and control valves will be closed upon loss of the electro-hydraulic control system following a LOCA, and the drain valve can be manually opened to provide a leakage pathway from the control room to the condenser via a safety-related power source following a LOCA. GE submitted on August 31, 1993, a revised set of Tier 1 design descriptions of the nuclear boiler system. In Section 2.1.2 and corresponding Figure 2.1.2b of this document, GE stated that the main control room will include main steam drain valve controls and status displays. The staff finds that the remote control requirement placed in the ABWR Tier 1 document is acceptable. The overall adequacy and acceptability of the ABWR Tier 1 design descriptions are evaluated in Chapter 14.3 of this report.

In the DFSER, the staff stated that the TS must address operability of the drain valve. Subsequently, the staff agreed with GE that the operability of the drain valve need not to be included in the ABWR STSs but should be included in the plant operating procedures. This position makes GE's TS consistent with the BWR STS developed under the Commission's TS improvement program. See the general discussion in Chapter 16 of this report. Therefore, this resolved DFSER TS Item 15.5.1.2.1-1.

The BWROG has also identified the same three leakage pathways as discussed above. Of these, the BWROG and GE proposed to use the drain line downstream of the MSIVs as a leakage pathway to the condenser. The staff also chose to use the main steamline drains for the MSIV leakage pathway in assessing iodine holdup and deposition for the ABWR design.

The BWROG and GE indicated that the bypass piping of the main steamline is another appropriate leakage pathway that can be used for MSIV leakage control. Either pathway is an acceptable approach, provided that the chosen pathway will be available for MSIV steam leakage to reach the main condenser. Comparing the two pathways, more iodine is removed (by holdup and deposition) through the turbine bypass pathway than through the drain line pathway because of the larger pipe size, which will have more surface area for deposition.

The staff believes that as long as either the turbine bypass or the drain line leakage pathway is available, MSIV leakage through the closed turbine stop and control valves (pathway 3) will be negligible and

essentially all of the releases will be through the main condenser. This is because there will be essentially no differential pressure in the main steamline downstream of the MSIVs following the closure of the valves.

Furthermore, any MSIV leakage through pathway 3, if any, will have been subjected to the same iodine removal processes in the main steamlines (up to turbine stop valves) as the other pathways, and it will be further subjected to iodine removal by deposition in the high-pressure turbine internal surfaces. The main condenser does not remove iodine in pathway 3.

In calculating the contribution to the LOCA dose, the staff assumed that one of the inboard isolation MSIVs failed to close, thus allowing contaminated steam to travel to the outboard valve. This outboard valve and the outboard valves from the other three steamlines were assumed to have a total leak rate of 3.96 m³/hr (140 ft³/hr).

• Iodine Transport Model

Basic chemical and physical principles indicate that gaseous iodine and airborne iodine particulate material will deposit on surfaces. Several laboratory and in-plant studies have demonstrated that gaseous iodine deposits will occur by chemical adsorption, and particulate iodine deposits will occur through a combination of sedimentation, molecular diffusion, turbulent diffusion, and impaction. Gaseous iodine exists in nuclear power plants in several forms: elemental (I₂), hypiodous acid (HIO), organic (CH₃), and particulates. In accordance with RG 1.3, the staff assumed 91 percent of iodine will be in the elemental form (inhypiodous acid), 5 percent in the particulate form, and 4 percent in the form of organic iodides.

Each of these forms deposits on surfaces at a different rate, described by a parameter known as the deposition velocity. The elemental iodine form, being the most reactive, has the largest deposition velocity, and organic iodide has the smallest. Further, studies of in-plant airborne iodine show that elemental and particulate iodine deposited on the surface undergoes both physical and chemical changes and can either be resuspended as an airborne gas or become permanently fixed to the surface. The data also shows that the iodine can change its form so that iodine deposited as one form (usually elemental) can be resuspended in the same or in another form (usually organic). Conversion can be described in terms of resuspension rates that are different for each iodine species. Chemical surface fixation can similarly be described in terms of a surface fixation rate constant.

Transient and Accident Analysis

The transport of gaseous iodine in elemental and particulate forms has been studied for many years and several groups have proposed different models to describe the observed phenomena. Examples of the studies are listed below.

- (1) NUREG/CR-2713, "Vapor Deposition Velocity Measurement and Consolidation for Iodine and Cesium Iodine," S.L. Nicolosi and P. Baybutt, May 1982.
- (2) NUREG/CR-4397, "In-Plant Source Term Measurements at Prairie Island Nuclear Generating Station," J.W. Mandler et al., September 1985.
- (3) IN-1394, Idaho Nuclear/National Reactor Testing Station, "Deposition of Iodine 131 in CDE Experiments," Nebecker et al., 1969.
- (4) BMI-1863, Fission Product Deposition and Its Enhancement Under Reactor Accident Condition: Deposition on Primary System Surfaces, J.M. Genco et al., May 1969.
- (5) "Transmission of Iodine Through Sampling Lines," 18th DOE Nuclear Airborne Waste Management and Air Cleaning Conference, P.J. Unrein et al., October 1984.

The staff used the model developed by an NRC contractor (J.E. Cline and Associates, Inc., 1991) for iodine removal in BWR main steamlines and the main condenser following a LOCA.

The staff model treats the MSIV leakage pathway as a sequence of small segments for which instantaneous and homogeneous mixing is assumed; the mixing computed for each segment is passed along as input to the next segment. The number of segments depends on the parameter of the line and flow rate and can be as many as 100,000 for a long, large-diameter pipe and a low flow. Each line segment is divided into five compartments that represent the concentrations of the three airborne iodine species, the surface that contains iodine available for resuspension, and surface iodine that has reacted and is fixed on the surface. The staff's model considers three iodine species: elemental, particulate, and organic. A fourth species, hypoiodous acid, is considered for the purpose of the staff's model to be a form of elemental iodine. All iodine in the segment undergoes radioactive decay. The resulting concentration from each segment of the deposition compartment serves as the input to the next segment.

The staff's transport model also assumes iodine transport through the condenser as a dilution flow rather than the plug flow as in the steamlines. The staff assumes that the iodine entering the condenser mixes instantaneously with a volume of air in the condenser and that the diluted air exhausts at the same time and same rate as the input air (MSIV leakage) flows into the condenser.

The staff developed the equations for iodine deposition velocities, resuspension rates, and surface fixation rates as a function of temperature using published data from the above-mentioned literature. The equations and data are contained in the report by Cline and Associates. The equation for the deposition velocity of elemental iodine is based on the least-squares fit to the available data. Deposition velocity equations for HOI and organic iodine are based on the values at 30 °C (86 °F); because of the lack of data at elevated temperatures, their temperature dependence is assumed to be similar to that of elemental iodine. The staff based its resuspension and fixation equations as a function of temperatures available in the literature based on measurements taken at ambient temperature. The staff assumed that resuspension and fixation rates will increase with increasing temperature.

The parameter and assumptions used in the development of the iodine transport model are listed in Table 15.6 of this report. Calculated iodine releases from the condenser after holdup and plate-out in the main steamlines and condensers are shown in Table 15.7 of this report.

The technical references mentioned above and the staff's model indicate that particulate and elemental iodine would be expected to deposit on surfaces at rates varying with temperature, pressure, gas composition, surface material, and particulate size. The staff, therefore, concludes that an appropriate credit for the removal of iodine in the main steamlines and main condensers should be provided in the radiological consequence assessment following a design-basis accident for the ABWR. The staff considers DSER Open Item 142 resolved.

The amount of iodine removal credit for the ABWR design is shown in Table 15.7 of this report. In the DFSER, the staff stated that the COL applicant will need to recalculate removal credit on the basis of its design characteristics of main steamlines, drain or bypass line, and main condenser. This was identified as COL Action Item 15.4.4.2.1. In Sections 15.6.5.5.1.2 and 15.6.5.5.1.3 of the SSAR, GE stated that COL applicants will recalculate iodine

removal credit on the basis of its design characteristics of main steamlines, drain lines, and main condenser. The staff finds this acceptable.

15.4.4.2.2 Seismic Analysis

Section III(c) and VI of Appendix A to 10 CFR Part 100 require that structures, systems, and components (SSCs) necessary to ensure the capability to mitigate the radiological consequences of accidents that could result in exposures comparable to the dose guideline exposures of Part 100 be designed to remain functional during and after an SSE. Thus, the main steamline, portions of its associated piping, and the main condenser are required to remain functional if the SSE occurs. Consequently, these components are required to be classified as safety-related and seismic Category I. In addition, Appendix A to 10 CFR Part 100 requires that the engineering method used to ensure that the safety functions are maintained during and after occurrence of an SSE involve the use of either a suitable dynamic analysis or a suitable qualification test.

For the purpose of giving credit to iodine holdup and plate-out in the main steamlines and condensers, the staff's model requires that the main steam piping (including its associated piping to the condenser) and the condenser to remain structurally intact following an SSE, so it can act as a holdup volume for fission products.

In the DFSE, the staff stated that the ABWR design did not fully comply with the requirements of 10 CFR Part 100 because portions of the main steamlines, bypass and drain piping, and the condenser were not classified as seismic Category I. This DFSE statement was incorrect. The staff's position as discussed in detail in Section 3.2.1 of the DFSE and of this report is that the ABWR design provides reasonable assurance that the main steam piping from the outmost isolation valve up to the turbine stop valve, the MSIV leakage pathway (i.e., the drain line or bypass line) up to the condenser, and the main condenser will remain structurally intact and leaktight, so that they can act as a holdup volume for fission product during and following an SSE. The staff has determined that the ABWR design does comply with the requirements of 10 CFR Part 100 because the SSCs described above are designed to remain functional during and following a SSE. This issue is also discussed in the staff's SECY-93-087 entitled, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs," dated July 21, 1993.

Open Item 15.4.4.2.2-1 in the DFSE indicated that the staff would have to describe new ABWR design requirements which would enable it to meet 10 CFR

Part 100 in a different manner or require that GE request exemption from 10 CFR Part 100 and provide an evaluation of the request in the SSAR. Section 3.4.1 of this report describes the key design requirements for the ABWR that resolve DFSE Open Item 15.4.4.2.2-1.

15.4.4.3 Post-LOCA Leakage Contribution from Engineered Safety Features Systems Outside Containment

Any leakage of water from ESF components (valve stems and pump seals) located outside of the primary containment releases fission products to the secondary containment during the recirculation phase of long-term core cooling following a LOCA. GE estimated such leakage from the RHR system, HPCF system, and reactor core isolation cooling (RCIC) system to be less than $6.2\text{E-}3$ L/min ($1.6\text{E-}3$ gpm). GE's leakage estimate indicated valve stem leakage of $4.5\text{E-}3$ L/min ($1.2\text{E-}3$ gpm) from a total of 45 valves and $1.7\text{E-}3$ L/min ($4.4\text{E-}4$ gpm) from a total of 5 pumps in these systems. In Section 5.2.5 of this report, the staff states that an identified leakage within the drywell (inside of the primary containment) could reach 5.8 m³/hr (25 gpm) at which point it would activate an alarm in the control room.

The staff used a conservative leakage value of 3.8 L/min (1.0 gpm) (instead of $6.2\text{E-}3$ L/min ($1.6\text{E-}3$ gpm) proposed by GE), which is the typical BWR leakage value used by the staff in its radiological consequence assessment. The COL applicant will provide this leakage value in its plant-specific TS. Section 17.6.3 (limiting condition for operation (LCO) 3.4.3 of the ABWR TS) requires unidentified reactor coolant system leakage to be less than 1 gpm. This resolved DFSE TS Item 15.4.4.3-1.

The staff further assumed that 10 percent of the waterborne iodine that leaks from the ECCSs (RHR, HPCF, and RCIC) systems will become airborne and be released to the environment through the SGTS after 20 minutes of the secondary building drawdown time (no filtration credit is given prior to the drawdown time). The offsite radiological consequences of an ESF component leak outside the primary containment are relatively small and are given in Table 15.1 of this report.

15.4.4.4 Conclusion

The staff reviewed GE's analysis and performed an independent analysis of the radiological consequences resulting from each of the transport paths described in Section 15.4.4 of this report. The calculated thyroid and whole-body doses are listed in Table 15.1. Based on the above evaluation and the recalculated radiological

Transient and Accident Analysis

consequences shown in Table 15.1, the staff concludes that the MSIV leak rate limit of $3.96 \text{ m}^3/\text{hr}$ ($140 \text{ ft}^3/\text{hr}$) total and the proposed ABWR design without an MSIV LCS are acceptable. It further concludes that GE's proposed bounding atmospheric relative concentrations (χ/Q) for the EAB and to the LPZ of the ABWR plant, in conjunction with the ESF systems provided in the ABWR design are sufficient to provide reasonable assurance that the radiological consequences of a postulated LOCA will be within the dose reference values in 10 CFR Part 100.

15.4.5 Fuel Handling and Fuel Cask Drop Accidents

In SSAR Sections 15.7.4 and 15.7.5, GE presented radiological consequence analyses of fuel handling and spent fuel cask drop accidents. The staff analyzed the radiological consequences for the following three different types of fuel handling accidents resulting from an unspecified failure of a lift mechanism due to (1) a single fuel assembly with 124 spent fuel rods dropped onto the irradiated fuel stored in the spent fuel pool, (2) a raised single fuel cask containing 18 fuel assemblies in it dropped from the level of the refueling floor to ground level through the refueling floor maintenance hatch, and (3) a raised heavy object (i.e., steam dryer, moisture separator) dropped onto the fuel in the reactor vessel during refueling operation.

For the single fuel assembly accident, the kinetic energy of a single falling fuel assembly was assumed to break open the maximum possible number of fuel rods (124 spent fuel rods) using perfect mechanical efficiency. Instantaneous release of noble gases and radioiodine vapor from the gaps of the broken rods (10 percent of noble gases and iodine inventories in the reactor core except 30 percent assumed for krypton-85) was assumed to occur, with the released gases bubbling up through the fuel pool water (with an overall effective DF of 160 for iodines and of 1 for noble gases). Radiation monitors located within the normal ventilation system are designed to isolate that system automatically and direct all fuel handling building exhaust to the SGTS.

The SGTS is designed to achieve a negative pressure of 0.635 cm (0.25 in.) water gauge in the secondary containment within 20 minutes (drawdown time). During the drawdown time, the staff assumed that radioactive iodines would be released directly to the environment without credit for filtration. Following the drawdown time, the staff assumed the SGTS charcoal absorber with a depth of 15.24 cm (6 in.) would remove iodine from the released gas at a 99 percent efficiency.

The fuel cask drop accident was assumed to result from an unspecified failure of the cask-lifting mechanism, thereby

allowing the cask to fall approximately 19.5 m (64 ft) from the level of the refueling floor to ground level through the refueling floor maintenance hatch. GE stated that each cask will have the maximum capacity of 1116 spent fuel rods (18 fuel assemblies) based on the largest capacity cask projected to be available. GE proposed, and the staff accepted, the minimum fuel storage (decay) time of 120 days before cask-loading operation commences after reactor fueling. The minimum storage time of 120 days is based on the administrative controls currently used by operating BWRs.

In NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," the staff discusses the potential for an accidental load drop on nuclear fuel or safety-related equipment causing excessive offsite radioactivity releases, inadvertent criticality, loss of water inventory in the reactor or spent fuel pool, or loss of safe-shutdown equipment. The NUREG also recommends guidelines to prevent or mitigate these potential consequences and states that the guidelines in NUREG-0554, "Single-Failure-Proof Cranes for Nuclear Power Plants," can be followed in lieu of upgrading existing crane and lifting devices. NUREG-0554 provides guidance for the design, fabrication, installation, and testing of new cranes that are of a high reliability design and it defines that a single-failure-proof crane system as a system designed so that a single failure will not result in the loss of the capacity of the system to safely retain the load.

In Section 9.1.5.2.1 of the SSAR, GE states that the reactor building crane will be designed to meet the single-failure-proof requirements of NUREG-0554, and in Section 9.1.6.6 of the SSAR, GE states that the COL applicant should provide, among other things, heavy load handling system operating and equipment maintenance procedures for NRC review. This review will confirm that a heavy load drop is not a design-basis event by determining that the crane system meets (1) single-failure-proof criteria, and (2) prevention of load unbalancing (e.g., improper placement of slings) which could potentially defeat the single-failure-proof criteria.

For the heavy load drop accident to occur, the following steps must occur in sequence: (1) a heavy load is slung wrong (procedural), (2) the heavy load is transported to an incorrect position over the core (procedural), (3) the sling fails (equipment failure), and (4) an alternate slung hook fails (equipment failure).

Nevertheless, the staff assumed in the DFSE that the heavy load drop accident occurs because the lifting device fails (unspecified cause), drops, or tips a heavy object onto the fuel in the reactor vessel and breaks open all fuel rods in 10 percent of fuel assemblies (approximately

80 assemblies) in the reactor vessel. The staff assumed the minimum radioactivity decay time of 7 days from the time of reactor shutdown until the lifting of the heavy object based on a conservative BWR refueling schedule.

The staff evaluated a postulated fuel handling accidents for the ABWR in accordance with the guidance of SRP Section 15.7.4, using assumptions consistent with Positions C.1.a through C.1.k of RG 1.25 (Rev. 0), "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling Storage Facility for Boiling and Pressurized Water Reactors," (Rev. 0).

The assumptions used for the spent fuel handling accidents are listed in Table 15.8. The offsite doses recalculated for these accidents are given in Table 15.1. The calculated offsite doses for all three cases considered are within the dose guidelines specified in SRP Section 15.7.4, specifically, the estimated doses resulting from this type of accident are less than or equal to 25 percent of the 10 CFR Part 100 dose limits (listed in Section 15.4 above). Therefore, the staff concludes that the standard ABWR design is adequate to control the release of fission products following postulated spent fuel handling accidents.

15.4.6 Postulated Radioactive Releases Resulting from Liquid Tank Failures

The staff reviewed this accident in accordance with SRP Section 15.7.3. Tanks and associated components containing radioactive liquids outside containment are acceptable if the failure of the equipment does not lead to radioactive concentrations that exceed the limits of 10 CFR Part 20, Appendix B, Table II, Column 2, at the nearest potable water supply in an unrestricted area or if special design features to mitigate the effects of the accident are incorporated in the design of systems that do not meet the requirements of 10 CFR Part 20.

All the liquid radwaste tanks, including the evaporator concentrate tanks, that could adversely affect the potable water supply if they fail are located in the radwaste building. The base mat and outside walls of the building are seismic Category I to a height necessary to retain spill liquids within the building (SSAR Section 11.2.1.2.2). Additionally, in accordance with SSAR Section 15.7.3.1, all compartments containing liquid radwastes are steel lined up to a height capable of containing the release of all the liquid radwastes into the compartments. For the above reasons, GE considers it as remote that any major accident that involves the release of liquid radwastes into these volumes would result in the release of liquid radwastes to the environment via the liquid pathway. In SSAR Section 15.7.3.1, GE states that initially the releases would

be contained in other holding tanks or emergency tanks. GE further states that plant operating techniques and administrative procedures will contain detailed system and equipment operating instructions and, therefore, will significantly minimize the potential for operator error that can cause liquid radwaste release. In addition, the liquid radwaste system design will include a positive action interlock system to prevent inadvertent opening of a drain valve. GE concludes that if a release of liquid radwaste occurs, the steel lining would contain the release until the operator could use the floor door sump pumps to pump the release into holding tanks or emergency tanks.

GDC 60 requires the control of releases of radioactive materials to the environment including both gaseous and liquid effluents under both normal and anticipated operational occurrences. Further, sufficient holdup capacity is required. As discussed above, GE's liquid radwaste design provides the necessary control and liquid effluent storage capacity to reduce the potential effect of a failure of a radioactive liquid containing tank and its associated components. Therefore, GE's design complies with GDC 60 and any potential release associated with a liquid tank failure will not result in radionuclide concentrations in water exceeding the limits of 10 CFR Part 20, Appendix B, Table II, Column 2, in any unrestricted area. The staff finds this aspect of the design acceptable.

15.4.7 Reactor Water Cleanup System Pipe Break Accident

In Section 15.6.6 of the SSAR, GE postulated a reactor water cleanup system pipe break outside the primary containment as a DBA in response to an inquiry by the Advisory Committee on Reactor Safeguards (ACRS) Subcommittee on ABWRs during its meeting on June 17, 1993, at San Jose, California. The break was assumed to be instantaneous, circumferential, and to occur on the downstream side of the outmost containment isolation valve but on the upstream side of the reactor water demineralizer. GE further assumed 75 seconds of break flow time (45-second builtin delay time for flow differential pressure instrumentation to activate an isolation signal and 30 seconds for the motor-operated isolation valve to close).

GE analyzed this hypothetical accident and concluded that no more than 28,800 kg (61,670 lb) of reactor coolant would be lost through the break before automatic isolation occurred and less than 9,900 kg (21,800 lb) of that would be lost as steam. The break line is equipped with a 140-cm² (23 in.²) flow limiter. The staff accepted and used in its assessment the reactor coolant break flow of 28,800 kg estimated by GE.

Transient and Accident Analysis

The current SRP neither lists nor classifies the reactor water cleanup system pipe break accident as a DBA. However, the staff assessed the radiological consequences of this accident assuming two reactor coolant conditions for the evaluation. In Case 1, the lost coolant was contaminated with radioactive iodine at the limits in the STS (i.e., $7.4\text{E-}3$ mBq) of dose-equivalent I-131 per gram (90.8 mCi of dose-equivalent I-131 per pound) at continued full-power operation. In Case 2, a concentration of $1.5\text{E-}1$ mBq of dose-equivalent I-131 per gram (1816.0 mCi of dose-equivalent I-131 per pound) was assumed (the limits in the STSs above which the reactor is required to be shut down).

The staff chose the dose reference values of 10 CFR 100.11 for Case 2 and less than 10 percent of these values for Case 1 as two acceptance criteria. These are the same as those given for the main steamline break outside containment in SRP Section 15.6.4. Dose-equivalent I-131 is defined as any mixture of iodine isotopes yielding the same inhalation thyroid dose as the stated amount of pure I-131. The staff also considered the amounts of 13 noble gas isotopes that also would be released.

The calculated doses are listed in Table 15.1 and are within the staff's acceptance criteria listed above. Therefore, the staff concludes that the ABWR design is adequate to control the release of fission products following a postulated reactor cleanup system pipe break accident.

15.5 Anticipated Transient Without Scram

15.5.1 Design Features

The ABWR design incorporates electric-hydraulic fine motion control rod drives (FMC RDs) to perform motor-driven scram and hydraulic scram. In response to a scram signal, the control rods will be inserted hydraulically by means of the stored energy in the scram accumulator, similar to the current operating BWR CRDs. A scram signal also will be given simultaneously to insert the FMC RD electrically via the FMC RD motor drive. This diversity, hydraulic and electric methods of scrambling, provides a high degree of assurance of rod insertion on demand.

The ABWR is designed with an alternate rod insertion (ARI) system that will be independent from the existing reactor protection system (RPS) from the sensor output to the final actuation device. The ARI system will have redundant scram air header exhaust valves. The ARI system is designed to perform its function in a reliable

manner. Detailed evaluations of the ARI and RPS systems are given in Chapter 7 of this report.

The ABWR also is designed with a standby liquid control system (SLCS) that will automatically inject 379 L/min (100 gpm) of sodium pentaborate solution into the reactor pressure vessel (RPV) with the simultaneous operation of both pumps. The 326 L/min (86 gpm) equivalency specified in the ATWS rule (10 CFR 50.62) for the 638 -cm (251 -in.) RPV is satisfied by the 379 L/min (100 gpm) provided for the 706 -cm (278 -in.) ABWR vessel. The ABWR SLCS initiation is automatic as required by the ATWS rule and is designed to perform its function in a reliable manner. The detailed evaluation of SLCS is given in Section 9.3.5 of this report.

ABWR has equipment to trip the RIPs automatically under conditions indicative of an ATWS. The RIPs are automatically tripped on reactor high pressure ($7,860$ kPa (1125 psig)) and RPV Level 2. The RIP trip equipment is designed to perform its function in a reliable manner as required by the ATWS rule.

The ABWR design also provides recirculation runback for all scram signals and feedwater runback on reactor high pressure and startup range neutron monitoring system (SRNM) not downscale for 2 minutes. Automatic depressurization system automatic inhibit is also provided with reactor water level 1.5 and average power range monitor ATWS permissive. This feature is discussed in Section 7.4.1.1 of this report.

The ABWR complies with the prescriptive design requirements of the ATWS rule and is designed to mitigate the effects of an ATWS event.

15.5.2 Analysis

GE submitted the ATWS analysis in SSAR Appendix 15E. GE analyzed the limiting transients identified in NEDO-24222 (proprietary) (a supporting document used for the ATWS rule). GE used the NRC-approved ODYNA and REDYA computer codes for the analysis. GE analyzed cases with the ARI system operational, without the ARI system but with FMC RD system operational, and without the ARI and FMC RD systems but with SLCS operational. GE compared the results with the performance guidelines for fuel integrity (coolable core geometry), containment integrity (45 psi (414 kPa)), suppression pool temperature (207 °F (97 °C)), primary system pressure ($1,500$ psig (10446 kPa)) and long-term shutdown cooling and found them acceptable.

The staff performed audit calculations to verify that the ABWR design is satisfactory to mitigate the effects of an

ATWS. The study focused on the consequences of manual SLCS actuation and no recirculation pump runback on scram signals other than reactor high pressure and reactor low level. One problem that was identified initially was the potential for fuel damage if only the FMCRD system provides slow scram (motor-driven scram) and there were no other actions to lower power. In this case, the power shifted to the top of the core when the control rods entered from the bottom and, in combination with the delay in shutting down the reactor because of the slowness of the FMCRD system, could lead to higher linear power densities and the potential for fuel damage. If the use of the FMCRD system was always in combination with a reactor internal pump trip or recirculation runback, then overall power would be reduced sufficiently to avoid excessive power densities when the power shifted to the top of the core.

The new design feature of recirculation runback on any scram signals or any ARI/FMCRD run-in signals ensures that there is no potential for any unacceptable power shift to the top of the core.

Another feature included of the initial study was the time available for SLCS actuation. Assuming the failure of ARI and FMCRD, the original design called for manual operation of SLCS; however, if the system was not started within a few minutes, containment integrity would be threatened. GE resolved this issue by changing the ABWR design so that the SLCS is automatically started at 3 minutes into an ATWS event.

The results of calculations to evaluate the thermal hydraulic stability of ABWR under the recirculation runback and feedwater runback conditions associated with ATWS events have been provided for staff review. ABWR design response to ATWS enables avoidance of large oscillations and the staff audit of stability calculations is complete. This is discussed in detail in Section 4.4 of this report (DFSER Open Item 4.4-1).

DSER (SECY-91-355) Open Items 140 and 144 are dependent on and are superseded by the DFSER Open Item 4.4-1. Since Open Item 4.4-1 has been resolved the DSER open items are also resolved.

16 TECHNICAL SPECIFICATIONS

16.1 Introduction

As part of its design certification application for the advanced boiling water reactor (ABWR) and as required by paragraph (1)(i) of 10 CFR 52.47(a), GE must provide Technical Specifications (TS) that are technically relevant and not site-specific for the ABWR design. The TS provided must comply with the requirements of paragraph (b)(6)(vi) 10 CFR 50.34, which states that proposed TS are to be prepared in accordance with the requirements of 10 CFR 50.36. 10 CFR 50.36 details the specific items (such as safety limits, limiting safety system settings, limiting control settings, limiting conditions for operation, etc.) that must be included in the TS.

In reviewing proposed TS for compliance with 10 CFR 50.36, the staff evaluates the conformance of the proposed TS to standard technical specifications (STS) which most closely resemble the plant design.

The staff review of the ABWR TS was closely coupled to the development of the improved STS under the TS Improvement Program in accordance with the interim Commission Policy Statement on TS Improvements for Nuclear Power Plants (52 FR 3788 dated February 6, 1987). Since the ABWR design evolved primarily from the BWR/6 design, most of the ABWR TS were modeled after NUREG-1434, "Standard Technical Specifications - General Electric Plants, BWR/6." However, since the ABWR containment most closely resembles plant containments covered by NUREG-1433, "Standard Technical Specifications - General Electric Plants, BWR/4," the ABWR containment TS were modeled after NUREG-1433. These improved STS reflect the accumulated operating experience of currently operating light water reactors.

The proposed ABWR TSs were developed from the BWR/6 and the BWR/4 STS. The staff reviewed the proposed ABWR TS to confirm similarities between them and the STS, as appropriate. The staff then concentrated its review on the parts of the ABWR TS that are unique because of ABWR-specific design features. Dispositions of comments resulting from the staff review were incorporated into the proof-and-review ABWR TS. The proof-and-review ABWR TS were then issued to GE and made available to the staff for comment. Dispositions of comments from GE and the staff on the proof-and-review TS have been incorporated into the final ABWR TS. The final ABWR TS will be produced in the industry format and certified as accurate by GE.

In the DFSER, the staff stated that GE should prepare the relevant portions of the TS for review by the staff in the

WordPerfect 5.1 format. This was DFSER Open Item 16-1. The NRC TS staff assisted GE in developing

the ABWR proof-and-review TS in the WordPerfect 5.1 format. GE committed to maintain the ABWR TS in this format. This resolved Open Item 16-1.

16.2 Evaluation

The staff evaluated the ABWR TS to confirm that they will preserve the validity of the design plant SSAR by ensuring that ABWR plants will be operated within the required conditions bounded by the SSAR and with operable equipment that is essential to prevent accidents and to mitigate the consequences of accidents postulated in the SSAR.

In the DFSER, the staff noted that for the ABWR design, GE had attempted the use of a three-subsystem concept for the engineered safety feature (ESF) systems in order to facilitate maintenance on one subsystem by extending the completion times (CTs) for one inoperable subsystem from 7 days to 30 days. GE was asked to provide justification that the emergency core cooling systems design consists of three independent redundant subsystems. This was DFSER Open Item 16-3. GE stated that it could not provide this justification. Instead, it stated that the ABWR design includes additional redundancy, beyond that for currently operating BWR plants, for the ESF systems. The staff reviewed GE's response and determined that, although this added redundancy does not allow extending the CT for an inoperable system from 7 to 30 days, it does allow extending the associated CTs beyond those specified for ESF systems in currently operating BWR plants. The staff's finding on the relaxed CTs, ranging from 8 hours to 14 days, is based on probabilistic risk evaluation, engineering evaluation, and operating experience and judgment for various components and combinations of components. The staff noted that these evaluated configurations match the configurations of inoperable components delineated in the TS limiting conditions for operation. The rationale for the relaxed CTs is given in Chapter 19 of the SSAR and the TS bases. On the basis of the above, DFSER Open Item 16-3 is resolved.

The ABWR instrumentation and controls (I&C) systems design concept incorporates microprocessor-based digital technology and multiplexed fiber optic signal transmission. TS limits for the reactor protection system are based on a four-sensor channel design. This requires that the applicable portions of the BWR/6 STS be restructured to address the unique ABWR data collection and transmission design features. In the DFSER, the staff asked GE to submit the ABWR TS so that it could complete its review of the ABWR I&C TS. This was DSER Open Item 16-2.

Technical Specifications

As part of the resolution of DFSER Open Item 16-1 noted previously, the I&C TS were issued by the staff to GE for comment. After meetings during which GE's comments were resolved, the staff issued the ABWR TS to GE for certification. This resolved DFSER Open Item 16-2.

In the DFSER, the staff stated that the COL applicant should include plant- and site-specific information in the ABWR TS. This was DFSER COL Action Item 16-1. SSAR Chapter 16, "Technical Specifications," contains guidelines enabling the COL applicant to complete the plant- and site-specific portions of the TS on the basis of as-procured hardware and software. This is acceptable.

As part of the TS Improvement Program, the staff concluded that portions of STS Section 5.0, "Administrative Controls," could be relocated to licensee-controlled documents. This improvement was incorporated into the ABWR TS.

COL applicants will have to ensure that the portions of Section 5.0 relocated to licensee-controlled documents are controlled in accordance with an administrative control system acceptable to the staff. To complete its review of the ABWR TS, the staff performed an independent audit to verify the accuracy and completeness of the TS. The staff provided comments to GE in a series of letters dated February 2, 10, 14, and 16, 1994, which were addressed by GE in the certified final ABWR TS provided in SSAR Amendment 34.

16.3 Conclusion

The staff concludes that the ABWR TS are consistent with the regulatory guidance in the BWR/6 and BWR/4 STS and contain design-specific parameters and additional TS requirements considered appropriate by the staff. Therefore, the ABWR TS satisfy 10 CFR 50.34 and 50.36 and are acceptable.

17 QUALITY ASSURANCE

17.1 Quality Assurance During the Design Phase

17.1.1 General

Standard safety analysis report (SSAR) Chapter 17 describes the quality assurance (QA) program for the design phase of the advanced boiling water reactor (ABWR) and references GE Nuclear Energy (GE) QA topical report, NEDO-11209-04A, which the staff reviewed and found acceptable. SSAR Chapter 17 also provides additional QA information specifically applicable to the ABWR. The staff assessed GE's description of the QA program for the design phase of the ABWR to determine if it complies with the requirements of 10 CFR Part 50, Appendix B, and with applicable QA-related regulatory guides listed in Table 17.1 of this report.

The basis of the staff's review was Standard Review Plan (SRP) Section 17.1, which addresses both design and construction QA. The development and implementation of the construction QA program were identified in the draft final safety evaluation report (DFSER) as DFSER combined license (COL) Action Item 17.1.1-1. GE addressed this item in Amendment 31 of the SSAR (SSAR Section 17.0.1.1), which is acceptable to the staff. GE also included this action item in the SSAR.

17.1.2 Organization

The structure of the organization responsible for the design of the ABWR and for the establishment and execution of the design-phase QA program is shown in Figure 17.1 of this report. The line organizations have been assigned specific QA responsibilities, including both internal audits and audits of suppliers, to ensure compliance with the QA program. Audits conducted by GE's Nuclear Quality Assurance (NQA) organization are superimposed on these audits.

The General Manager of Nuclear Operations is responsible for ensuring that (1) the intent of GE's nuclear quality policy is reflected in its nuclear products and services, (2) a system is in place to independently assess the performance of organizations that affect the quality of these products and services, and (3) a system is in place to resolve issues that could affect GE's ability to satisfy its nuclear quality policy and other quality-related commitments.

NQA is a staff organization responsible for establishing the nuclear quality policy and procedures that are issued by the Vice President and General Manager of GE. NQA is also responsible for (1) auditing the various line organizations involved in the nuclear business and ensuring conformance

of these organizations' procedures and practices with applicable corporate and nuclear quality-related policy and procedures, (2) ensuring integration of the organizations' quality planning into an effective QA program, (3) participating in management review boards that operate independently of the design verification by the line organizations, and (4) specifying how the line organizations are to comply with the nuclear quality policy and procedures. For the ABWR design, NQA is responsible for coordinating and integrating the QA program as it relates to engineering and management of the project.

A quality council aids NQA in fulfilling its responsibilities. The council's responsibility is to ensure total quality system coverage, uniformity, consistency, and continuity and to eliminate system deficiencies. The Manager of NQA chairs the quality council. Members of this council, as shown in Figure 17.1, are the managers responsible for QA in each of the major nuclear organizations. The council provides these managers direct access to top-level management and acts as a forum for the review of quality problems and corrective actions.

The line organizations are responsible for planning and implementing the QA functions performed within their areas of responsibility so that each organization's QA program complies with the nuclear QA policy and procedures established by NQA. The individual QA managers report to their department-level management and have the organizational independence and authority to identify quality-related problems; initiate, recommend, or provide solutions pertaining to conditions adverse to quality; and verify implementation of such solutions.

GE and its major technical associates, Hitachi and Toshiba, are designing the ABWR. The lead responsibility to produce each specification (through the major purchasing specifications) and drawing is assigned to one design organization within GE, Hitachi, or Toshiba. The content of each of these common engineering documents is reviewed and approved by GE engineering personnel, and GE is responsible for the design and the supporting calculations and records for the ABWR.

GE engineering organizations are responsible for the U.S. ABWR design and design control by

- ensuring incorporation of applicable regulatory requirements, codes, standards, criteria, and design bases into the design
- ensuring incorporation of project design requirements into the design

Table 17.1 Quality assurance regulatory guide commitments

No.	Title	Revision	Date
1.8	Personnel Selection and Training	1	September 1975
1.26	Quality Group Classification, and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants	3	February 1976
1.28	Quality Assurance Program Requirements (Design and Construction), using NQA-1 and NQA-2*	3	August 1985
1.29	Seismic Design Classification	3	September 1978
1.30	Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment	0	August 1972
1.37	Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants*	0	March 1973
1.38	Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling of Items for Water-Cooled Nuclear Power Plants*	2	May 1977
1.39	Housekeeping Requirements for Water-Cooled Nuclear Power Plants	2	September 1977
1.58	Qualification of Nuclear Power Plant Inspection, Examination, and Testing Personnel*	**	
1.64	Quality Assurance Requirements for the Design of Nuclear Power Plants*	**	
1.74	Quality Assurance Terms and Definitions	**	
1.88	Collection, Storage, and Maintenance of Nuclear Power Plant Quality Assurance Records*	**	
1.94	Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants	1	April 1976
1.116	Quality Assurance Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems*	0-R	June 1976
1.123	Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants*	**	
1.144	Auditing of Quality Assurance Programs for Nuclear Power Plants	**	
1.146	Qualification of Quality Assurance Program Audit of Personnel for Nuclear Power Plants*	**	

* NRC accepted the GE positions given in Topical Report NEDO-11209-04A, Revision 8, March 31, 1989.

** Superseded by Revision 3 of Regulatory Guide 1.28.

Figure 17-1 ABWR design organization structure

Quality Assurance

- translating the design information into the appropriate design documents
- verifying the design adequacy either through independent design review, the use of alternative or simplified calculational methods, or the performance of a suitable testing program
- coordinating design activities among interfacing design engineers and design organizations
- reviewing, approving, issuing, and distributing design documents under a controlled document system
- controlling design changes and changes to design documents in accordance with documented procedures
- providing for the retention, storage, control, and retrievability of design record documents
- taking corrective action as necessary to correct design errors and to improve the design control function

17.1.3 Quality Assurance Program

GE structured its nuclear QA program to satisfy Appendix B to 10 CFR Part 50 and the provisions of the applicable Nuclear Regulatory Commission (NRC) regulatory guides identified in Table 17.1 of this report. GE uses this QA program to control its design of the ABWR. GE has written policies, procedures, and instructions to implement the program. These documents control quality-related activities in accordance with the requirements of Appendix B to 10 CFR Part 50 and with applicable regulations, codes, and standards. The GE QA organizations ensure that procedures and instructions are provided for meeting the QA requirements. In addition, QA personnel conduct reviews and audits to verify the effective implementation of the program.

GE's nuclear QA program requires that implementing documents encompass detailed controls for (1) translating codes, standards, regulatory requirements, technical specifications, engineering requirements, and process requirements into drawings, specifications, procedures, and instructions; (2) developing, reviewing, and approving procurement documents and changes thereto; (3) prescribing all quality-related activities by documented instructions, procedures, drawings, and specifications; (4) issuing and distributing approved documents; (5) purchasing items and services; (6) identifying materials, parts, and components; (7) performing special processes; (8) inspecting and/or testing materials, equipment, processes, and services; (9) calibrating and maintaining measuring and test equipment; (10) handling, storing, and shipping items;

(11) identifying the inspection, test, and operating status of items; (12) identifying and dispositioning nonconforming items; (13) correcting conditions adverse to quality; (14) preparing and maintaining QA records; and (15) auditing activities that affect quality.

Training and experience requirements are defined for each position in the GE organization. In addition, GE indoctrinates and trains personnel performing activities affecting quality to ensure that appropriate proficiency is achieved and maintained and that personnel responsible for quality-related activities are instructed as to the purpose, scope, and implementation of the quality-related manuals, instructions, and procedures. The indoctrination and training are carried out through documented procedures, on-the-job training, personal contacts, and meetings.

The ABWR design and changes to it are formally verified. Design verification is a process for an independent review of the design against design requirements to confirm that the designer's methods and conclusions are consistent with requirements and that the resulting design is adequate for its specified purpose. Design verification is performed and documented by persons other than those responsible for the design, using the method specified by the design organization. Designs are verified by one or more of the following methods: design review, qualification testing, alternative or simplified calculations, or checking. Team design reviews are continuing reviews of a design, selected by engineering management, to evaluate design adequacy that includes concepts, the design process, methods, analytical models, criteria, materials, applications, or development programs. When appropriate, team design reviews are used to verify that product designs meet functional, contractual, safety, regulatory, industrial codes and standards, and GE requirements. The selection of the design review team depends on the product design and the type of review. The technical competence of the members of each team encompasses three broad categories: (1) those with broad experience on similar products; (2) those with specialized technical expertise such as in heat transfer, materials, and structural analysis; and (3) those with a functional expertise such as QA, manufacturing, engineering, and product service.

For the international ABWR design, the lead design organization prepares the common engineering design document and circulates it internally for engineering review, approval, and design verification. Evidence of verification is entered into the design records of the responsible design organization. Each document is distributed to the design organizations of the other parties for their review and approval of technical content and design interfaces. All comments resulting from this process are resolved. After the comments are resolved,

the design verification is reviewed and, when necessary, updated to ensure that changes did not invalidate the original verification. After final agreement is reached, the document is finalized by the lead design organization, circulated to the other parties for their approval signatures, and issued. Changes to ABWR documents are handled similarly. Differences between international and U.S. ABWR designs are identified in a controlled list (called the design action list) for future design action and application.

GE's QA organizations are responsible for establishing and implementing the audit program. Audits are performed in accordance with preestablished written checklists by qualified personnel not having direct responsibilities in the areas being audited. Periodic audits are performed to evaluate all aspects of the QA program, including the effectiveness of implementation. The QA program requires the review of audit results by the person having responsibility in the area audited to determine and take corrective action where necessary.

Followup audits are performed to determine if nonconformances and deficiencies have been effectively corrected and the corrective action precludes repetitive occurrences. Audit reviews, which indicate performance trends and the effectiveness of the QA program, are reported to responsible management for review and assessment.

The staff concludes that GE's QA program for the design phase of the ABWR describes requirements, procedures, and controls that, when properly implemented, will comply with the QA requirements of Appendix B to 10 CFR Part 50, applicable QA-related regulatory guides, and the acceptance criteria in SRP Section 17.1, related to QA during design and construction. In addition, SSAR Chapter 17 references GE QA topical report, NEDO-11209-04A, which the staff has reviewed and found acceptable.

During its review of the QA program described in the ABWR SSAR, the staff audited the implementation of the program at GE's offices in San Jose, California, during the week of February 6, 1989. The report of this audit is in the Commission's Public Document Room, the Gelman Building, 2120 L Street NW, Washington, DC. On the basis of the sample of design activities audited, which included Hitachi and Toshiba documents requested by the staff and translated into English, the auditors concluded that the design QA programs implemented by GE, Hitachi, and Toshiba met the applicable requirements of Appendix B to 10 CFR Part 50 and were acceptable for designing the ABWR.

An inspection of the ABWR design process was performed from September 7 through 10, 1993. The inspection results are documented in NRC Inspection Report 99900403/93-02. The inspection scope included an examination of GE QA controls applied to the ABWR project. This included a review of design record files (DRFs), selected computer codes used for accident analysis and transient modeling, test activities, design calculations, and audits. The inspection questioned the technical adequacy of supporting calculations generated by the international technical associates (TAs). Some test data for the Full Integral Simulation Test could not be retrieved by GE, and some calculation notebooks were poorly maintained. The staff evaluation of GE's response to the findings of that inspection was Open Item F17.1.3-1.

GE provided a response to the staff's inspection report on November 24, 1993, which addressed the items of concern and proposed corrective and preventive actions such as: verifying the accuracy of an input parameter for a LOCA analysis and performing related sensitivity studies, disseminating training reminders to technical staff about the QA requirements for design analysis and DRFs, increasing the GE audit emphasis on the content of DRFs, verifying that installed test instrumentation was within specified tolerances, supplementing transient analysis code DRFs, confirming that engineering services were provided under the auspices of an Appendix B quality program, correcting SSAR inaccuracies, and performing design verification on a design calculation. The staff found these proposed actions to be acceptable with a few exceptions. A request for further information and clarification was sent to GE on December 22, 1993, for the issues involving the technical oversight by GE of supporting calculations generated by the TAs and the conduct of computer code design verification. GE's response dated January 17, 1994, was found to be acceptable with one exception discussed as follows.

During the course of the inspection in September 1993, the staff identified that the common engineering documents (design specifications, process flow diagrams (PFDs), instrument block diagrams (IBDs), and piping and instrument diagrams (P&IDs)) have received a considerable level of GE design review. However, the level of GE review performed on the supporting calculations generated by the international TAs was not found to be rigorous. For example, the NRC inspection found that the depth of technical review afforded by the GE program reviews (QA audits) was minimal as the audit teams had not been supplemented by technical reviewers. In addition, little documented evidence was found in the DRFs to substantiate GE's review of the supporting calculations.

Quality Assurance

GE informed the staff that a sufficient level of confidence was obtained in the supporting calculations through the performance of GE program reviews of each TA, the GE engineering reviews of the common engineering documents, and participation by GE staff in numerous design review meetings. In addition, GE provided amplifying information during meetings with the staff on March 14 and 15, 1994, with respect to the extensive GE involvement during the ABWR design evolution. GE stated that, during the period from 1978 through 1985, extensive technical interaction transpired between GE and the TAs.

On March 22 through 24, 1994, a second NRC inspection was performed to substantiate the extent of the GE technical oversight of the TA's supporting design and analysis efforts. The inspection spanned a representative sampling of ABWR systems for which a TA had lead design responsibility. The staff examined the associated GE DRFs, interviewed cognizant GE design engineers, reviewed engineering correspondence from the TAs, and searched for examples of GE verification of TA calculations.

The three-day inspection resulted in the identification of evidence of GE's technical oversight of the supporting design as documented by the Phase 3 "Advanced BWR Plant Evaluation Report," GE comparisons of the ABWR design parameters with respect to the BWR 5 and 6 plant designs, thorough GE review of the common engineering documents that included proposed design revisions and independent GE calculations, the existence of selected TA supporting calculations in the GE DRFs, and GE review of system analysis, system performance, and capacity calculations generated by the TAs.

The inspection determined that reasonable assurance was provided by the depth, extent, and duration of the GE technical oversight of the joint design process to resolve the remaining issue from the September 1993 inspection. During the March 1994 inspection the staff additionally reviewed selected GE corrective and preventive actions that had been implemented in response to other concerns raised during the September 1993 inspection and found them satisfactory. Therefore, Open Item F17.1.3-1 is resolved based on the March 1994 inspection findings and the corrective and preventive measures instituted by GE in response to the QA and design control concerns identified in NRC Inspection Report 99900403/93-02.

An applicant for a COL, when completing its detailed design and equipment selection during the COL design phase, will submit its QA program for the design phase for staff review. This will be in addition to the staff review of the COL applicant's QA program for both the construction

and the operation of the facility. When the COL applicant's QA programs are submitted, whether they are the GE QA programs augmented with information by the COL applicant, or a completely new QA program, the staff will perform the necessary reviews to ensure compliance to 10 CFR Part 50, Appendix B. This was DFSEER COL Action Item 17.1.1-1.

17.2 Quality Assurance During the Operations Phase

The operations QA program is beyond the scope of GE's application for design certification (DC) and was identified by the staff in the DFSEER as DFSEER COL Action Item 17.2-1. This item was addressed by GE Amendment 31 of the SSAR (SSAR Section 17.0.1.1), which is acceptable to the staff. GE has also included this action item in the SSAR. For a discussion on the relationship of the COL applicant's Operational Reliability Assurance Process to the QA program, see Section 17.3 of this report.

17.3 Reliability Assurance Program

Introduction

SSAR Section 17.3 describes the reliability assurance program (RAP) for the design phase of the ABWR. GE implements the design reliability assurance program (D-RAP) for its scope of design during detailed design and specific equipment selection phases to ensure that the important ABWR reliability assumptions of the probabilistic risk assessment (PRA) will be considered throughout plant life. The COL applicant will augment and implement the D-RAP for its scope of design and equipment selection (See SSAR 17.3.13). Additionally, the COL applicant should develop and implement a process whose objectives are to monitor equipment performance and evaluate equipment reliability to provide reasonable assurance that the plant is operated and maintained commensurate with PRA assumptions so that the overall safety is not unknowingly degraded and remains within acceptable limits (See SSAR 17.3.13). This process could be described as an operational reliability assurance process (O-RAP) that should be included under existing programs for quality assurance and maintenance. When structures, systems, and components (SSCs) monitoring and evaluation identifies performance or condition problems, appropriate corrective action will be taken to ensure SSCs remain capable of performing their intended functions. However, the RAP does not attempt to statistically verify the numeric values used in the PRA through performance monitoring.

The staff has evaluated SSAR Chapter 17.3, which included the GE response (dated March 5, 1992) to the

staff's request for additional information (RAI) contained in its request for resolution of issues related to SSAR Chapter 19, Appendix K (letter dated September 19, 1991). The acceptability of the particular method to identify and prioritize risk-significant structures, systems, and components (SSCs) associated with PRA was reviewed as part of SSAR Chapter 19 and is addressed in Chapter 19 of this report.

Background

The need for a safety-oriented reliability effort for the nuclear industry was identified by the NRC in Three Mile Island (TMI) Action (NUREG-0660) Item II.C.4. Subsequently, initial NRC research in the area of reliability assurance began in the early 1980s. The results of this research showed that an operational reliability program based on a feedback process of monitoring performance, identifying problems, taking corrective action, and verifying the effectiveness of these actions was needed and that other NRC initiatives (e.g., maintenance inspections, performance indicators, aging programs, and technical specification improvements) would address this need. The NRC concluded from this research that an operational reliability program could be implemented most effectively in a performance-based, nonprescriptive regulation, where NRC mandates the level of safety performance to be achieved. For example, licensees could be required to set availability and reliability targets for selected systems and to measure performance compared to the targets.

The TMI item was closed out for operating reactors in October 1988 without further action because several NRC initiatives had effectively subsumed the operational reliability program effort. These initiatives included efforts to (1) improve maintenance and better manage the effects of aging, (2) improve technical specifications, (3) develop and use plant performance indicators, and (4) develop an operational reliability program as an acceptable means of meeting the station blackout rule (10 CFR 50.63).

NUREG-1070, "NRC Policy on Future Reactor Designs" included the concept of a systems reliability program to ensure that the reliability of components and systems important to safety would remain at a sufficient level. To ensure that reliability objectives are met and to prevent degradation of reliability during operation, the NRC envisioned that the PRA, performed at the design stage, would be used as a tool in making detailed design decisions affecting procurement, testing, and the formulation of operations and maintenance procedures.

In a few specific instances, the NRC is studying or has established reliability targets for systems and components.

For example, SRP Section 10.4.9 requires that an acceptable auxiliary feedwater system design have an unreliability in the range of $10E-4$ to $10E-5$ per demand. The resolution of Generic Issue B-56 involves efforts to determine, monitor, and maintain emergency diesel generator reliability levels. Additional regulatory bases for key elements of a RAP can be found in Appendices A and B to 10 CFR Part 50 and 10 CFR 50.65.

In SECY-89-013, "Design Requirements Related to the Evolutionary Advanced Light Water Reactors," dated January 19, 1989, the staff identified several issues for next-generation light water reactors that go beyond present acceptance criteria defined in the SRP. RAP, as one of these issues, was defined as a program to ensure that the design reliability of safety-significant SSCs is maintained over the life of a plant. In SECY-89-013, the staff informed the Commission that RAP would be required for final design approval or design certification (DC). In November 1989, potential applicants for DC were informed by letter that "the NRC staff was considering matters that went beyond the current SRP . . . that [the NRC] expects these advanced reactor designs to embody." Reliability assurance was identified as one of these matters.

In SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs," dated April 2, 1993, the staff recommended that the Commission approve its interim position that a high-level commitment to a RAP application to design certification be required as a non-system generic Tier 1 requirement with no associated inspections, tests, analyses, and acceptance criteria (ITAAC). The details of the D-RAP, including the conceptual framework, program structure, and essential elements, should be provided in the SSAR. The SSAR for the D-RAP should also (1) identify and prioritize a list of risk-significant SSCs based on the DC PRA and other sources; (2) ensure that the vendor's design organization determines that significant design assumptions, such as equipment that satisfies the design reliability and unavailability, are realistic and achievable; (3) provide input to the procurement process for obtaining equipment that satisfies the design reliability assumptions; and (4) provide these design assumptions as input to the COL applicant for consideration in the O-RAP. A COL applicant would augment the D-RAP with site-specific design information and would implement the balance of the D-RAP, including input to the procurement process (See SSAR 17.3.13).

The RAP consists of two distinct parts: (1) D-RAP and (2) O-RAP. D-RAP involves a top-level program at the design stage that is used to define the scope, conceptual framework, and essential elements of an effective RAP. D-RAP is also used to implement those aspects of the

Quality Assurance

program that are applicable to the design process. In addition, D-RAP is used to identify the relevant aspects of plant operation, maintenance, and performance monitoring for the risk-significant SSCs for the operator's consideration in developing an O-RAP. The O-RAP objectives should be incorporated into existing programs (i.e., quality assurance and maintenance) that will be used to monitor equipment performance and evaluate equipment reliability to provide reasonable assurance that the plant is operated and maintained commensurate with PRA assumptions so that the overall safety is not unknowingly degraded and remains within acceptable limits. When SSC monitoring and evaluation identifies performance or condition problems, appropriate corrective action will be taken to assure SSCs remain capable of performing their intended functions. However, the RAP does not attempt to statistically verify the numerical values used in the PRA through performance monitoring.

The staff's final position on RAP was presented in the Commission paper on the Regulatory Treatment of Non-Safety Systems (RTNSS) SECY-94-084 dated March 28, 1994. The Commission approved the following applicable regulation for D-RAP:

An application for design certification or for a combined license must contain:

- (a) the description of the reliability assurance program used during the design that includes, scope, purpose, and objectives;
- (b) the process used to evaluate and prioritize the structures, systems, and components in the design, based on their degree of risk-significance;
- (c) a list of structures, systems, and components designated as risk-significant; and
- (d) for those structures, systems, and components designated as risk-significant:
 - (i) A process to determine dominant failure modes that considered industry experience, analytical models, and applicable requirements; and
 - (ii) Key assumptions and risk insights from probabilistic, deterministic and other methods that considered operation, maintenance and monitoring activities.

Each COL applicant that references the ABWR design must implement the D-RAP approved by the NRC.

The staff evaluated the SSAR on the basis of the applicable regulation stated above.

A COL applicant would augment the designer's RAP to reflect plant-specific information and implement those elements applicable during the construction and operation phases. The staff's COL application review will be similar to the design certification review and include an evaluation of the updated (site-specific) PRA, probabilistic, deterministic and other insights (e.g., operating experience) to assess any changes to risk-significant SSCs and site-specific vulnerabilities. The staff will review the COL applicant's proposed design reliability assurance program plan to determine if it satisfies the above requirements at the time of the COL application.

Evaluation

In its RAI dated September 19, 1991, the staff stated that the GE ABWR RAP submittal should (1) describe the basic framework of a RAP including the scope, purpose, objective, basic definitions, and elements (RAI Question (Q)1); (2) when describing the RAP concepts and elements, include a discussion on performance goal or targets, problem prediction and recognition, problem prioritization and correction, and problem closeout (RAI Q2); (3) describe how RAP will address plant aging concerns (RAI Q3); (4) describe the organizational and administrative aspects for implementing an effective RAP (RAI Q4); (5) describe the approach for providing feedback to the designer when actual plant performance data consistently differ from the designers PRA and RAP assumptions (RAI Q5); (6) describe the major programmatic interface between the RAP and areas such as design, construction, startup testing, operations, maintenance, engineering, safety, licensing, quality assurance, and procurement (RAI Q6); and (7) provide an example of how the GE RAP would be implemented using a specific SSC identified as risk significant in the PRA (RAI Q7). These questions included the use of the term "RAP;" however, the intent was for the questions to apply to that portion of the RAP that GE is responsible for preparing and implementing (e.g., the ABWR D-RAP). The RAI questions provided an outline of the staff's expectations on RAP for evolutionary designs and explicitly stated the details of these expectations.

In its letter dated March 5, 1992, GE stated that a new SSAR Section 17.3 would address the staff's RAI questions. After reviewing the initial GE response and the modifications to SSAR Section 17.3 in Amendment 26

(dated March 24, 1993) the staff determined that (1) RAI Q1 was answered by Sections 17.3.2, 17.3.3, 17.3.4, 17.3.6, 17.3.7, and 17.3.8; (2) RAI Q2 was answered by Sections 17.3.6, 17.3.9, and 17.3.10; (3) RAI Q3 was answered by Section 17.3.10; (4) RAI Q4 was answered by Section 17.3.5; (5) RAI Q5 was answered by Section 17.3.10; (6) RAI Q6 was answered by Section 17.3.10; and (7) RAI Q7 was answered by Section 17.3.11. Therefore, the answers to the RAI questions contained in SSAR Section 17.3 address the staff's expectations on RAP for evolutionary designs and explicitly state GE's response to these expectations. The details of the staff's evaluation are presented in Sections 17.3.1 through 17.3.11 below.

17.3.1 General

RAP for the design phase of the ABWR described in SSAR Section 17.3 ensures that the important ABWR reliability assumptions of the PRA will be considered throughout plant life. The PRA is used to evaluate the anticipated plant response to initiating events. The PRA evaluates the plant response to initiating events to substantiate, in part, that plant damage has a very low probability and risk to the public is very low. Input to the PRA includes details of the plant design and assumptions about the reliability of the plant risk-significant SSCs. Changes to equipment and system reliabilities will be reevaluated and as necessary will be reflected in a revised PRA throughout plant life. GE started the D-RAP during design of the ABWR. The COL applicant will utilize the ABWR D-RAP during the detailed design and specific equipment selection phases to complete the D-RAP. The COL applicant will complete the D-RAP and will also incorporate the objectives of the O-RAP into existing programs (i.e., maintenance and quality assurance) that will be used to monitor equipment performance and evaluate equipment reliability to provide reasonable assurance that the plant is operated and maintained commensurate with PRA assumptions so that the overall safety is not unknowingly degraded and remains within acceptable limits. When SSC monitoring and evaluation identifies performance or condition problems, appropriate corrective action will be taken to assure SSCs remain capable of performing their intended functions.

GE states that the D-RAP will include the design evaluation of the ABWR. It will be used to identify relevant aspects of plant operation, maintenance, and performance monitoring of important plant SSCs for the COL applicant's consideration in ensuring safety of the equipment and limited risk to the public. The COL applicant will specify the policy and implementation procedures for using D-RAP information provided by GE

(See SSAR 17.3.1-13). SSAR Section 17.3 also includes a descriptive example of how the D-RAP will apply to the standby liquid control system (SLCS). This example shows how the principles of D-RAP will be applied to other systems identified by the PRA as significant with regard to risk.

The staff concludes that Section 17.3.1 of the SSAR meets the requirement of the applicable regulation for D-RAP to provide a description of the RAP used during the initial design as discussed above in Section 17.3 of this report and is acceptable.

17.3.2 Scope

In response to the part of RAI Q1 associated with the RAP scope, GE provided the additional information in Section 17.3.2 of the SSAR. In that section, GE states that the scope of the ABWR D-RAP includes identifying relevant aspects of plant operation, maintenance and monitoring of plant risk-significant SSCs. The PRA and other industry sources are used to identify and prioritize those SSCs that are important to prevent or mitigate plant transients or other events that could present a risk to the public.

The staff reviewed SSAR Section 17.3.2 with respect to the scope of the ABWR D-RAP and concludes that it is responsive to the staff's RAI question, meets the requirement of the applicable regulation for D-RAP to include the scope of the RAP as described in Section 17.3 of this report and is acceptable.

17.3.3 Purpose

In response to the part of RAI Q1 associated with the RAP purpose, GE provided the additional information in Section 17.3.3 of the SSAR. In that section, GE states that the purpose of the ABWR D-RAP is to ensure that plant safety, as estimated by the PRA, is maintained as the detailed design evolves through the implementation and procurement phases. Additionally, GE states that pertinent information is to be provided in the design documentation to the COL applicant so that equipment reliability, as it affects plant safety, can be maintained through operation and maintenance during the entire plant life.

The staff reviewed SSAR Section 17.3.3 with respect to the purpose of the ABWR D-RAP and concludes that it is responsive to the staff's RAI question, meets the requirement of the applicable regulation for D-RAP to include the purpose of the RAP as described in Section 17.3 of this report and is acceptable.

Quality Assurance

17.3.4 Objective

In response to the part of RAI Q1 associated with the RAP objective, GE provided the additional information in Section 17.3.4 of the SSAR. In that section, GE states that the objective of the ABWR D-RAP is to identify those plant SSCs that are significant contributors to risk, as shown by the PRA or other sources, and to ensure that, during the implementation phase, the plant design continues to utilize risk-significant SSCs whose reliability is commensurate with the PRA assumptions. The D-RAP also will be used to identify key assumptions regarding any operation, maintenance, and monitoring activities that the COL applicant should consider in developing its approach to implementing an O-RAP using existing programs such as quality assurance and maintenance to provide reasonable assurance that such SSCs can be expected to operate with reliability commensurate with that assumed in the PRA. A major factor in the ABWR D-RAP is risk-focused maintenance that considers all plant modes and equipment directly relied on in ABWR emergency operating procedures (EOPs).

The staff reviewed SSAR Section 17.3.4 with respect to the objective of the ABWR D-RAP and concludes that it is responsive to the staff's RAI question, meets the requirement of the applicable regulation for D-RAP to include the objective of the RAP as described in Section 17.3 of this report and is acceptable.

17.3.5 GE Organization for D-RAP

In response to RAI Q4 associated with the RAP organizational aspects, GE provided the additional information in Section 17.3.5 of the SSAR dated March 5, 1992. However, in its DFSER, the staff identified an inconsistency between the SSAR Section 17.3.5 narrative and SSAR Figure 17.3-1. Additionally, GE described the D-RAP organizational structure in the future tense in SSAR Section 17.3.5. This item remained open subject to a revision of SSAR Section 17.3.5 (DFSER Open Item 17.3.5-1). On January 18, 1993, GE submitted a markup to SSAR Section 17.3.5 that satisfactorily addressed DFSER Open Item 17.3.5.1 and deleted the organization chart (SSAR Figure 17.3-1). This markup was incorporated into SSAR Section 17.3.5 as Amendment 26 dated March 24, 1993. GE has also included this information in the SSAR. On the basis of this evaluation, this item is resolved.

In SSAR Section 17.3.5, GE states that the reliability analyses and the PRA, including SSAR Appendix 19K, were performed by GE. GE also developed the D-RAP definition. Responsibility for the design of key equipment,

components, and subsystems was shared by GE and external organizations, including the organization performing architect-engineering functions. The GE manager assigned the responsibility of managing and integrating the D-RAP program had direct access to the ABWR project manager and kept him abreast of D-RAP critical items, program needs, and status. He had organizational freedom to (1) identify D-RAP problems; (2) initiate, recommend, or provide solutions to problems through designated organizations; (3) verify implementation of the solution; and (4) function as an integral part of the final design process.

The staff reviewed Amendment 26 of SSAR Section 17.3.5, which incorporated the markup submitted on January 18, 1993, with respect to the GE organizational description and accountability for implementing the ABWR D-RAP for DC, and concludes that it is responsive to the staff's DFSER open item concern, is responsive to the staff's original RAI Q4, satisfies the staff position that the D-RAP ensures that the vendor's design organization determine that significant design assumptions are realistic and achievable as discussed in SECY-93-087 and Section 17.3 of this report, and is acceptable. The staff also concludes that the GE organizational description and accountability for implementing the ABWR D-RAP were acceptable and resolved DFSER Open Item 17.3.5-1.

A COL applicant completing its detailed design and equipment selection during the COL design phase will submit its specific D-RAP organization for staff review (See SSAR 17.3.13).

17.3.6 SSC Identification and Prioritization

In response to the part of RAI Q1 associated with the RAP definitions and elements and the part of RAI Q2 associated with the RAP scope, GE provided the additional information in Section 17.3.6 of the SSAR.

In SSAR Section 17.3.6, GE states that the PRA prepared for the ABWR will be the primary source for identifying risk-significant SSCs that should be given special consideration during detailed design and procurement phases and considered for inclusion in the COL applicant's O-RAP. It also is possible that risk-significant SSCs will be identified from sources other than the PRA, such as nuclear plant operating experience, other industrial experience, and relevant component failure databases. SSAR Chapter 19 describes the method of identifying risk-significant SSCs using the PRA.

The staff reviewed SSAR Section 17.3.6 with respect to identifying and prioritizing risk-significant SSCs for the ABWR D-RAP and concludes that it is responsive to the

staff's RAI questions, meets the requirement of the applicable regulation for D-RAP to describe the methodology used to evaluate and prioritize risk significant SSCs as described in Section 17.3 of this report, and is acceptable. The acceptability of the PRA methods or techniques to prioritize SSCs is addressed in Chapter 19 of this report.

17.3.7 Design Considerations

In response to the part of RAI Q1 associated with the RAP definitions and elements, GE provided the additional information in Section 17.3.7 of the SSAR. In SSAR Section 17.3.7, GE states that the reliability of SSCs identified by the PRA as risk significant will be evaluated at the detailed design stage by appropriate design reviews and reliability analysis. Current databases will be used to identify appropriate values for failure rates of equipment as designed, and these failure rates will be compared with those used in the PRA. Normally, the failure rates will be similar, but in some cases they may differ because of recent design or database changes. Whenever failure rates of designed equipment are significantly greater than those assumed in the PRA, an evaluation will be performed to determine if the equipment is acceptable or if it must be redesigned to achieve a lower failure rate.

For those SSCs identified by the PRA or other sources as risk significant redesign will be considered as a way to reduce the core damage frequency (CDF) contribution. If there are practical ways to redesign a risk-significant SSC, it will be redesigned and the change in system fault tree results will be calculated. Following the redesign phase, dominant SSC failure modes will be identified so that protection against such failure modes can be accomplished by appropriate activities during plant life. (See Chapter 19 of this report.)

For the COL applicant, GE will identify in the PRA, or other design documents, the risk-significant SSCs and the associated reliability assumptions, including any pertinent bases and uncertainties considered in the PRA (See Chapter 19 of this report). GE will also provide this information for the COL applicant to consider in developing an O-RAP to help assure that PRA results will be achieved over the life of the plant. The COL applicant can use this information for establishing appropriate reliability targets and the associated maintenance practices for achieving them.

The staff reviewed SSAR Sections 17.3.6, 17.3.7, and 17.3.8 and concludes that GE has provided a process for evaluating risk-significant SSCs for redesign and a process for providing information to a COL applicant for establishing appropriate reliability targets and the

associated maintenance practices for an O-RAP. The staff also concludes that SSAR Section 17.3.7 is responsive to its RAI question, meets the requirement of the applicable regulation for D-RAP to describe the methodology used to evaluate and prioritize risk significant SSCs as described in Section 17.3 of this report, and is acceptable.

17.3.8 Defining Failure Modes

In response to the part of RAI Q1 associated with the RAP definitions and elements, GE provided the additional information in Section 17.3.8 of the SSAR. In SSAR Section 17.3.8, GE uses the methodology of NUREG/CR-5695, Section 5, to determine dominant failure modes of risk-significant SSCs in the D-RAP. The method includes using historical information, analytical models, and existing requirements.

The staff reviewed SSAR Sections 17.3.6, 17.3.7, and 17.3.8 and concludes that GE has provided a method for determining dominant failure modes for risk-significant SSCs in the ABWR D-RAP. The staff also concludes that SSAR Section 17.3.8 is responsive to its RAI question, meets the requirement of the applicable regulation for D-RAP to define failure modes as described in Section 17.3 of this report, and is acceptable.

17.3.9 Operational Reliability Assurance Activities

In response to the part of RAI Q2 associated with the RAP performance goals and targets, problem prediction, and problem recognition, GE provided the additional information in Section 17.3.9 of the SSAR. In SSAR Section 17.3.9, GE states that once the dominant failure modes are determined for risk-significant SSCs, an assessment is required to determine suggested O-RAP activities that will ensure acceptable performance during plant life. Such activities may consist of periodic surveillance inspections or tests, monitoring of SSC performance, or periodic preventive maintenance (PM).

Periodic testing of SSCs may include startup of standby systems, surveillance testing of instrument circuits to ensure that they will respond to appropriate signals, and inspection of passive SSCs to show that they are available to perform as designed. Performance monitoring, including condition monitoring, can consist of measurements of output, measurement of magnitude of an important variable, and testing for abnormal conditions. Periodic PM will be performed at regular intervals to preclude problems that could occur before the next PM interval.

Planned maintenance activities will be integrated with regular operating plans. Maintenance that will be

Quality Assurance

performed more frequently than during refueling outages must be planned to avoid disrupting safe operation or causing a reactor scram, engineered safety feature actuation, or abnormal transient. Maintenance performed during refueling outages must not adversely affect plant safety.

The staff reviewed SSAR Sections 17.3.9 and 17.3.10 and concludes that GE has provided a process to determine operational reliability assurance activities using the dominant failure modes identified in the ABWR D-RAP. The staff also concludes that SSAR Section 17.3.9 is responsive to its RAI question, to include a description of O-RAP activities, and is acceptable. The COL applicant should incorporate O-RAP activities into existing programs such as maintenance and quality assurance and provide the staff with a description of how these activities are met at the time of the COL application. (See SSAR 17.3.13).

17.3.10 COL Applicant's Reliability Assurance Process

In response to the part of RAI Q2 associated with the RAP performance goals and targets, problem prediction, and recognition, problem prioritization and correction, and problem closeout, GE provided the additional information in Section 17.3.10 of the SSAR. Additionally, the O-RAP description of plant aging, feedback to designer, and programmatic interfaces responds to RAI Q3, Q5, and Q6, respectively. In SSAR Section 17.3.10, GE states that the O-RAP will be prepared and implemented by the COL applicant, using the information provided by GE. The information will help the COL applicant determine activities that should be included in the O-RAP. Examples of activities that might be included in an O-RAP are:

- reliability performance monitoring
- reliability methodology
- problem prioritization
- root cause analysis
- corrective action determination
- corrective action implementation
- corrective action verification
- plant aging
- feedback to designer
- programmatic interfaces

The COL applicant will address in its O-RAP the interfaces with construction, startup testing, operations, maintenance, engineering, safety, licensing, QA, and procurement of replacement equipment.

The staff concludes that the outline of an O-RAP provided by GE to be used by the COL applicant in SSAR Section 17.3.10 is responsive to its RAI questions, to include a description of O-RAP activities, and is acceptable. The COL applicant should also incorporate O-RAP activities into existing programs such as maintenance and quality assurance and provide the staff with a description of how these activities are met. The COL applicant will provide the D-RAP for completion of the detailed design and specific equipment selection phases (e.g., procurement of risk-significant SSCs), and a complete O-RAP to be reviewed by the staff as described in Sections 17.3.1 and 17.3.9 of this report.

17.3.11 D-RAP Implementation

In response to RAI Q7, GE provided in Section 17.3.11 of the SSAR, an example of how the GE RAP would be implemented using a specific SSC identified as risk significant in the PRA. For example purposes only, the SLCS was assumed to be a significant contributor to CDF or to offsite risk. The system description (including operation and differences from current BWRs) and a system fault tree were provided in the example. Seven SLCS risk-significant components identified in the example as having high importance in the SLCS fault tree were considered for redesign. Also, failure modes and maintenance requirements for the seven components were identified.

The staff concludes that the SSAR Section 17.3.11 example using the SLCS satisfactorily demonstrated GE's cognitive understanding of the RAP concept and their ability to incorporate it into the design, is responsive to its RAI question, and is acceptable. The process description for the implementation of the ABWR D-RAP by a COL applicant will be reviewed by the staff at the time a COL application is submitted.

17.3.12 Conclusion

The staff has reviewed Section 17.3 of the ABWR SSAR. The staff finds that the ABWR SSAR satisfies the requirements of the applicable regulation for D-RAP for the ABWR design phase of the reliability assurance program as described in Section 17.3 of this report, and is therefore acceptable.

18 HUMAN FACTORS ENGINEERING

To perform its evaluation of Standard Safety Analysis Report (SSAR) Chapter 18, the staff reviewed the information described in Section 18.1.2 of this report. The review was based on the current regulatory requirements in 10 CFR 52.47, 10 CFR 50.34(g), and 10 CFR 50.34(f) and the guidance in Standard Review Plan (SRP) Sections 13 and 18; NUREG-0700, "Guidelines for Control Room Design Review," September 1981; and NUREG-0933, "Prioritization of Generic Safety Issues Main Report," April 1983. The staff developed additional review criteria to provide a basis for its review of aspects of the advanced boiling water reactor (ABWR) human factors engineering (HFE) program that were not fully addressed by the previously mentioned documents. These criteria are contained in the staff's "Human Factors Engineering Program Review Model (PRM) and Acceptance Criteria for Evolutionary Reactors" which was forwarded to the Commission in SECY-92-299 dated August 27, 1992, and is attached as Appendix J to this report.

Section 18.1 of this report describes the methodology used to conduct the review, including the development of general review criteria that supplement the regulatory requirements and established guidelines. The results described in Sections 18.2 through 18.9 address the following eight major topics:

- design goals
- main control room (CR) standard design features
- inventory of controls, displays, and alarms
- remote shutdown system (RSS)
- local valve position indication (VPI)
- unresolved and generic safety issues
- emergency procedure guidelines (EPGs)
- design and implementation process

Section 18.10 gives a summary of the evaluation findings and overall conclusions.

As a result of the staff's initial review of the SSAR, many outstanding issues were identified and documented in the draft safety evaluation report (DSER) (SECY-91-320), and subsequently in the DFSER (SECY-92-349). One of the major issues to emerge from the initial review was that detailed human-system interface (HSI) information concerning the final design was not available for staff review as part of the design certification evaluation. GE's HSI analysis and design efforts resulted in a list of key CR design features characterized at a general level (not a detailed specification) and a minimum inventory of fixed safety-significant information and control requirements derived from an analysis of the ABWR EPGs and probabilistic risk assessment (PRA). Evaluation of the key features and the inventory is part of the certification

review. However, they reflect a design in its preliminary stages (not a detailed design or specification) and by themselves do not provide a basis on which a safety determination can be made.

In SECY-92-053, "Use of Design Acceptance Criteria During 10 CFR Part 52 Design Certification Reviews," dated February 19, 1992, the staff proposed using design acceptance criteria (DAC) as an approach to the ABWR design review because detailed design information was unavailable for selected areas of rapidly changing technology, including human factors aspects of the CR and remote shutdown station design. Therefore, it was inadvisable to require detailed design specifications at this point. The inclusion of DAC in specific system inspections, tests, analyses, and acceptance criteria (ITAAC) is discussed in Section 14.3 of this report.

The staff will verify conformance with the ITAAC at several points during the CR design process. The documents that demonstrate satisfactory implementation of the ITAAC will be available for inspection as they are completed. In the DFSER the staff introduced the concept of conformance review points as part of the DAC process. These points would be key points during the DAC process at which the staff would complete an adequacy review. As its review of the development of the ITAAC and DAC continued, the staff determined that five specific conformance points were unnecessary and were impractical. The deletion of the discussion of specific conformance review points was the result of the staff's continued review of the design certification applications from several vendors. At the time the DFSER was written, the staff envisioned that a design would progress in such a fashion that at a given time (e.g., a conformance review point) all information related to a particular aspect of the design process associated with that particular conformance point would be available; for example, all implementation plans would be available at one time. Because of the nature of the design process, it is not practical to assume that all documentation on a specific conformance review would be available at one point in time.

DAC are prescribed limits, parameters, procedures, and attributes on which the staff relies to make a final safety determination to support design certification. The DAC are measurable or testable and must be verified in order for the staff to accept the final design. DAC delineate the process and requirements that a combined license (COL) applicant must implement during the development of detailed design information for the CR and the remote shutdown station. The adequacy of the detailed design will be periodically assessed as it develops. The COL applicant must demonstrate that the DAC are met. Failure to

Human Factors Engineering

successfully complete ITAAC and any supporting DAC may require repeating earlier ITAAC and/or changing the system design.

Because the criteria for review of the design and implementation process for a CR or other control system were not clearly defined in current regulations and guidance documents, the staff developed criteria as part of this review. These criteria provided the basis with which to (1) assess if the appropriate HFE elements are included in the design and implementation process, (2) identify what materials need to be reviewed for each element, and (3) evaluate the adequacy of DAC and ITAAC to be used by the staff to verify each of the review elements, as developed by GE.

The staff design certification evaluation is based partially on design information and partially on an implementation process plan that describes the HFE program elements required to develop the key features and inventory into an acceptable detailed design specification. Along with the design and implementation process plan, GE has submitted the necessary DAC and ITAAC to ensure that the design and implementation process is properly executed by the COL applicant. GE has submitted a design and implementation process plan for the major design activities for the ABWR HFE effort. The first part of the plan presents GE's plant and system design elements; the second part describes the elements that must be implemented by a COL applicant to complete the design activity. The staff required that the design and implementation process plan contain descriptions of all required human factors activities (elements) that are necessary and sufficient for the development and implementation of the ABWR HSI.

18.1 Methodology

The staff review was performed in two phases. A preliminary review was performed on early versions of the SSAR and was documented in the DSER (SECY-91-320). This review is summarized in Section 18.1.1 of this report. DSER (SECY-91-320) issue resolution and further development of the SSAR were then reviewed and documented in the DFSER (SECY-92-349). The scope of this subsequent review is described in Section 18.1.2. As part of the final review, the staff developed the HFE PRM, or the review model, identified above for the evaluation of a design process. Development of the HFE PRM, found in Appendix J of this report, is described in Section 18.1.3.

18.1.1 Preliminary Review and Draft Safety Evaluation Report Issues

The primary source of information reviewed by the staff for the DSER (SECY-91-320) was SSAR Chapter 18 (updated through Amendment 15) and Chapter 13 (updated through Amendment 7) and GE's responses to staff requests for additional information (RAI), Questions 620.1 through 620.37, as documented in Chapter 20 of the SSAR. The review focused on four important aspects of ABWR human factors considerations:

- the organizational structure of the human factors function
- design goals and assumptions
- design processes
- the specification of HSI design requirements

In addition, the review included GE's resolution of those safety issues (unresolved safety issues, generic safety issues and the construction permit/manufacturing license (CP/ML) rule of 10 CFR 50.34(f)) related to human factors considerations addressed in SSAR Chapters 13 and 18.

From its initial review, the staff concluded that the human factors program for the HSI was generally inadequate as presented in GE's initial documentation and that SSAR Chapter 18 and Sections 13.2 and 13.5 did not provide sufficient information to support a determination that the ABWR design as proposed by GE for certification would adequately incorporate accepted human factors considerations in a manner that would achieve required safety and reliability. The principal reasons for this finding were: (1) design bases were specified in the SSAR without supporting rationale, (2) a design process was presented in insufficient detail and without results, (3) HSI design requirements were presented without evidence that they were derived from the design process and without supporting tests and evaluations, and (4) the documentation did not provide sufficient detail to support the review of the ABWR human factors efforts to a level necessary for design certification. Twenty-four issues were identified as requiring resolution. The issues are listed in Table 18.1 of this report. The table shows the section in this chapter where each DSER (SECY-91-320) open issue is addressed. In the discussions with GE that followed issuance of the DSER (SECY-91-320), two issues identified below were added in order to address the overall lack of design detail. These two issues are discussed in detail in Sections 18.4 and 18.9 of this report.

Table 18.1 DSER (SHCY-91-320) Chapter 18 human factors engineering issues

Issue Number	Issue	DSER Section	SHR Section Were Addressed
18.01	Qualifications of GB ABWR human factors design team	18.3.1	18.1.1
18.02	Human-systems interface (HSI) design and evaluation process	18.3.1	18.3
18.03	Number of staff in control room	18.3.2	18.2
18.04	Operator and system reliability	18.3.2	18.9
18.05	Operator workload analysis	18.3.2	18.9
18.06	Tests and analysis to support design implementation:	18.3.2	18.3
	a. Analyses conducted to date		
	b. Further testing		
18.07	ABWR human factors program plan (HPPP)	18.3.3.1	18.9
18.08	Control room prototype:	18.3.3.1	18.3
	a. Standardized features		
	b. Prototype evaluation		
18.09	Operator workload	18.3.3.2	18.3
18.10	Detailed task analyses	18.3.2	18.9
18.11	Tests, evaluations, studies to support design approaches	18.3.2	18.3
18.12	Adequacy of HSI design requirements	18.3.4.1	18.3
18.13	HSI design requirements for cathode ray tube (CRT), flat panel, and large-screen displays	18.3.4.1	18.9
18.14	Analysis to justify sole operator attentiveness and rationale for number of operators at main console	18.3.4.2	18.2
18.15	CRT display information	18.3.4.2	18.9
18.16	Power generation control system reliability	18.3.4.2	18.9
18.17	Alarm suppression criteria, alarm points	18.3.4.2	18.9
18.18	Safety parameter display system design scope	18.3.4.2	18.3
18.19	Remote shutdown system design rationale	18.3.4.2	18.5
18.20	Local valve position indication	18.3.4.2	18.6
18.21	Procedure development	18.3.4.4	18.1.1
18.22	Training materials	18.3.4.4	18.1.1
18.23	Unresolved safety issues and generic safety issues	18.4.1	18.7
18.24	Construction permit/manufacturing license-issues	18.4.2	18.7
18.25	Design process	-	18.9
18.26	Inventory	-	18.4

Human Factors Engineering

- Issue 18.25 - Design Process
- Issue 18.26 - Inventory

DSER (SECY-91-320) Issues 18.01, 18.21, and 18.22 were resolved before the SSAR review for reasons discussed below.

DSER (SECY-91-320) Issue 18.01: Qualifications of the GE ABWR Human Factors Design Team

In Section 18.3.1 of the DSER (SECY-91-320), the staff stated that "additional detailed information on the human factors qualifications of the GE ABWR human factors design team is required." This information was deemed important in the DSER review because of the assumption that the SSAR was to include an essentially complete design for the HSI. GE responded by providing design team information in its letter of October 1, 1991.

Evaluation: Since HFE is to be primarily performed by the COL applicant, the issue is no longer a concern. Instead, the focus on the design team was shifted toward establishing the qualification requirements of the design team that will actually implement the HFE program discussed in Section 18.9 of this report. Because of the change in focus for certification from an HFE design to a design process, GE's initial response to this issue in its letter of October 1, 1991, provided sufficient information for the staff's evaluation of this issue. Since GE has addressed the qualification requirements for its HFE design team, this item is resolved.

DSER (SECY-91-320) Issue 18.21: Procedure Development

In the DSER (SECY-91-320), the staff stated that system-level operating procedures had been developed concurrently with the development of the ABWR systems design. These procedures and the associated task analyses on which the HSI requirements are based were not included in the references (SSAR Section 18.6); thus, they could not be evaluated. For the ABWR design certification, the staff expects the COL applicant to provide detailed program descriptions for the development of standardized plant procedures and standardized plant personnel training materials. Further, the staff expects the COL applicant to develop integrated operating procedures that reflect the full level of detail consistent with and included as part of the final plant design. In addition, the COL applicant should develop procedure development guidelines (e.g., procedure writers guide, verification and validation (V&V) guidelines, and generic technical guidelines) with sufficient detail to ensure that the COL applicant's implementation of the processes and criteria delineated in these guidelines, when revising procedures

will preserve the human factors insights and the overall ABWR design.

Evaluation: The staff determined that the development of detailed procedures and training materials was beyond the scope of the ABWR design certification and was the responsibility of the COL applicant. This was DFSEER Confirmatory Item 18.9.2.2.7-1. Chapter 13.5.3.3 of the SSAR states that procedures will be used during the V&V process as described in Article VII of Table 18B-1. The staff accepts GE's determination that procedure development is beyond the scope of the ABWR design certification, and GE has included a COL action item that is acceptable. Therefore, this item is resolved.

DSER (SECY-91-320) Issue 18.22: Training Materials

In the DSER (SECY-91-320), the staff stated that it "expects GE to develop and submit for certification a detailed program description for developing the training material as part of the design certification for the ABWR." After the DSER was issued, the staff determined that training materials were already a part of the established licensing review process under 10 CFR Part 50 for the COL applicant and did not need to be addressed in the certification review under 10 CFR Part 52. The staff accepts that the development of training materials is beyond the scope of the ABWR design certification and that the materials will be developed and submitted by the COL applicant as part of the licensing process; therefore, this item is resolved. A specific COL action item was deemed unnecessary for this item, because 10 CFR Part 50 requirements are sufficiently explicit and the COL action item list is not intended to be exhaustive.

18.1.2 Final Standard Safety Analysis Report Review

The sources of information used for the final review described in this chapter were SSAR Chapter 18 through Amendment 34, GE's responses to the DSER (SECY-91-320) issues as documented in the SSAR, and GE's responses to the RAIs as documented in Chapter 20 of the SSAR. In support of design certification, GE personnel presented additional information on the standard design features to the NRC staff at a meeting during a visit to Japan to observe the Japanese ABWR CR prototypes ("Advanced Reactor Programs - ABWR Control Room Design" - presented to the United States Nuclear Regulatory Commission, M.A. Ross (GE), April 3 and 4, 1992, Tokyo, Japan - GE Proprietary). This information was later summarized and included in the design certification application as Appendix 18G to the SSAR. These standard design features are evaluated in Section 18.3 of this report.

18.1.3 Development of Review Criteria

18.1.3.1 Objectives

As stated in the beginning of this chapter, complete detailed HSI design information would not be available for review before design certification. Certification is based on the staff's approval of GE's design and implementation process plan. For a design and implementation process plan to result in an acceptable design, it must contain (1) descriptions of all required HFE program elements for the design and development and implementation of the ABWR HSI and (2) DAC for the conformance reviews under the ITAAC.

To review the GE-proposed HFE process, the staff had to (1) assess if all the appropriate HFE elements were included, (2) identify what materials needed to be reviewed for each element, and (3) evaluate the proposed DAC and ITAAC to verify each of the elements. To conduct the review, the staff identified (1) which aspects of the HSI design process were required to ensure that HFE safety design in support of safe plant operation is achieved and (2) the review criteria by which each element is assessed. Review criteria independent of that provided by GE were required to ensure that GE's plan reflects currently accepted HFE practices and is a thorough, complete, and workable plan. To support such a review, the staff developed a technical basis for review of the HSI design process. The specific objectives of this effort were:

- (1) To develop an HFE PRM to serve as a technical basis for the review of the GE-proposed process for certification. The HFE PRM is (a) based on currently accepted HFE practices, (b) well defined, and (c) validated through experience with the development of complex, high-reliability systems.
- (2) To identify the HFE elements in a system development, design, and evaluation process that are necessary and sufficient for successful integration of the human component in complex systems.
- (3) To identify which aspects of each HFE element are key to a safety review and are required to monitor implementation of the process.
- (4) To specify the acceptance criteria by which HFE elements can be evaluated as design development progresses.

18.1.3.2 HFE PRM Development

The staff reviewed current HFE guidance and practices described in a wide range of nuclear industry and non-nuclear industry documents to identify important human factors program plan (HFPP) elements relevant to a design process review. A generic system development, design, and evaluation process was defined with eight key HFE elements that included criteria by which they could be assessed. This is referred to as the HFE PRM, or the review model.

The HFE PRM was based largely on applied general systems theory and the Department of Defense (DOD) systems development process (which is rooted in systems theory). Applied general systems theory provides a broad approach to system design and development that is based on a series of clearly defined developmental steps, each with clearly defined goals and specific management processes to attain them. Systems engineering has been defined as "the management function which controls the total system development effort for the purpose of achieving an optimum balance of all system elements. It is a process which transforms an operational need into a description of system parameters and integrates those parameters to optimize the overall system effectiveness." (F. Kockler, et al., Systems Engineering Management Guide (AD/A223 168), Defense Systems Management College, Fort Belvoir, Virginia, 1990.)

Use of the DOD system development process and procedure in the development of the HFE PRM was based on several factors. DOD policy identifies personnel as a specific component of the total system. A systems approach implies that all system components (hardware, software, personnel, support, procedures, and training) are given adequate consideration in the developmental process. A basic assumption is that the personnel component receives serious consideration from the very beginning of the design process. In addition, DOD has the most experience in applying HFE to the development of complex, technical systems (as compared with nonmilitary system developers); thus, its process is mature, formalized, and represents the most highly developed and well defined model of the HFE process available.

Within the DOD system, the development of a complex system begins with the mission or purpose of the system and the capability requirements needed to satisfy mission objectives. Systems engineering methods must be used as early as possible to develop the system concept and to

Human Factors Engineering

define the system requirements. During the detailed design of the system, systems engineering ensures

- balanced influence of all required design specialties
- resolution of interface problems
- effective conduct of tradeoff analyses
- effective conduct of design reviews
- V&V of system performance

The effective integration of HFE considerations into the design is accomplished by providing (1) a structured topdown approach to system development that is iterative, integrative, and interdisciplinary and (2) a management structure that details the HFE considerations in each step of the overall process. A structured topdown approach to nuclear power plant (NPP) HFE is consistent with the approach to new CR design as described in Appendix B to NUREG-0700 and the more recent internationally accepted industry standard, International Electrotechnical Commission 964, for advanced CR design. The approach also is consistent with the recognition that human factors issues and problems emerge throughout the NPP design and evaluation process; therefore, human factors issues are best addressed with a comprehensive topdown program.

The scope of the HFE PRM excluded a training program development element because training is adequately addressed by existing NRC requirements. In addition, human reliability analysis was excluded and is addressed in Section 19.1 of this report.

The HFE PRM incorporates the requirements (as discussed below in Section 18.1.3.3) of 10 CFR 50.34(f)(2)(iii) as required by 10 CFR 52.47(a)(1)(ii). The HFE PRM is briefly described below and is included in Appendix J of this report.

18.1.3.3 HFE PRM Model Description

The overall purpose of the HFE PRM review is to ensure that

- the applicant has integrated HFE into plant development and design
- the applicant has provided HSIs that make possible safe, efficient, and reliable operator performance of operation, maintenance, test, inspection, and surveillance tasks
- the HSIs reflect "state-of-the-art human factors principles" as required by 10 CFR 50.34(f)(2)(iii).

State-of-the-art human factors principles are defined as those principles currently accepted by human factors

practitioners. "Current" refers to the time when a program management or implementation plan is prepared. "Accepted" refers to a practice, method, or guide that is (1) documented in the human factors literature within a standard or guidance document that has undergone a peer-review process or (2) can be justified through scientific or industry research and practices.

All aspects of HSI will be developed, designed, and evaluated on the basis of a structured topdown system analysis using accepted HFE principles based on current HFE practices. HSI is used here in the very broad sense and shall include all operations, maintenance, test, and inspection interfaces and procedures materials.

The model developed to achieve this commitment contains eight elements:

- (1) human factors engineering program management
- (2) operating experience review (OER)
- (3) system functional requirements analysis
- (4) allocation of function
- (5) task analysis
- (6) human-system interface design
- (7) plant and emergency operating procedure development
- (8) human factors verification and validation

The elements and their interrelationships are illustrated in Figure J.1 of Appendix J of this report. Also illustrated are the minimal set of items to be submitted by the COL applicant for NRC staff review of the applicant's HFE efforts. A description of the purpose of each element follows.

Element 1 - Human Factors Engineering Program Management

To ensure the integration of HFE into system development and the achievement of the goals of the HFE effort, an HFE design team and an HFE program plan must be established to ensure the proper development, execution, oversight, and documentation of the HFE program. An HFE issue tracking system (to document and track HFE-related problems, concerns, and issues, and their solutions throughout the HFE program) will be established as part of the program plan. The HFE issue tracking system will be used as a mechanism to log ABWR-specific design issues as part of the COL applicant's overall design process.

Element 2 - Operating Experience Review

The accident at Three Mile Island (TMI) in 1979 and other reactor incidents have illustrated that significant problems

in the actual design and design philosophy of NPP HSIs exist. There have been many studies as a result of these accidents and incidents. Utilities have implemented both NRC-mandated changes and additional improvements on their own initiative. However, the changes were formed on the basis of the constraints associated with backfits to existing CRs using early 1980s technology, which limited the scope of corrective actions that might have been considered (i.e., more effective changes can be made in the case of a new CR with the modern technology typical of advanced CRs). Problems and issues encountered in similar systems of previous designs must be identified and analyzed so that they are avoided in the development of the current system, or in the case of positive features, to ensure their retention.

Element 3 - System Functional Requirements Analysis

System requirements shall be analyzed to identify those functions that must be performed to satisfy the objectives of each functional area. System function analysis shall (1) determine the objective, performance requirements, and constraints of the design and (2) establish the functions that must be accomplished to meet the objectives and required performance.

Element 4 - Allocation of Function

Functions shall be allocated to take advantage of human strengths and to avoid functions that would be affected by human limitations. To ensure that functions are allocated according to accepted HFE principles, a structured and well-documented methodology of allocating functions to personnel, system elements, and personnel-system combinations shall be developed.

Element 5 - Task Analysis

Task analysis shall include the systematic study of the behavioral requirements of the tasks personnel are required to perform in order to achieve the functions allocated to them. The task analysis shall

- provide one of the bases for making design decisions (e.g., determining before hardware fabrication, to the extent practicable, whether system performance requirements can be met by combinations of anticipated equipment, software, and personnel)
- ensure that human performance requirements do not exceed human capabilities
- be used as basic information for developing procedures
- be used as basic information for developing staffing, skill, training, and communication requirements of the system

- form the basis for specifying the requirements for the displays, data processing, and controls needed to carry out tasks

Element 6 - Human-System Interface Design

Human engineering principles and criteria shall be applied along with all other design requirements to identify, select, and design the particular equipment to be operated, maintained, and controlled by plant personnel.

Element 7 - Plant and Emergency Operating Procedure Development

Plant and emergency operating procedures (EOPs) shall be developed to support and guide human interaction with plant systems and to control plant-related events and activities. Human engineering principles and criteria shall be applied along with all other design requirements to develop procedures that are technically accurate, comprehensive, explicit, easy to use, and validated. The types of procedures covered in the element are

- normal plant and system operations (including startup, power, and shutdown operations)
- abnormal and emergency operations
- alarm response

Element 8 - Human Factors Verification and Validation

Using HFE procedures, guidelines, standards, and principles, the acceptability of the final HSI design shall be evaluated as an integrated system. The integrated system includes all

- human-hardware interfaces
- human-software interfaces
- communications (human-human interfaces)
- procedures
- workstation and console configurations
- control room design
- remote shutdown system (RSS)
- design of the overall work environment

High fidelity with regard to the final design is expected (i.e., only minor differences between the actual final design and the evaluated design are acceptable). Validation should be accomplished through dynamic task performance of trained operating personnel using evaluation tools that are appropriate to the accomplishment of this objective as stated in Table 3.1 of the certified design material (CDM) and SSAR Appendix 18E.

Human Factors Engineering

18.2 Design Goals

18.2.1 General Discussion in the Standard Safety Analysis Report

The HSI design goals are described in SSAR Section 18.2. GE's "primary goal for the operator interface design is to facilitate safe, efficient, and reliable operator performance during all phases of normal plant operation, abnormal events, and accident conditions." It further states that, to achieve these goals, the HSIs will be designed and provided in a manner consistent with "good human factors engineering practices."

Within the context of this review, GE's eight specific design bases are considered to be design assumptions since they are "givens" and, as presented, have not been derived from analysis. They thus become design drivers. These design assumptions are evaluated in Section 18.2.2.2 of this report.

18.2.2 Analysis

18.2.2.1 DSER (SECY-91-320) Issue Resolution

Two DSER (SECY-91-320) issues related to the design goals are summarized below, including the resolution that was achieved.

DSER (SECY-91-320) Issue 18.03: Number of Staff in the Control Room

In the DSER (SECY-91-320), the staff stated that the number of main control room (CR) operating staff needed to be clearly established for the ABWR. Design Basis 1 in the original SSAR stated that for "normal operations, the ABWR shall be operable by one senior reactor operator who will be directly involved in manipulation of the reactor controls, one assistant CR SS (shift supervisor), one CR SS (shift supervisor), and two auxiliary equipment operators." The operating crew could be increased during accident conditions. In the DSER (SECY-91-320), the staff stated that the specification of a single operator at the control boards during normal operations was considered to be a significant design driver.

Evaluation: This issue is related to DSER (SECY-91-320) Issue 18.03 and was similarly resolved. GE satisfactorily clarified this issue in amended SSAR Section 18.2 by indicating that the ABWR operating crew will be consistent with the requirements of 10 CFR 50.54(m) and that two operators will be available during all phases of ABWR operation. Therefore, this item is resolved.

DSER (SECY-91-320) Issue 18.14: Analysis To Justify Sole Operator Attentiveness and Rationale for Number of Operators at Main Console

In the DSER (SECY-91-320), the staff stated that an appropriate analysis should be provided to justify how one operator at the main console will remain attentive to his/her duties, and that the maximum number of operators who are expected to monitor and operate the plant during an emergency at the main control console needs to be specified with the rationale to support this number. The staff stated that this rationale should be based on the function and task analysis performed to support the CR design. The staff required the rationale for the number of operators anticipated to monitor and control functions on the main control console while the operator is performing other functions.

Evaluation: GE satisfactorily clarified this issue in amended SSAR Section 18.2 by indicating that the ABWR operating crew will be consistent with the staffing requirements of 10 CFR 50.54(m) and that two operators will be available during all phases of ABWR operation. Therefore, this item is resolved.

18.2.2.2 Evaluation of the Current SSAR Design Bases

GE defined the following eight discrete design bases for use in the design development of the ABWR CR:

Design Basis 1 - The ABWR will be operated by two reactor operators, and four licensed operators will be on shift at all times, consistent with the requirements of 10 CFR 50.54(m).

Design Basis 2 - Efficient and reliable operation will be promoted through increased automation.

Design Basis 3 - Only proven technology will be used for the HSI design.

Design Basis 4 - Safety-related systems monitoring displays and control capability will meet the requirements for independence and electrical separation.

Design Basis 5 - The operator interface design will be highly reliable and provide functional redundancy.

Design Basis 6 - The principal functions of the ABWR safety parameter display system (SPDS) will be integrated into the HSI design.

Design Basis 7 - GDC 19 will be met (GDC 19 states that a control room shall be provided from which actions can be taken to operate the nuclear power unit and that equipment outside of the control room shall be provided to shutdown the reactor.)

Design Basis 8 - Design bases for the RSS will be specified in SSAR Section 7.4.

The staff reviewed the GE design bases which will be met through the design and implementation process described in SSAR Chapter 18. For two of the bases issues were identified which are described in detail below.

Design Basis 1 states that the ABWR will be operated by two reactor operators and that four licensed operators will be on shift at all times consistent with the requirements of 10 CFR 50.54(m). These will include a licensed senior reactor operator SS and assistant SS. While this may be a reasonable design goal on the basis of the preliminary analyses and evaluations conducted thus far, the capability of the main control console to accommodate two operators will have to be validated as part of HFE V&V activities (Element 8). Further, it will have to be verified that no more than two operators need to access the controls and displays at the main control console under all normal, abnormal, and accident conditions. The roles and responsibilities of the SS and assistant SS also will need to be defined, including their information requirements and access to displays. As reported in the preliminary validation tests, using the Japanese CR prototypes, in SSAR Appendix 18G, the main console would be crowded if more than two operators were required there. SSAR Table 18E-1 states that the COL applicant will validate that the CR design will support acceptable performance of all tasks assigned to the operating crew under a variety of plant conditions. In the DFSER the evaluation of the number of operators needing access to controls at the main console and the specification of the roles and responsibilities of the SS and assistant SS was identified as DFSER COL Action Items 18.2.2.2-1 and 18.2.2.2-2. GE has adequately incorporated these issues in the SSAR as Item 18.8.2 in SSAR Section 18.8. The staff agrees with GE's assertion that this issue (results of the evaluation shall be placed in the HFE Issue Tracking System - Item II.2 of SSAR Table 18E-1) should be resolved by the COL applicant as noted in SSAR Section 18.8.2 as part of the design and implementation process.

Design Basis 2 states that efficient and reliable operation will be promoted through increased automation. This design basis is acceptable only if it can be demonstrated that the increases in automation promote operational reliability and that automation is not introduced in such an arbitrary manner that it may impair human and/or system

performance. This basis must be evaluated as part of the requirements of Element 4, "Allocation of Function." Decisions regarding which functions should be automated are more effectively made after the function analyses have been conducted and functions have been allocated as documented in NUREG/CR-3331, "A Methodology for Allocating Nuclear Power Plant Control Functions to Human or Automatic Control." In the DFSER, the evaluation of automation strategies and their effects on operator reliability was identified as COL Action Item 18.2.2.2-3. GE has acceptably incorporated this issue (results of the evaluation shall be placed in the HFE Issue Tracking System - Item II.2 of SSAR Table 18E-1) as noted in SSAR Section 18.8.3 as part of the design and implementation process.

No issues were identified for SSAR Design Bases 3 through 8 listed above. Therefore, these items are resolved.

18.2.3 Finding

As discussed in Section 18.2.2.1 of this report, DFSER (SECY-91-320) Issues 18.03 and 18.14 are resolved. In addition, the staff identified three issues to be addressed, as part of the design and implementation process, as described in SSAR Table 18E-1.

- evaluation of the number of operators needing access to controls and displays at the main console
- specification of the roles and responsibilities of the SS and assistant SS
- evaluation of the impact of automation on operator reliability

SSAR Section 18.8 states that these issues will be resolved by the COL applicant after system functional requirements analyses, the allocation of functions, and the task analysis are performed.

18.3 Main Control Room Standard Design Features

18.3.1 General Discussion in the Standard Safety Analysis Report

The CR is characterized by 18 standard features, each of which is reviewed in Section 18.3.2.2 below. The features were derived from a system analysis and verified through V&V testing using two Japanese CR prototypes. The

Human Factors Engineering

standard design features are described in SSAR Section 18.4, and their development is described in SSAR Sections 18.3.1 and 18.3.2.

A possible ABWR main CR is partially characterized in SSAR Appendix 18C. Since this serves as an example of how the standard features might be implemented, it has no specific application in the safety review of the ABWR for design certification. SSAR Appendix 18C should not be misinterpreted as providing any information specific to the ABWR CR design safety finding.

18.3.2 Analysis

18.3.2.1 DSER (SECY-91-320) Issue Resolution

DSER (SECY-91-320) Issues 18.02, 18.06, 18.08, 18.09, 18.11, 18.12, and 18.18 were related to the standard features that are described in SSAR Section 18.4. Each of these issues is summarized below, the path to resolution that was proposed in the discussions with GE after the DSER (SECY-91-320) was issued is given, and issue resolution is evaluated. These issues generally address the need for information regarding the design process and analyses leading to the standard features and their test and evaluation.

DSER (SECY-91-320) Issues 18.13 (HSI design requirements for the cathode ray tube (CRT), flat panel, and large-screen displays), 18.15 (CRT display information), and 18.17 (alarm suppression criteria, alarm points) address a level of design detail beyond the description of the standard features in the SSAR. Thus, these DSER issues are addressed in Section 18.9 because they will be resolved by the COL's design and implementation process.

DSER (SECY-91-320) Issue 18.02: HSI Design and Evaluation Process and Issue 18.06: Tests and Analysis To Support Design Implementation

In the DSER (SECY-91-320), the staff stated that additional detailed information regarding the HSI design and evaluation process was necessary (Issue 18.02). Additional detailed information also was necessary about the methods, criteria, and results of analyses performed to support the level and type of staffing, automation, and function allocation to achieve the goals of safe and reliable performance of the operating crew and overall system (Issue 18.06).

Evaluation: These issues are addressed in SSAR Section 18.4 and Appendix 18G. The standard design features were the result of a 5-year development program that included

- the preparation of implementation plans for major design and evaluation activities
- the derivation of general HSI requirements from the design of individual systems
- task analyses for safety-related functions based on manual operations
- a systematic allocation-of-function strategy based on workload analysis and an analysis of such task characteristics as degree of repetitiveness and complexity
- an analysis of current trends and technology assessments of the major CR features including approaches to automation, console design, video display units (VDUs), display techniques, large display panels, use of fixed-position displays, alarms, and CR layout

After the features were identified, GE assessed them in a validation testing program using two Japanese CR prototypes that had the standard features. Three teams of operators participated in the validation tests by performing a range of operational tasks, including normal operation, equipment failures, scrams, and accidents. Information collected for the validation tests included that collected by videotape and observations and operator opinion.

The test results generally supported the use of the standard features. However, the staff noted several limitations in the tests:

- The standard features were not individually tested; instead, the entire design as a package (which included standard features and other design detail) was evaluated. Also, the features as implemented in the validation tests were designed at a considerably greater level of detail when compared with their SSAR definition as standard features. Thus, it is possible that the same set of standard features (as defined in the SSAR) could be improperly designed and/or poorly integrated to result in an unacceptable design.
- The data collected were limited mainly to observations and subjective evaluations. A more complete performance measurement evaluation such as that described by HFE PRM, Element 8, was not used.
- The selection of accident and transient test scenarios was limited. More extensive tests will be needed but will be performed as part of the Element 8 validation test program (as defined in the HFE PRM).

- The results were expressed in general terms without the compilation of specific findings related to each of the general features (e.g., the relative merits and problems encountered for each specific feature).
- The test program showed that the main control console would be crowded if more than two operators were required there. The result is specific to the design configuration and test scenarios used in the test program. The SSAR level of detail does not include console dimensions; therefore, the test results from the Japanese prototypes do not directly pertain to the key feature as represented in the SSAR. Analysis of the console's suitability under maximum staffing demands should be evaluated as part of the detailed task analysis and validation test program.

Although the limitations noted above of the testing of the standard features were identified, the staff determined that the level of validation provided by the testing was sufficient to support the use of these features for the ABWR design because the suitability of these features would be revisited in more detail during the V&V of actual ABWR CR designs by the COL applicant.

DSER (SECY-91-320) Issues 18.02 and 18.06 became DFSER Confirmatory Item 18.3.2.1-1. GE has provided the description of the design development and validation testing in SSAR Appendix 18G. The staff agrees that GE's description of the design development and validation testing as discussed above is acceptable and therefore considers this item resolved.

In addition, the standard features are defined at a very general level and provide a general approach to CR design. They are not a final design specification and not at the level of detail needed for a final safety determination without consideration in the context of the design development process. Therefore, the detailed design implementation of the standard features and their integration into the CR and RSS designs will be included as part of the staff's review of the COL's DAC submittals related to the activities of HFE PRM Elements 6 and 8. In addition, the validation of the final design of the standard features is specifically identified as COL Action Item 18.8.5 in SSAR Section 18.8.

DSER (SECY-91-320) Issue 18.08: Standardized Features and Prototype Evaluation

In the DSER (SECY-91-320), the staff stated that additional detailed information was necessary to precisely indicate the aspects of the CR design that are part of the standardized design and that are unique to a COL applicant's implementation consistent with accepted human

factors principles and practices and the requirements of 10 CFR Part 52. The staff believes development of a fully functional CR prototype of the standard design is appropriate in order to demonstrate acceptable human performance. Thus, there are three parts to this issue: (1) the aspects of the CR that are part of the standardized design, (2) the level of detail with which the standard features are described, and (3) the use of a prototype.

Evaluation: For Parts 1 and 2 of this issue, the staff conducted a feature-by-feature evaluation to determine the level of design detail that was supported by GE's design efforts. GE agreed to modify the description of the standard features to a level of detail supported by the design and evaluation efforts discussed with respect to Issue 18.06 above. The results were provided in GE's letter of February 18, 1992. This was DFSER Confirmatory Item 18.3.2.1-2.

SSAR Section 18.4 gives a revised description of the standard CR design features. The description has been modified to a level of detail commensurate with the test and evaluation program. The staff has determined that the design development description provided by GE is acceptable on the basis of the approved DAC.

In regard to Issue Part 3, the use of a prototype in design and evaluation is addressed as part of the design process discussed in Section 18.9 of this report.

GE has submitted the description of the design development and validation testing and revised the description of the standard features in SSAR Section 18.4. On the basis of its review, the staff finds GE's submittals acceptable; therefore, this item is resolved.

DSER (SECY-91-320) Issue 18.09: Operator Workload

In the DSER (SECY-91-320), the staff stated that GE had not indicated how the workload was defined and measured (in the context of allocation of functions) or what constitutes an appropriate operator workload level. The staff further stated that it was unclear how validating allocation-of-function decisions by a COL applicant at this late point in the design process could result in a standardized design. Thus, there are three parts to this issue: (1) workload definition for allocation-of-function studies, (2) determination of satisfactory workload, and (3) implications for postcertification evaluations that require modification of certified aspects of the design, in this case, allocation of function.

Evaluation: GE addressed this issue in its letter of February 18, 1992. Parts 1 and 2 of this issue are addressed in the evaluation of Issue 18.06 above. The

Human Factors Engineering

specification of workload evaluations during postcertification is addressed in the discussion of the design process in Section 18.9 of this report and is required by the COL applicant in SSAR Table 18.E-1. This was DFSER Confirmatory Item 18.3.2.1-3. GE has submitted the information requested by the staff on operator workload in SSAR Appendix 18G. The staff has reviewed GE's submittal and found it to be acceptable; therefore, this item is resolved.

DSER (SECY-91-320) Issue 18.11: Tests, Evaluations, Studies to Support Design Approaches

In the DSER (SECY-91-320), the staff stated that information was needed on tests, evaluations, and trade studies performed to support the selection of design approaches (e.g., the use of touch-screen interfaces).

Evaluation: GE addressed this issue in its letter of February 18, 1992, and it is discussed under Issue 18.06. After discussions with the staff, GE eliminated several design details from the specification as standard features, including the use of touch-screen interfaces. Those that remain are those supported by the test program. DSER (SECY-91-320) Issue 18.11 was to be resolved subject to receipt of the amended SSAR. This was DFSER Confirmatory Item 18.3.2.1-4. GE has provided, in SSAR Appendix 18G, the information to support the staff's determination that GE's design details, as well as the appropriate DAC, are acceptable. Therefore, this item is resolved.

DSER (SECY-91-320) Issue 18.12: Adequacy of HSI Design Requirements

In the DSER (SECY-91-320), the staff stated that, in the absence of a systems analysis and test and evaluation results, there was no basis to evaluate the reasonableness and adequacy of the HSI design requirements from a top-down (or bottom-up) perspective.

Evaluation: GE addressed this issue in its letter of February 18, 1992, as is discussed under Issue 18.06 above. DSER (SECY-91-320) Issue 18.12 was to be resolved subject to receipt of the amended SSAR. This was DFSER Confirmatory Item 18.3.2.1-5. GE has provided, in SSAR Appendix 18G, the information to support the staff's determination that the HSI design requirements and appropriate DAC are acceptable. Therefore, this item is resolved.

DSER (SECY-91-320) Issue 18.18: Safety Parameter Display System Design Scope

In the DSER (SECY-91-320), the staff stated that, at the present stage of design, it could not determine if the ABWR safety parameter display system (SPDS) will meet all the NRC SPDS design criteria in NUREG-0737, Supplement 1, "Clarification of TMI Action Plan Requirements," 1982. The requirements regarding the SPDS in the ABWR CR are discussed in detail under "Standard Feature N."

The SPDS function and the list of critical parameters, as described in SSAR Section 18.4.6, did not include parameters that would provide operators with information about radioactivity control should there be a release of radioactive materials. The SSAR further stated that the COL applicant may provide a radioactivity release control information display. GE's initial approach to meeting NRC requirements for the SPDS function was not sufficient.

Evaluation: SPDS design is part of the ABWR standard features. Therefore, it is reviewed in the next section under "Standard Feature N." The commitment to the requirements of NUREG-0737, Supplement 1, has been incorporated into the description in SSAR Section 18.2, which states that the SPDS functions will comply with the NUREG report requirements. Therefore, this item is resolved.

18.3.2.2 Evaluation of the Current SSAR

As a result of the DSER (SECY-91-320) issue resolutions discussed above, GE agreed to

- clarify how the standard features were defined from the design process
- provide support for the validation of the standard features
- redefine the SSAR descriptions of the standard features to bring them in line with the supporting design and validation efforts

The list of standard feature descriptions below was the result of that process. It is important to reemphasize that a final safety determination for the detailed design of the standard features will be made by the staff using the DAC as part of ITAAC.

Standard Feature A

Feature Description - The use of a single, integrated control console staffed by two operators; the console has a low profile so that the operators can see over the console from a seated position.

Feature Evaluation - This feature, as presented, is generally supported by GE's design analyses and evaluations as discussed in Section 18.3.2.1 above.

Standard Feature B

Feature Description - The use of an on-screen control video display unit (VDU) for safety system monitoring and non-safety system control and monitoring that is driven by the plant process computer system.

Feature Evaluation - This feature, as presented, is generally supported by GE's design analyses and evaluations as discussed in Section 18.3.2.1 above.

Standard Feature C

Feature Description - The use of a separate set of on-screen control VDUs for safety system control and monitoring and separate on-screen control VDUs for non-safety system control and monitoring; the operation of these two sets of VDUs is entirely independent of the process computer system. Further, the first set of VDUs and all equipment associated with their functions of safety system control and monitoring are divisionally separate and qualified to Class 1E standards.

Feature Evaluation - This feature, as presented, is generally supported by GE's design analyses and evaluations as discussed in Section 18.3.2.1 above.

Standard Feature D

Feature Description - The use of dedicated function switches on the control console.

Feature Evaluation - This feature, as presented, is generally supported by GE's design analyses and evaluations as discussed in Section 18.3.2.1 above. GE's rationale for specifying the use of dedicated switches for the identified functions is consistent with human factors engineering practices and is acceptable for this application. This feature is described in SSAR Section 18.4.2.5, indicating that several different types of switches are used, incorporating a technology that has been retained from the previous BWR designs. The type of switch will be determined by the design implementation process addressed in Section 18.9 of this report.

Standard Features E, F, and G

Feature Descriptions -

E - Operator selectable automation of predefined plant operational sequences.

F - The incorporation of an operator selectable semiautomated mode of plant operations. This mode will provide procedural guidance to the operators using the plant operating procedures as a basis for that guidance. This "feature" is stated at the general level and will be further specified during the design implementation process.

G - The capability to conduct plant operations in an operator manual mode.

Feature Evaluation - These features, as presented, are generally supported by GE's design analyses and evaluations as discussed in Section 18.3.2.1 above. The features as stated are at the level of general requirements for the levels of automation available to the operator. The operator maintains the capability to assume manual control at any time. This basic approach is consistent with current human factors engineering practice and is acceptable for this application.

Standard Feature H

Feature Description - The incorporation of a large display panel that presents information for use by the entire CR operating staff.

Feature Evaluation - This feature, as presented, is generally supported by GE's design analyses and evaluations as discussed in Section 18.3.2.1 above. It should be noted that the safety significance of this display approach will be dependent on the final design of display formats and distribution of information between the main control console and the large display. This is addressed as part of the COL's HSI design and V&V ITAAC.

Standard Feature I

Feature Description - The inclusion on the large display panel of fixed-position displays of key plant parameters and major equipment status

Feature Evaluation - This feature, as presented, is generally supported by GE's design analyses and evaluations as discussed in Section 18.3.2.1 above. Information on the large panel will be designed so that it can be observed from the supervisor's console (which is

Human Factors Engineering

farther from the panel than the main control console). As the COL applicant develops the design after certification, evaluations are required by the HFE PRM to confirm the allocation of information between the large panel and the main control console. This is addressed as part of the COL's HSI design and V&V ITAAC.

Standard Features J and K

Feature Descriptions -

J- The inclusion in the fixed-position displays of both Class 1E qualified (those that contain safety-related information) and non-Class 1E display elements.

K- The independence of the fixed-position displays from the plant process computer.

Feature Evaluation - The SSAR acceptably requires that the fixed-position displays of safety-related information conform to Class 1E standards and be independent of the plant process computer.

Standard Feature L

Feature Description - The inclusion in the large display panel of a large VDU that is driven by the plant process computer system.

Feature Evaluation - This feature, as presented, is generally supported by GE's design analyses and evaluations as discussed in Section 18.3.2.1 above.

Standard Feature M

Feature Description - The incorporation of a "monitoring only" supervisor's console that includes VDUs on which display formats available to the operators on the main control console also are available to the supervisors.

Feature Evaluation - The role of the supervisor and the supervisor's information requirements need to be more completely specified by the COL applicant before a determination of the design requirements of the console can be made. The present design of the main control console is for two seated operators. Making no provisions for control capability from the supervisor's console leads to the assumption that under no circumstances would control capability beyond that of the two operators be required. This will be considered by the COL applicant in the design process under Element 8 of the HFE PRM and will be resolved by the COL applicant on completion of the functional analysis as required by SSAR Table 18.E-2(II).

Standard Feature N

Feature Description - The incorporation of the SPDS function as part of the plant status summary information that will continuously be displayed on the fixed-position displays on the large display panel.

Feature Evaluation - This feature, as presented, is generally supported by GE's design analyses and evaluations as discussed in Section 18.3.2.1 above. Details pertinent to the SPDS are discussed below.

The staff reviewed GE's proposed approach to SPDS development in accordance with the requirements in 10 CFR 50.34(f)(2)(iv) and NUREG-0737, Supplement 1, and the guidance in NUREG-1342 "A Status Report Regarding Industry Implementation of Safety Parameter Display Systems" (1989), which describes SPDS implementation methods acceptable to the NRC staff as well as problem areas identified in operating plant SPDS reviews. GE describes its SPDS design in SSAR Section 18.4.2.11, as well as in DSER (SECY-91-320) Responses 3.b.1(n), 3.b.5, Table 3.b-1, and 5.e (GE letter of February 18, 1992).

The SPDS review for the ABWR is part of the staff's review to be conducted as part of HFE PRM Elements 6 (Design) and 8 (V&V). The ABWR CR and SPDS design, while not complete, is described in the SSAR. The discussion below addresses several SPDS requirements.

Paragraph 3.8a of NUREG-0737, Supplement 1, discusses the integration of the SPDS with related emergency response capabilities and includes the following comment on the SPDS:

- (1) Review the functions of the NPP operating staff that are necessary to recognize and cope with rare events that (a) pose significant contributions to risk, (b) could cause operators to make cognitive errors in diagnosing them, and (c) are not included in routine operator training programs.

This guidance was not specifically addressed in GE's response. Therefore, the COL applicant will need to consider incorporation of insights from the PRA into the SPDS selection process. For example, loss of power/station blackout (SBO) was very important to risk in the ABWR PRA (90 percent of core damage frequency (CDF) in the first revision of the PRA). Hence, there should be some monitoring of electric power sources as part of the SPDS. Also, with the addition of a gas turbine generator to the design to reduce the dominance of the SBO sequence, parameters related to the gas turbine should be included on the SPDS. Other safety-system failures that

appear prominently in the accident sequences are those of the high-pressure core flooders, reactor core isolation cooling (RCIC) system, low-pressure core flooders, residual heat removal system, and automatic depressurization system (ADS).

Additionally, cognitive errors made by operators and items not covered in operator training programs would need to be addressed similarly to the risk-significant items. Addressing the criteria of Paragraph 3.8a of NUREG-0737, Supplement 1, will be part of the COL applicant's CR design responsibility. This was DFSEER COL Action Item 18.3.2.2-1. The commitment to the criteria in NUREG-0737, Supplement 1, is specified in SSAR Section 18.2 and is incorporated in the DAC. This is acceptable to the staff.

Each of the paragraphs of Section 4.1 of NUREG-0737, Supplement 1, contains specific details pertaining to the SPDS. Some areas that require further consideration by the COL applicant in the design process are noted below.

- Paragraph 4.1a calls for a "concise" SPDS display. The concept of a "concise" display is further amplified in NUREG-1342, Section III.A.1. It is not clear from the description of the ABWR SPDS how the "concise" criteria will be met. This issue will be part of the COL applicant's detailed design development process.
 - Paragraph 4.1c requires procedures and training on the SPDS. This will be addressed by the COL applicant in the development of procedures as addressed in SSAR Chapter 13.
 - Paragraph 4.1d was addressed in GE's DSER (SECY-91-320) response, in which GE stated that the selection of information for inclusion in the SPDS display was based on the current BWR Owners Group (BWROG) EPGs rather than the ABWR EPGs. GE stated that this would be corrected to specifically address the ABWR EPGs. This was DFSEER Confirmatory Item 18.3.2.2-1. The SSAR has been modified to state that "selection of the parameters for inclusion in the SPDS display is based upon the Advanced Boiling Water Reactor Emergency Procedure Guidelines" Therefore, this item is resolved.
 - Paragraph 4.1f addresses the functional information required in the SPDS. While the specific parameters are to be determined by the COL applicant, GE has provided specific parameters in SSAR Chapter 18.4.2.11 that follow the NUREG-1342 recommendations, except as noted below.
- The COL applicant should consider the need for some standby liquid control (SLC) indication along with the reactivity control covered in the ABWR parameters. Existing BWR designs do not have this indication; however, because the ABWR is a new reactor design, the COL applicant should consider the items listed in NUREG-1342 as "desirable enhancements." Also, the comment on PRA insights from Paragraph 3.8a above and the fact that SLC is needed to mitigate the anticipated transient without scram (ATWS) sequence (which is 31 percent of CDF in the probabilistic risk analysis) in the ABWR illustrates the importance of having SLC indication.
 - GE states that indication of radioactivity control will be selected by the first ABWR license applicant. Thus, parameter selection and display implementation will be the responsibility of the COL applicant as part of the design development.

DSER (SECY-91-320) Issue 18.18 concerning the SPDS was resolved, and the remaining issues listed above will be addressed as part of the COL applicant's design process through incorporation into the HFE issue tracking system. This was DFSEER Confirmatory Issue 18.3.2.2-2. GE has included the SPDS design in SSAR Section 18.8, "COL License Information," as Item 18.8.4 and has committed to meeting NUREG-0737, Supplement 1, and the SPDS guidance in SSAR Sections 18.2 and 18.4.2.11. This is acceptable; therefore, this item is resolved.

EXEMPTION FROM 10 CFR 50.34(f)(2)(iv) FOR AN SPDS CONSOLE

The regulation 10 CFR 50.34(f)(2)(iv) requires that an application:

Provide a plant safety parameter display console that will display to operators a minimum set of parameters defining the safety status of the plant, capable of displaying a full range of important plant parameters and data trends on demand, and capable of indicating when process limits are being approached or exceeded (I.D.2).

GE, as part of the ABWR SSAR, commits to meet the intent of this requirement. However, as discussed below, the functions of the SPDS will be integrated into the control room design rather than on a separate "console." The purpose of the requirement for an SPDS, as stated in NUREG-0737, Supplement 1, is to ". . . provide a concise display of critical plant variables to the

Human Factors Engineering

control room operators to aid them in rapidly and reliably determining the safety status of the plant. ... and in assessing whether abnormal conditions warrant corrective action by operators to avoid a degraded core."

The ABWR design does not provide a separate SPDS, but rather, the functions of the SPDS are integrated into the overall control room display capabilities. In lieu of the requirements in 10 CFR 50.34(f)(2)(iv) for a "console," GE has proposed the following commitments in the ABWR SSAR:

- (1) Section 18.2(6) states that the functions of the SPDS will be integrated into the design,
- (2) Section 18.4.2.1(14) states that the SPDS function will be part of the plant summary information which is continuously displayed on the fixed-position displays on the large display panel,
- (3) Section 18.4.2.8 states that the information presented in the fixed-position displays includes the critical plant parameter information, and
- (4) Section 18.4.2.11 describes the SPDS for the ABWR and states that the displays of critical plant variables sufficient to provide information to plant operators about the following critical safety functions are continuously displayed on the large display panel as an integral part of the fixed-position displays:
 - (a) Reactivity control,
 - (b) Reactor core cooling and heat removal from the primary system,
 - (c) Reactor coolant system integrity,
 - (d) Radioactivity control, and
 - (e) Containment conditions.

The Commission may, upon its own initiative or at the request of an applicant, grant exemptions from the requirements of the regulations of Part 50. The exemption must comply with 10 CFR 50.12(a) criteria regarding special circumstances. An exemption from the "console" of the SPDS may be granted since not having an SPDS "console" (1) does not present an undue risk to the public health and safety, and is consistent with the common defense and security (10 CFR 50.12(a)(1)); and (2) special circumstances exist that application of the regulation to the ABWR design of the SPDS rule is not necessary to achieve the

underlying purpose of the SPDS rule (10 CFR 50.12(a)(2)(ii)). As presented here, the staff uses the special circumstances in 10 CFR 50.12(a)(2)(ii) to justify the deviation from the regulation (exemption) for an SPDS "console" for the ABWR design.

In conclusion, the staff finds an exemption from the requirement for an SPDS "console" to be appropriate based upon (1) the description in the GE SSAR of the intent of the ABWR design to incorporate the SPDS function as part of the plant status summary information which is continuously displayed on the fixed-position displays on the large display panel; and (2) a separate "console" is not necessary to achieve the underlying purpose of the SPDS rule which is to display to operators a minimum set of parameters defining the safety status of the plant. The staff, therefore, finds that GE has adequately supported an exemption from 10 CFR 50.34(f)(2)(iv) because SSAR Sections 18.2(6), 18.4.2.1(14), 18.4.2.8 and 18.4.2.11 achieve the underlying purpose of the rule by ensuring that the SPDS functional requirements are satisfactorily incorporated in the control room design without a separate "console."

Standard Feature Q

Feature Description - The use of fixed-position alarm tiles on the large display panel.

Feature Evaluation - This feature, as presented, is supported by GE's design analyses and evaluations as discussed in Section 18.3.2.1 above. In addition, the use of fixed-position alarms for key parameters is supported by research and industry experience with advanced alarm systems.

Standard Feature P

Feature Description - The application of alarm processing logic to prioritize alarm indications and to filter unnecessary alarms.

Feature Evaluation - This feature, as presented, is supported by GE's design analyses and evaluations as discussed in Section 18.3.2.1 above. GE's alarm prioritization scheme is based on three basic principles: mode suppression, redundancy suppression, and consequence suppression. GE does not propose advanced alarm suppression techniques, such as expert system-based analyses. Thus, the suppression is based on well-understood techniques that have been tested in other nuclear industry studies and have been found to be beneficial. In addition, the design specifies operator

control over suppression. Operators may turn suppression on or off. As with other CR features, the alarm processing techniques must be verified and validated as required by the V&V ITAAC in Table 3.1 of the CDM and SSAR Appendix 18E analyses.

It should be noted that this standard feature only includes the application of alarm processing logic to prioritize alarm indications and to filter unnecessary alarms. The general issue of alarm processing is the subject of much industry and NRC research, and much additional research data pertaining to alarm implementation will be available for the COL applicant during the CR design process.

Standard Feature Q

Feature Description - A spatial arrangement between the large display panel, the main control console, and the shift supervisor's console allows the entire CR operating crew to conveniently view the information presented on the large display panel.

Feature Evaluation - This feature, as presented, is generally supported by GE's design analyses and evaluations as discussed in Section 18.3.2.1 above.

Standard Feature R

Feature Description - The use of VDUs to provide alarm information in addition to the alarm information provided by the fixed-position alarm tiles on the large display panel.

Feature Evaluation - This feature, as presented, is generally supported by GE's design analyses and evaluations as discussed in Section 18.3.2.1 above. Component-level alarms will be provided on main console VDUs while important alarms and system-level alarms will be provided on the fixed-position tiles on the large display panel. The alarms presented on the large panel also will be available on the VDUs.

18.3.3 Finding

The staff did not identify any safety issues in the description of the standard features for the level of design detail presented. It is important to note that the description of the standard features is at a general level and essentially presents a general approach to CR design. The features are not a final design specification and are not described at a level of detail sufficient for a final safety determination without consideration of the design development process. When the design becomes more detailed as part of the implementation and integration of the design process, the COL applicant will evaluate the standard features in accordance with Item 18.8.5 in SSAR Section 18.8, the

V&V ITAAC in Table 3.1 of the CDM, and the SSAR Appendix 18E analyses.

DSER (SECY-91-320) Issues 18.02, 18.06, 18.08, 18.09, 18.11, and 18.12 became DFSER Confirmatory Item 18.3.3-1. As discussed above, GE amended the SSAR and included Appendix 18G, which satisfactorily addressed each of these issues. Therefore, these items are resolved.

DSER (SECY-91-320) Issue 18.18, "SPDS Integration With Related Emergency Response Capabilities," will be addressed by the COL applicant and has been included as Item 18.8.4 in SSAR Section 18.8. This is acceptable; therefore, this item is resolved.

18.4 Inventory of Controls, Displays, and Alarms

18.4.1 General Discussion in the Standard Safety Analysis Report

GE's initial SSAR contained insufficient information about controls, displays, and alarms to be utilized for the ABWR CR, resulting in a DSER (SECY-91-320) open issue. As part of the general resolution of the lack of CR detail, GE provided the detailed CR design implementation process through which the specific controls, displays, and alarms will be specified and designed. However, to provide an initial set of controls, displays, and alarms for transient mitigation before design certification, GE developed the inventory presented in SSAR Appendix 18F. This inventory was developed by analyzing the ABWR EPGs and the important operator actions specified as a result of the ABWR PRA analysis (refer to SSAR Section 19.D.7). Subsequently, GE described an additional fixed-position subset of these controls, displays, and alarms (i.e., a minimum inventory) for inclusion in the design description (DD) and ITAAC (CDM Section 2.7.1 and Table 2.7.1.A) for the main CR panels.

As part of the review, GE submitted to the staff detailed task analysis information (formerly SSAR Tables 18F-1 through 18F-12, Amendment 21) to support the identification of important displays, controls, and alarms for EPG implementation. The staff's review and its comments were given in the DSER (SECY-91-320) and DFSER, respectively. In response to the staff comments, GE submitted revised inventory analyses in SSAR Amendments 25 and 30 and in a letter providing proprietary task analysis information dated June 9, 1993. In Amendment 34, GE submitted SSAR Appendix 18H which provided the supporting analysis for the inventory of controls, displays, and alarms in SSAR Appendix 18F. The staff reviewed these submittals and confirmed that the

Human Factors Engineering

issues described in the DSER and DFSER were satisfactorily addressed. Each confirmatory item is discussed below. The staff's review of the issue is complete, the DFSER confirmatory items are considered resolved, and the minimum inventory of displays, controls, and alarms is adequate.

18.4.2 Analysis

18.4.2.1 DSER (SECY-91-320) Issue Resolution

One DSER (SECY-91-320) issue related to the inventory is summarized below along with the path to resolution that was proposed in the discussions with GE. The evaluation of the issue is given in the following section.

DSER (SECY-91-320) Issue 18.26: Inventory

The staff raised general questions in the DSER (SECY-91-320) about the absence of level of detail for controls, displays, and alarms to be incorporated into the main CR. GE and the staff agreed that since a detailed design regarding controls and displays would not be available for the design certification review, GE would use the EPGs to develop an inventory of the key minimum information and control requirements for the operator to perform necessary safety-related functions. This approach to developing an inventory is discussed below.

18.4.2.2 Evaluation of the Current SSAR

18.4.2.2.1 Review Methodology

The staff reviewed SSAR Sections 18.3.1, 18.3.3, and Appendices 18A, 18F, and 18H to determine if the inventory in Appendix 18F provided a reasonable minimum set of fixed controls, displays, and alarms to adequately implement the EPGs for the ABWR.

The analysis methods used for this evaluation included

- EPG Review: Selected steps of the EPGs were compared with the corresponding portions of SSAR Tables 18H-1 through 18H-14 to determine accuracy and technical validity of conclusions.
- PRA/Human Reliability Analysis (HRA) Review: The PRA/HRA was compared with SSAR Tables 18H-1 through 18H-14 to determine whether significant human actions were selected and if the analysis was correct.
- Summary Table Review: Summary SSAR Tables 18H-11 through 18H-13 were selectively compared with SSAR Tables 18H-1 through 18H-10 for accuracy.

GE's analysis process for the EPGs provided a large amount of specified equipment. Each step, caution, and note in the large body of EPGs was separately reviewed, analyzed, and documented in a table containing 14 columns. A number of important controls, displays, and alarms were identified. On the basis of discussions with GE, the staff determined that the results of the analysis would be provided to the COL applicant for use in the CR design implementation process. This will help ensure that the important indications, controls and alarms derived from the analysis are appropriately implemented in the HSI design. The staff has determined that GE's analysis process is acceptable.

18.4.2.2.2 General Results

(1) Level of HSI Detail

Discussions in earlier SSAR Section 18F.1 indicated that GE had made significant HSI design implementation decisions regarding displays and controls in the main CR. This contradicted other statements in SSAR Chapter 18. For example, GE stated: "In Tables 18F-1 through 18F-12, the particular method of design implementation for each control, display, and alarm function is indicated in brackets as part of each relevant table entry." Particular implementation methods were described for hundreds of items. In general, the bracketed information in the inventory was more detailed than what the rest of the SSAR supported. GE also stated that "all remote control equipment of a particular system can be accessed and controlled by touch operations when the VDU operate mode is selected." However, touch operations are not part of the standard features identified in the SSAR. GE committed to revise the discussions in Table 18F-1 and the output from the inventory to make them consistent with the remainder of SSAR Chapter 18.

In Amendment 25 of the SSAR, GE provided a revised version of the inventory that addressed the staff concerns about the level of HSI detail. The staff reviewed the revisions and found GE had adequately removed the specification of design implementation for each control, display, and alarm and had adequately revised the introduction to Appendix 18E to reflect this. The staff found these revisions acceptable; therefore, the issue is resolved.

(2) Fixed Versus VDU Selection Rationale

For each step of the EPGs and each important PRA operator action, GE defined the information

(Column 4) and control functions (Column 5) necessary to perform that step. The parameter displays (Column 6), controls (Column 7), and alarms (Column 8) that are needed also were defined. GE further specified whether they were fixed or on VDUs. GE, however, did not provide information as to how the use of fixed displays or VDU displays was determined. GE committed to provide the appropriate criteria and rationale for this determination in SSAR Appendix 18F.

As a result of further discussion with the staff, GE agreed to remove the VDU designation from the inventory list since specifying an implementation strategy before implementing the human-machine interface design implementation process would be premature. In SSAR Amendment 30, GE provided a revised version of the inventory that addressed the staff concerns about VDU selection rationale by removing the reference to design implementation detail (e.g., VDU). This is acceptable; therefore, the issue is resolved.

(3) HFE Input

Although the inventory contains a list of key minimum displays, controls, and alarms necessary to carry out operator actions associated with the EPGs, the COL applicant will need to identify and further define additional detailed characteristics of these displays and controls (e.g., ranges, scales, physical dimensions, and actual information presentation) during the detailed task analysis and HSI design efforts. On the basis of its discussions with GE, the staff concluded that at the time in the design implementation process the inventory is used for the actual CR design, the COL applicant will review the minimum inventory against the results of the detailed task analyses to ensure that the appropriate information is displayed for operations activities. Any discrepancies identified from this comparison will be documented and resolved by the COL applicant through the design process. As a result, the minimum inventory has been added to the required information to be used in the detailed HSI design element of the human-machine interface design implementation process. Therefore, the staff concerns regarding HFE input have been addressed as noted before and this issue is resolved.

(4) Use of Important Operator Actions

GE's earlier SSAR Section 18F-2 listed five actions considered to be important based on the results of GE's PRA sensitivity study. However, the SSAR

did not contain a discussion of the rationale for the selection of these as the important actions. Additionally, Item 7 of the PRA/HRA review of March 1991 contained a discussion of the number of human errors in the HRA. Six errors were noted to be the only ones described in any detail in the PRA. Of these six errors, which were treated as important in the PRA, only one was on the list of five actions considered in the inventory development (failure to inject water from the facility fire protection suppression system into the reactor pressure vessel (RPV)). The others related to inhibiting ADS during an ATWS, initiating SLC during an ATWS, controlling flow during an ATWS, failing to depressurize the reactor, and failing to isolate a failed heat exchanger. These errors did not appear to be analyzed for inclusion in the inventory. GE stated that some of these were covered in EPG steps and, hence, were addressed in the inventory. Others were not included because of automation decisions that removed the need for certain operator actions (e.g., ATWS control). GE agreed to provide a discussion of the rationale for the selection of the important operator actions included in the inventory in the PRA discussion in Chapter 19 of the SSAR. GE also agreed to identify each operator action already covered in the body of the EPGs that was also identified through the PRA study. Finally, GE committed to update the inventory based on any additional important human actions from PRA after the PRA/HRA had been completed.

As a result of further discussion with the staff, GE revised SSAR Section 19.D.7 to discuss the operator actions considered important in the ABWR PRA. Additionally, the GE revised detailed design files (e.g., SSAR Tables 18F-1 through 18F-11) submitted on June 9, 1993, to reflect the operator actions identified through the PRA. Currently, SSAR Appendix 18H contains those detailed design files. Therefore, the staff concerns about the use of important operator actions as identified in Chapter 19 have been addressed in the inventory described in Appendix 18F of the SSAR. The staff finds this approach to be acceptable; therefore, this issue is resolved.

(5) Scope of the Inventory

GE has developed a minimum set of fixed displays, controls, and alarms required to mitigate transients and accidents associated with the EPGs and the PRA sensitivity study. It should be noted, however, that GE committed to providing additional

Human Factors Engineering

fixed displays (e.g., in SSAR Section 18.4.2.11 on the SPDS and in Section 18.4.2.11 on dedicated hardware alarm windows of entry conditions for EOPs) beyond those identified in SSAR Tables 18H-1 through 18H-14. GE also identified additional fixed displays as part of the inventory analysis that were not considered as part of the minimum inventory.

GE committed to revising the SSAR to describe the scope of the inventory as limited to the EPGs and the PRA sensitivity study and to clarify that additional required fixed displays would not be superseded by this inventory. GE also agreed to provide a discussion in the introduction to Appendix 18F of the SSAR on the scope of the inventory and to describe the integration of the inventory with the detailed design process by the COL applicant.

SSAR Amendment 30 included a discussion on the scope of the inventory that was consistent with the scope previously established by the staff for the inventory specification. In addition, the introduction to Appendix 18F notes that other requirements on the CR panel inventory stemming from other design commitments were not necessarily incorporated into the minimum inventory. The staff reviewed the revised introduction to this appendix and found that it addressed its concerns about the scope of the minimum inventory. The issue is resolved.

18.4.3 Findings

In the DFSER, the staff concluded that GE had developed an acceptable minimum set of displays, controls, and alarms that will mitigate transients and accidents associated with the EPGs and the PRA sensitivity study subject to the incorporation of the comments in the DSER (SECY-91-320). As a result of further discussions with GE, the staff determined that the revision to SSAR Amendment 30, Appendix 18F, addressed its concerns and was, therefore, acceptable. The minimum inventory of displays, controls, and alarms described by GE is adequate; therefore, DSER (SECY-91-320) Issue 18.26 is resolved.

18.5 Remote Shutdown System

18.5.1 General Discussion in the Standard Safety Analysis Report

SSAR Sections 7.4 and 18.5 describe the RSS. The RSS will use conventional hardwired controls and indicators to

maintain diversity from the main CR. Discussion of other HSIs outside the CR are tied to GE's design and implementation process contained in SSAR Appendix 18E.

18.5.2 Analysis

18.5.2.1 DSER (SECY-91-320) Issue Resolution

DSER (SECY-91-320) Issue 18.19: Remote Shutdown System Design Rationale

In the DSER (SECY-91-320), the staff stated that additional information (i.e., tests, evaluations, and results) addressing how human performance is affected when operators are required to use mixed control and display technologies (i.e., digital and analog) during emergency plant operations was needed to support GE's position and rationale on the RSS design for the ABWR. (This issue is discussed further in Section 18.5.2.2 below.)

Evaluation: GE stated that the RSS will not use digital technology in order to maintain diversity from the CR. An assessment of the mix of analog and digital technologies in the plant as a whole was to be included in the postcertification test activities conducted by the COL applicant as part of the V&V element. The staff noted that independence (i.e., isolation and separation) and diversity were needed for the RSS. This was DFSER Confirmatory Item 18.5.2.1-1. GE has included information regarding the RSS as a COL action in Item 18.8.6 of SSAR Section 18.8 and in the HFE design acceptance criteria (ITAAC) of Table 3.1 of the CDM. This approach is acceptable; therefore, this item is resolved.

18.5.2.2 Evaluation of the Current SSAR

SSAR Sections 18.5 and 7.4 describe the RSS. GE intends the RSS to use conventional, hardwired controls and indicators to maintain diversity from the main CR. One generally acknowledged HFE design principle is to maintain consistency across HSIs when similar tasks are being performed in order to (1) minimize the time operators must spend "switching gears" to adopt operations to different HSIs for similar tasks and (2) minimize the potential for human errors that arise from incorrect transfer of learned activities from one HSI to another. In response to RAI Question 620.32 concerning the possible human factors implications of using analog hardware in the RSS design rather than a digital design consistent with the main CR, GE submitted a rationale for the diversity that included protecting "against the improbable event of common mode hardware or software failure in the plant instrumentation and control systems." The importance of diversity is acknowledged; however, caution should be exercised with regard to when the diversity is applied.

The staff was, therefore, concerned with GE's rationale for using mixed HSI technology. In its response to RAI Question 620.32, GE stated that the human factors testing of the RSS would be conducted during part of the plant power ascension test program and that, because RSS operations are relatively simple, training the operators to adjust to the analog RSS should not be an undue burden on them. GE did not give the basis or rationale to support these conclusions. As discussed above, the issue is being incorporated as Item 18.8.6 in SSAR Section 18.8 as a COL action item, thus addressing the staff's concern.

An NRC human factors generic issue includes the RSS. An evaluation of the risk significance of this issue for an RSS and a value-impact assessment of proposed human factors fixes is documented in NUREG/CR-5572, "An Evaluation of the Effects of Local Control Station Design Configurations on Human Performance and Nuclear Power Plant Risk" (1992). This study showed that functionally centralizing (integrating many functions into one panel) the RSS into one integrated panel was risk significant. Since the issue of RSS HSI design versus the CR HSI can only be addressed when design detail is developed, the staff expects that RSS functional centralization will be addressed by the COL applicant during the detailed design process described in SSAR Appendix 18E.

18.5.3 Finding

The design of the RSS is covered under the COL applicant's design and implementation process addressed by the HFE ITAAC in Table 3.1 of the CDM and detailed in SSAR Appendix 18E. DSER (SECY-91-320) Issue 18.19 is resolved on inclusion of Item 18.8.6, a COL action item, in SSAR Section 18.8.

18.6 Local Valve Position Indication

DSER (SECY-91-320) Issue 18.20: Local Valve Position Indication

In the DSER (SECY-91-320), the staff stated that the ABWR design should include complete local valve position indicator (VPI) based on accepted human factors principles and practices. "Local" in this instance means at the location of the valve in the plant. However, GE's commitment to VPI was unclear. This was identified as DSER (SECY-91-320) Issue 18.20.

Evaluation: In its response of October 1, 1991 (Response 5g), GE stated that only valves in the CR task analysis are required to have positive position indication. In the DSER (SECY-91-320) issues response dated February 18, 1992, GE stated that "the ABWR design does not include requirements for local position indication on all valves."

However, the response did not indicate which local valves will have position indication.

In the process of plant and system design development, including CR task analyses, a nuclear power plant vendor will determine the valves that require remote position indication in the CR. In the large majority of cases, the valves with remote VPI in the CR are motor-operated valves. However, in some cases of hydraulically operated valves, pneumatic valves, and even manual or check valves (e.g., valves for the low-pressure coolant injection line for BWRs) have CR VPI. The ABWR should have a full complement of CR VPI, with specific details (e.g., which valves and what type of displays) determined by the COL applicant through the approved CR design implementation process plan.

While the NRC has focused on CR VPI, developments over the past few years have shown the importance of VPI at the valve itself. The recognition of the need to know valve position has come about through review of operating events and an increased attention to good human-systems interface design, including VPI for manual valves. These developments indicate the need for local VPI at various types of valves (e.g., manual and motor operated).

Brookhaven National Laboratory (BNL) reviewed historic records (BNL Report A-3972-4-91) to determine the extent and type of human engineering deficiencies that exist at local control stations (LCSs) in actual plants and to determine the type of plant-level problems caused by these deficiencies. Table 4.b of the BNL report catalogues a number of such problems as a result of inadequate VPI identified in licensee event reports (LERs), NUREG/CR reports, and EOP inspections. Table 5 in the BNL report also identifies problems related to inadequate VPI and discusses an Institute of Nuclear Power Operations recommendation for local means of visually verifying actual and normal position of valves.

During an NRC research project, related to potential upgrades to LCSs, both the costs and benefits associated with improved VPI for manual valves were investigated. This project showed that human factors improvements (which included VPI as a key component) to only selected important manual valves could result in risk changes of about 10^{-5} core damage events per reactor-year. Additionally, cost analyses showed these upgrades or backfits to the important manual valves were cost beneficial. Further, cost analysis showed that most of the costs were related to the backfit situation. That is, when local VPI was included as part of the original valve design specification, the added costs were minimal, especially

Human Factors Engineering

when compared to the increased assurance that valves were positioned properly at all times. Thus, it was concluded that most local valves for new plants should have VPI. For certain small local manual valves (e.g., root valves), VPI was determined not to be necessary.

The staff's position is that remote VPI should be provided in the CR as discussed above and local VPI should be provided for

- all power-operated valves (e.g., motor, hydraulic, and pneumatic)
- all large manual valves (5 cm (2 in.) or greater)
- those small manual valves (< 5 cm (2 in.)) determined to be important

The staff noted that DSER (SECY-91-320) Issue 18.20 will be addressed by the COL applicant as part of the design development process. This was DFSEER COL Action Item 18.6-1. GE has included a COL action item regarding VPI in Item 18.8.7 of SSAR Section 18.8. This is acceptable.

18.7 Unresolved and Generic Safety Issues

18.7.1 General Discussion in the Standard Safety Analysis Report

As required by 10 CFR 52.47, the applicant for design certification must demonstrate compliance with any technically relevant portions of the TMI requirements in 10 CFR 50.34(f), which is sometimes referred to as the CP/ML rule. 10 CFR 52.47 also requires proposed technical resolutions of USI/GSI. The safety issues that relate to human factors are addressed in Section 18.7.2, which follows.

GE discusses its approach and proposed resolution of these issues in Appendix 19B to the SSAR. The staff's initial review of the issues related to human factors is documented in DSER (SECY-91-320) Chapter 18. Several of the items were left outstanding at that time. This section will address the human factors aspects of these issues, as well as those items left as outstanding in the DSER (SECY-91-320).

18.7.2 Analysis

18.7.2.1 DSER (SECY-91-320) Issue Resolution

Two DSER (SECY-91-320) issues related to the standard features are summarized below along with the path to

resolution that was proposed in the discussions with GE. The evaluation of these issues is given in the following section.

Issue 18.23: Unresolved Safety Issues and Generic Safety Issues (USI/GSI)s

In the DSER (SECY-91-320), the staff identified several USIs/GSIs (see discussion below in Section 18.7.2.2).

Evaluation: This issue was to be resolved as part of the design implementation process review (see discussion below in Section 18.7.2.2).

Issue 18.24: Construction Permit/Manufacturing License (CP/ML) Rule Issues

In the DSER (SECY-91-320), the staff identified several CP/ML rule issues (see discussion below in Section 18.7.2.2).

Evaluation: This issue was to be resolved as part of the design implementation process review (see discussion below in Section 18.7.2.2).

18.7.2.2 Evaluation of the Current SSAR

In its response dated February 21, 1992, GE addressed the specific outstanding items associated with 10 CFR 50.34(f) that were identified in the DSER (SECY-91-320). These are summarized below.

USI/GSI Item HF-1.1 and related items

The staff considers Items HF-1.1, "Shift Staffing," and I.A.1.4, "Long Term Upgrade of Operating Personnel and Staffing," to be beyond the scope of the design certification. The COL applicant will be responsible for addressing these issues as part of the licensing process. This was DFSEER COL Action Item 18.7.2.2-8. As Item 18.8.1 in SSAR Section 18.8, GE has included a general COL action item to conduct the detailed HFE design according to the design and implementation process defined by the DD and Table 3.1 of the CDM and SSAR Appendix 18E. The staff interprets this process to include the analysis of these USI/GSI items; therefore, this approach is acceptable.

USI/GSI Items HF-5.1 and related items

The staff considers Items HF-5.1, "Local Control Stations," HF-01.3.4.a, "Local Control Stations," and II.K.1(5), "Safety Related Valve Position Indication," to be beyond the scope of the design certification; the COL applicant will need to address these issues. This was

DFSER COL Action Item 18.7.2.2-7. GE has included information regarding these two issues as Item 18.8.11, "Local Control Stations," and 18.8.7, "Local Valve Position Indication," in SSAR Section 18.8. This approach is acceptable.

USI/GSI Items HF-5.2, B-17, and related items

The staff has reviewed the GE detailed CR design process and finds that the COL applicant will address several USI/GSI items as part of the detailed design implementation process. Among these items are (1) HF-5.2, "Review Criteria for Human Factors Aspects of Advanced Control Room Instrumentation"; (2) B-17, "Criteria for Safety Related Operator Actions"; (3) HF-01.3.4b, "Interface Annunciators"; (4) HF-01.3.4c, "Operational Aids"; (5) HF-01.3.4d, "Automation and Artificial Intelligence"; and (6) HF-01.3.4e, "Computers and Computer Displays." This was DFSER COL Action Item 18.7.2.2-6. As Item 18.8.1 in SSAR Section 18.8 GE, has included a general COL action item to conduct the detailed HFE design according to the design and implementation process defined by the DD and Table 3.1 of the CDM and SSAR Appendix 18E. The staff interprets this process to include the analysis of these USI/GSI items. In addition, for resolving Item B-17, GE has included an evaluation of the adequacy of the HSI to provide necessary controls, displays, and alarms for the timely performance of critical tasks as Item 18.8.15 in SSAR Section 18.8. This approach is acceptable.

TMI Action Item I.A.4.2

10 CFR 50.34(f)(2)(i) corresponds to TMI Action Item I.A.4.2 on simulator capabilities. GE states that "simulator facilities for use in performing operator training are outside the scope of the standard plant design certification." This is consistent with the treatment of training in SSAR Chapter 13 and is acceptable because training is to be addressed by the COL applicant. This was DFSER COL Action Item 18.7.2.2-1. GE, with the agreement of the staff, has included the requirement that the operator training program meet 10 CFR Part 50 as Item 18.8.8 in SSAR Section 18.8. This approach is acceptable.

TMI Action Item I.D.1

10 CFR 50.34(f)(2)(iii) corresponds to TMI Action Item I.D.1 on CR design. Item I.D.4 and Item I.D.5(1) also relate to CR design issues. GE states in the SSAR that these issues will be addressed by the COL applicant in the detailed design implementation process. This is acceptable and is further addressed by the HFE design process discussed in Section 18.8 below. This was DFSER COL Action Item 18.7.2.2-3. GE has included detailed

CR development in Table 3.1 of the CDM, Item 5, and in SSAR Appendix 18E. Information regarding this issue is given as a COL action item in Item 18.8.1 of SSAR Section 18.8. This approach is acceptable.

TMI Action Item I.D.2

Section 50.34(f)(2)(iv) corresponds to TMI Action Item I.D.2 on the SPDS. GE addressed this item in SSAR Section 18.4.2.11 (see the previous discussion of the SPDS in Section 18.3 of this report). Additionally, Item 125.I.3 in NUREG-0933 on SPDS availability will be addressed as part of the detailed CR design process. This was DFSER COL Action Item 18.7.2.2-4. GE has included the COL action of SPDS design in Item 18.8.4 of SSAR Section 18.8 and, in SSAR Sections 18.2 and 18.4.2.11, has committed to meeting NUREG-0737, Supplement 1, SPDS requirements. This approach is acceptable.

TMI Action Item I.D.3

10 CFR 50.34(f)(2)(v) corresponds to TMI Action Item I.D.3 on the status of bypassed and inoperable systems. This issue is covered in SSAR Chapter 7; however, the human factors aspects are not addressed. GE states that these will be addressed by the COL applicant in the detailed CR design implementation process. In addition, the COL applicant will be required to meet RG 1.47, which requires automatic indication at the system level of the bypassed or deliberately induced inoperable protection system and systems activated or controlled by the protection system. This was DFSER COL Action Item 18.7.2.2-5. GE has included a general COL action item to conduct the detailed HFE design according to the design and implementation process (defined by Table 3.1 of the CDM and SSAR Appendix 18E) in Item 18.8.1 of SSAR Section 18.8. The staff interprets this process to include the analysis of these USI/GSI items; therefore, this approach is acceptable.

TMI Action Items II.F.1 and II.F.2

These items address detailed CR design issues related to instrumentation (II.F.1, "Additional accident monitoring instrumentation" and II.F.2, "Instrumentation for Detection of Inadequate Core-Cooling"). GE states that these issues will be addressed by the COL applicant in the detailed design implementation process. This is acceptable and is further addressed by the HFE design process discussed in Section 18.8 of this report. This was DFSER COL Action Item 18.7.2.2-3. GE has included detailed CR development in Table 3.1 of the CDM, Item 5, and in SSAR Appendix 18E. The COL applicant's action with the process is included in Item 18.8.1 of SSAR Section 18.8. Further, GE has specifically identified COL

Human Factors Engineering

actions for each of these TMI issues in Items 18.8.13 and 18.8.14, respectively, of SSAR Section 18.8. This approach is acceptable.

18.7.3 Finding

The staff concludes that detailed resolution of the TMI, USI, and GSI technical issues are adequately addressed by GE's design and implementation process as reflected in SSAR Appendix 18E and COL license information given in SSAR Section 18.8.

18.8 Emergency Procedure Guidelines

The staff approved Revision 4 of the BWROG EPGs, in an NRC letter (A.C. Thadani to D. Grace) dated September 12, 1988. This revision of the EPGs formed the basis for the ABWR EPGs. In Appendix 18B of the SSAR, GE submitted to the staff a list of differences between the ABWR EPGs and the BWROG EPGs, Revision 4. The following is a summary of the major differences and a description of the unresolved item, F18.8.4-1, when the advance version of the SER was issued. The resolution of this item is addressed in Sections 18.8.2 and 18.8.4 of this report.

18.8.1 Containment Temperature

Section CN/T of the BWROG EPGs contains guidelines for monitoring and controlling containment temperature by using available containment cooling. The control functions specified in this section - operation of containment cooling, initiation of suppression pool sprays, and RPV depressurization when containment temperature cannot be maintained below a prescribed limit - are specified in Sections SP/T and DW/T of the ABWR EPGs. The steps in these sections are carried out concurrently. Step SP/T-1 directs the operator to initiate all available suppression pool cooling. Step SP/T-3 directs the operator to depressurize the RPV in accordance with the heat capacity temperature limit (HCTL) curve. Step DW/T-2 calls for the initiation of containment sprays.

Section CN/T of the BWROG EPGs was developed specifically for the BWR/6 Mark III containment where temperature can be controlled by the previously stated control functions. Although the ABWR containment design incorporates the concept of a Mark III suppression pool, it is analogous to a Mark II BWR containment design for the purpose of controlling the wetwell space temperature.

Because Section CN/T is design specific for the BWR/6 Mark III containment and the control functions specified in this section are carried out concurrently in other sections

in the ABWR EPGs, this section has been eliminated from the ABWR EPGs.

18.8.2 Venting

Revision 4 of the BWROG EPGs directs the operator to manually vent the containment before the primary containment pressure limit (PCPL) is reached in order to prevent uncontrolled containment failure. The PCPL is defined as the lowest pressure of the following: (1) pressure capability of the containment, (2) maximum containment pressure for vent valves to open and close, (3) maximum containment pressure at which safety/relief valves (SRVs) can open and remain open, and (4) maximum containment pressure for RPV vent valves to open and close for containment flooding. In the staff's SER for Revision 4 of the BWROG EPGs, the staff's stated goal was to limit venting to a "last resort" action. The major staff concern was centered on the appropriate containment pressure for venting. As a result, the venting pressure should be established to be as high as reasonably achievable.

In the ABWR, primary containment overpressure protection will not be accomplished through manual venting. This will be the function of the passive containment overpressure protection system (COPS), which is described in Section 19.2.3.3, "ABWR Containment Vent Design," of this report. The COPS is a passive system that is designed to actuate at 0.62 MPaG (90 psig) at 93 °C (200 °F) before the primary containment reaches a pressure corresponding to the ASME Service Level C of 0.67 MPaG (97 psig) at 260 °C (500 °F).

The COPS meets the primary containment overpressure protection philosophy described in the SER for Revision 4 of the BWROG EPGs because the set point is set as high as is reasonably achievable. Therefore, this change in vent design and the associated changes to the EPGs are acceptable.

The low pressure venting issue following the November 4, 1993, conference call with GE required GE to address the following items:

- (1) Revise EPGs (PC/P) to show that venting is restricted to the 5 cm (2 in.) line in the drywell.
- (2) Address suppression pool bypass mechanism through interconnection in the ACS and show the effect on the existing bypass analysis. Ensure that no other bypass pathways exist that have not been accounted for.

- (3) Address containment isolation configuration of interconnection in the ACS between the wetwell and drywell. GE should justify automatic control of the ACS over a normally closed penetration ensuring containment integrity.
- (4) Address suppression pool level issue in EPGs relating to the wetwell to drywell interconnection level. The EPGs appear to be inconsistent with the design.
- (5) Address suppression pool level and pressure control EPGs for injection from sources outside of containment. The EPGs appear to direct conflicting actions in that SP/L-3.3 directs operators to stop injection from sources outside containment when the suppression pool level reaches 27.2 m (89.5 ft). Whereas, PC/P-6 directs operators to spray the containment when the water level reaches 27.2 m (89.5 ft) with use of sources external to the containment.

Resolution of Items 1 and 5 was provided in Amendment 33. GE provided revised design information for Items 2, 3, and 4 in SSAR Amendment 34. For Item 2, the suppression pool bypass mechanism was shown to be insignificant when compared to the suppression pool bypass capability discussed in Section 6.2.1.1.5 of the SSAR. For Item 3, GE provided a description of the isolation provisions. For Item 4, GE modified the EPGs to specify the correct water level. The staff finds the information acceptable; therefore, the low pressure venting portion of Open Item F18.8.4-1 is resolved.

18.8.3 Drywell Spray Initiation Limit

The drywell spray initiation limit (DSIL) curve, used in Revision 4 of the BWROG EPGs, is defined to be the highest drywell temperature at which initiation of drywell sprays will not result in an evaporative cooling pressure drop to below either (1) the drywell-below-wetwell differential pressure capability or (2) the high drywell pressure scram set point. The curve contains a single peak. The curve to the left of the peak is a function of the high drywell pressure scram set point, while the right-hand side of the curve is limited by the drywell-below-wetwell differential pressure capability.

The DSIL curve used in the ABWR EPGs differs from the one used in Revision 4 of the BWROG EPGs. The curve to the right of the peak has been eliminated for the ABWR because a large pressure differential between the wetwell and the drywell at the onset of drywell sprays is not likely since the drywell and wetwell sprays are designed to actuate simultaneously. There will be some differential

pressure between the wetwell and the drywell as spraying proceeds because of different injection line-filling times and rates of spray flow into the two volumes.

GE calculated the differential pressure between the wetwell and drywell with concurrent spraying in the two volumes using the GE computer code SHEX. For the SHEX calculations, only six of eight vacuum breakers were assumed operable. Initial drywell pressure varied from maximum normal operating to maximum design, and relative humidity varied from 0 to 100 percent. Wetwell pressure varied from one atmosphere to normal operating and a wetwell temperature of 35 °F (95 °F) was assumed along with 100-percent relative humidity. The maximum drywell-to-wetwell pressure differential did not exceed -3.45 kPaD (-0.5 psid), which is less than the design valve of -13.7 kPaD (-2.0 psid) design value.

Actuation of drywell spray only is possible through a series of operator actions or following failure of the wetwell spray injection valve to open when containment sprays (both wetwell and drywell) are actuated. In a letter dated May 26, 1994, GE analyzed the effect of drywell spray actuation alone on the negative pressure capability of the containment and drywell/wetwell interface. GE concluded that the analysis supports all modes of spray actuation on the right side of the curve.

The staff finds the revised DSIL curve acceptable because the drywell and wetwell sprays will normally actuate simultaneously in the ABWR, thus eliminating the possibility of a significant pressure differential between the wetwell and the drywell at the onset of drywell sprays. This effect was confirmed by calculations performed by GE using the SHEX computer code. In the event of drywell spray actuation alone, GE provided analysis to demonstrate that the negative differential pressure capability of the containment would not be exceeded.

18.8.4 Heat Capacity Temperature Limit

The advance version of the SER identified Open Item F18.8.4-1 which consisted of two issues - the heat capacity temperature limit and the low pressure venting. The low pressure venting issue is discussed in Section 18.8.2 of this report.

For the ABWR, GE proposes the use of HCTL which would require reactor vessel depressurization when the suppression pool temperature reaches 103.9 °C (219 °F). Increasing the allowable suppression pool temperature before reactor vessel depressurization would begin permits the operation of the RCIC system for vessel injection when all other dedicated plant systems would be postulated to fail. This could occur during a SBO. This proposal raises

Human Factors Engineering

several phenomenology issues related to hydrodynamic loads.

Concerns were raised regarding the suppression pool with steam discharges from SRV or the RCIC turbine exhaust during pump operation, since the suppression pool temperature has traditionally been restricted by HCTL curves to ensure reactor vessel depressurization at 66 °C (151 °F). With a steam discharge from a SRV quencher or RCIC turbine exhaust sparger at suppression pool temperatures approaching 103.9 °C (219 °F), should a unstable steam condensation process occur, the containment liner may be subjected to an excessive buckling load from a low pressure region occurring at the containment liner/suppression pool water interface. Also, a suppression pool bypass issue arises if a steam plume extends from the quencher to the suppression pool surface. This was HCTL part of Open Item F18.8.4-1.

To resolve the above issue on unstable collapse for an extended plume where the steam jet extends beyond the quencher condensation zone, GE relied on testing performed by Drs. Chun and Sonin as described in GE's submittal of January 20, 1994. In current generation reactors, steam discharge from a SRV quencher is condensed within a cylindrical region about the quencher arms called a condensation zone. The radius is in part of function of the amount of sub-cooling which exists within the suppression pool during a discharge, with the basis for sub-cooling being set forth in NUREG-0783, "Suppression Pool Temperature Limits for BWR Containments."

With the proposal of permitting steam discharge in the suppression pool at a higher pool temperature than what has been traditional discussed with the NRC, the staff pursued the potential consequences, as follows;

- (1) Potential generation of a high quality steam plume extending beyond the quencher condensation zone with the plume being ingested by the ECCS inlet piping,
- (2) Potential for sudden collapse and an unacceptably high condensation oscillation (CO) load, should the steam plume discussed above become sufficiently buoyant to detach from the quencher source,
- (3) Potential extension of a steam plume from the quencher to the pool surface, thereby leading to a pool bypass path for particle scrubbing,
- (4) Potential for a steam discharge from the RCIC turbine exhaust sparger, causing a CO or chugging load higher than the CO or chugging load for LOCA or discharge of all SRVs.

The ABWR ECCS inlet piping is located approximately 1 m (3.3 ft) below the SRV quencher devices. The staff concluded that steam plume injection by the ECCS is not possible due to the buoyant nature of the steam plume.

In the January 20, 1994, submittal, GE presented a discussion by Dr. A. Sonin addressing the potential large steam plumes drifting into cooler region of the pool, thereby creating the initial conditions for sudden collapse of the plume. The conclusion that was reached in the above stated paper was that the cooler regions (with respect to the local temperature about the quencher which is discharging) of the suppression pool are at a low elevation and azimuthally away from the quencher. During the quencher discharge, a circulatory drift motion occurs as the surrounding water is entrained into the plume. As the pool temperature increases during an extended discharge, the area about the expanding steam plume is expected to be relatively well mixed horizontally. Thermal stratification will be primarily vertical, with the highest temperature being in the warm buoyant layer near the surface and the colder temperature near the bottom of the pool. The staff finds that GE's position that a condition where the steam plume could move from a warm region of the pool to a significantly colder region to be implausible is justified based on the above stated paper and experiments performed by Drs. Chun and Sonin.

A question was also discussed concerning a steam plume extending from the SRV quencher to the pool surface and creating a potential suppression pool bypass pathway, which would negate any scrubbing action by the water. This issued appears to be unfounded based on the discussion in the January 20, 1994, submittal. The argument against the notion of a long continuous high quality steam plume extending to the pool surface appears unlikely due to turbulence about the buoyant high velocity jet formed at the quencher hole. The turbulence caused at the plume in close proximity to the quencher entrains water into the plume from the sides causing rapid loss of plume temperature and steam volume fraction with increasing distance from the quencher. In addition, independent calculations by the staff show that the wetwell airspace pressurization during pool heatup produces sufficient pressure to maintain a minimum of 40 degrees K subcooling in the pool, based on bulk pool temperature. The staff concludes that a pool bypass is not a concern based on the proposed HTCL curve.

RCIC turbine exhaust discharge during suppression pool heat was reviewed for the potential of producing pool boundary loads which may exceed LOCA loads. This issued was raised because the turbine exhaust is discharged into the pool via a sparger which may not have the same performance features for condensing steam as a X-

Quencher. GE evaluated the sparger and has determined that the potential for producing CO and chugging loads greater than a LOCA seems unlikely based on a steam mass flux of about 48 kg/m²-sec (9.83 lbm/ft²-sec). At a mass flux of this magnitude, it is unlikely that CO and chugging loads could be produced which would be of the same magnitude as LOCA loads. In addition, the ABWR SSAR specifies a bounding asymmetric load case which assumes that half the drywell vents are 180 degrees out of phase with remaining vents for chugging. Based on the asymmetric loading requirement for chugging, the low mass flux at the sparger and that the sparger design has been in use on current-generation BWR without a reported failure or problem, the staff finds that the HCTL curve as drawn would not produce higher loads on the containment than LOCA loads currently assumed. This is acceptable and resolved HCTL portion of Open Item F18.8.4-1.

18.8.5 Primary Containment Flooding

An override statement has been placed in front of ABWR EPG Step C6-2. This step directs the operator to terminate all injection into the primary containment when drywell water level reaches the bottom of the RPV if containment radiation is greater than the core damage radiation level (CDRL) and RPV water level is below the top of the active fuel.

GE stated that containment flooding is to be terminated when the drywell water level reaches the bottom of the RPV during severe accident conditions when the core has melted through the vessel and dropped to the lower drywell. Flooding will be terminated to avoid covering the wetwell vent path, which has the containment rupture diaphragms. The wetwell vent is located at an elevation above the bottom of the RPV.

GE stated that it had set the CDRL at a level that will differentiate between an accident that has led to the melting of most of the fuel and an accident that results in damage of a few fuel pins. Once most of the fuel has melted the possibility of an ex-vessel event that leads to pressurization of the primary containment becomes much more likely. This pressurization may require actuation of the COPS. The staff finds this approach to primary containment flooding acceptable.

18.8.6 ATWS Stability Strategy

GE, in SSAR Amendment 32, submitted changes to the EPGs (departed from the Rev. 4 BWROG EPGs) incorporating ATWS stability strategy related to initiation of the SLC system and lowering of the RPV water level. This strategy is similar to that proposed by the BWROG for current BWRs and, in that context, is still a subject of

discussion between the staff and the BWROG. On the basis of its review (see Section 4.4 of this report), the staff concludes that the proposed processes for boron insertion and lowering of the RPV water level below the feedwater sparger level are acceptable for reducing large power oscillations to acceptable levels, should they occur, and for reducing power level in general for ATWS.

18.9 Design and Implementation Process

The final CR design is an area of rapidly changing technology and it is important that the Tier 1 certified DD and the ITAAC do not "lock in" a CR design that would be obsolete at the time of construction. The staff's approach for ensuring CR human factors considerations for design certification is to "lock in" a design process and specific DAC that, if met, would result in a design that is acceptable. GE provides an overview of the design process in SSAR Sections 18.3.4 and 18.7 and gives details in SSAR Appendix 18E. Section 3.1 of the CDM gives the HFE ITAAC and DAC associated with the design and implementation process. At the time of the DFSER review, the ITAAC and DAC "Design Description" for Section 3.1 of the CDM were not complete. This was identified as DFSER Open Item 18.9.1-1. The material was submitted by GE in SSAR Amendment 32 and is reviewed below in this section.

The DAC will be in the same format as the ITAAC used for other systems in that they will specify the certified design process commitment and the method of demonstration that the commitment has been met. The method of demonstration will be inspection, test, or analysis against established acceptance criteria. The CR acceptance criteria will describe a formal design implementation process with test, analysis, and acceptance criteria.

General acceptance criteria are specified in the Tier 1 ITAAC and DAC material for each of the program elements shown, along with specific criteria for submitting several COL applicant technical reports. The general criteria are derived from accepted HFE practices. The Tier 2 SSAR (Appendix 18E) material contains applicable guidance documents for the development of the material for each of the program elements.

The staff performed the safety evaluation of the design and implementation process using Tier 1 HFE ITAAC and DAC described in Table 3.1 of the CDM and the Tier 2 criteria that appear in SSAR Tables 18E-1 through 18E-4. Each table provides the Tier 2 acceptance criteria. The design and implementation process review is described in Section 18.9.2, which follows. The Tier 2 material contains all the requirements identified in the Tier 1

Human Factors Engineering

material and gives additional detail regarding how Tier 1 requirements are satisfied. The evaluation of the Tier 1 material is provided in Section 18.9.3, which follows.

The Tier 2 commitments described in the SSAR provide methods and descriptions of the implementation of the Tier 1 requirements. The determination that the plant has been constructed in accordance with the design certification will require the use of the information contained in both the Tier 1 and Tier 2 documents. The Tier 2 material contained in SSAR Appendix 18E was used to support the safety finding with regard to the design and implementation process. Thus, any change to the SSAR Appendix 18E commitments by the COL applicant would involve an unreviewed safety question and, therefore, would require NRC review and approval before implementation. Any requested change to SSAR Appendix 18E commitments shall either be specifically described in the COL application or be submitted for license amendment after COL issuance.

18.9.1 General Discussion in the Standard Safety Analysis Report

In GE's design and implementation process (as described in SSAR Appendix 18E and Table 3.1 of the CDM), the following HFE activities are defined:

- HFE design team
- HFE program and implementation plans
- system functional requirements analysis
- allocation of functions
- task analyses
- HSI design
- human factors V&V

GE's key HFE design activities and their acceptance criteria were developed to address the staff's HFE PRM. It is important to note that the ITAAC and DAC description as presented in Table 3.1 of the CDM and the Tier 2 criteria that appear in SSAR Tables 18E-1 through 18E-4 have a scope limited to the main CR and the RSS (as agreed to by the NRC), whereas the HFE PRM has a broader scope. In addition, although procedures are within the scope of the HFE PRM as Element 7, GE has not included procedures within the scope of SSAR Appendix 18E. Procedure development is addressed as a COL responsibility in SSAR Section 13.5. In addition to scope, there are other differences between the staff's HFE PRM and GE's design and implementation process. However, a high degree of similarity exists between the two. The relationship between the PRM elements and the SSAR Tier 2 criteria is given below.

<u>HFE PRM Element</u>	<u>SSAR Tier 2</u>
1	Table 18E-1
2	Table 18E-1
3	Tables 18E-1, 18E-2
4	Tables 18E-1, 18E-2
5	Tables 18E-1, 18E-2
6	Tables 18E-1, 18E-3
7	Section 13.5
8	Tables 18E-1, 18E-4

The following review focuses on (1) the evaluation of the acceptability of the differences between GE's process and the staff's PRM and (2) the closure of DSER (SECY-91-320) issues. Since the GE process was evaluated against the criteria depicted in the PRM, the organization of the report design and implementation plan review follows the organization of the PRM (i.e., Elements 1 to 8).

18.9.2 Analysis

18.9.2.1 DSER (SECY-91-320) Issue Resolution

The initial SSAR provided little detailed information about the ABWR HSIs. As part of the general resolution of the lack of design detail, GE committed to provide a detailed HFE design and implementation process through which the HSIs will be designed and evaluated. This became DSER (SECY-91-320) Issue 18.25. However, because many other DSER (SECY-91-320) issues addressed design detail and were, therefore, beyond the scope of the certified design review, they too became incorporated into the design process review for subsequent consideration by the COL applicant as design development proceeds. These DSER (SECY-91-320) issues are identified in Table 18.2. As indicated previously in Section 18.1 of this report, an eight-element HFE PRM was developed to provide review criteria for the process. Table 18.2 shows which HFE PRM element addresses each DSER (SECY-91-320) issue. The issue is discussed in the section of this report identified in the table.

18.9.2.2 Evaluation of the Current SSAR

One general open issue pertained to Tier 2 guidance descriptions for each of the GE process elements. Element 1 (in the HFE PRM), for example, specifies the documents that are to be used as guidance. Each element has a similar specification. When the DFSEER was issued, GE had not incorporated the specific documents, to serve as guidance, from the staff's review model into its process. The list of guidance documents was identified as DFSEER Open Item 18.9.2.2-1.

Table 18.2 DSER (SECY-91-320) HFE issues to be addressed in the process plan

Issue Number	Issue	SER Section	HFE PRM Element
18.04	Operator and system reliability	18.9.2.2.8	Element 8 - Verification & Validation
18.05	Operator workload analysis	18.9.2.2.8	Element 8 - Verification & Validation
18.06	Tests and analysis to support design implementation	18.9.2.2.8	Element 8 - Verification & Validation
18.07	ABWR HFPP	18.9.2.2.1	Element 1 - HFE Program
18.08	Control room prototype	18.9.2.2.8	Element 8 - Verification & Validation
18.10	Detailed task analyses	18.9.2.2.5	Element 5 - Task Analysis
18.13	HSI design requirements for CRT, flat panel, and large-screen displays	18.9.2.2.6	Element 6 - Interface Design
18.15	CRT display information	18.9.2.2.6	Element 6 - Interface Design
18.16	PGCS display reliability	18.9.2.2.8	Element 8 - Verification & Validation
18.17	Alarm suppression criteria, alarm points	18.9.2.2.6	Element 6 - Interface Design
18.21	Procedure development	18.9.2.2.7	Element 7 - Procedure Development

GE has incorporated the list into SSAR Appendix 18E, and the staff finds it acceptable based on a comparison to the HFE PRM; therefore, all of the basic documents necessary to conduct the design and implementation process that are identified in the HFE PRM have been incorporated into SSAR Appendix 18E. Therefore, this item is resolved.

For each of the following elements, the staff compared the criteria in the HFE PRM with GE's element criteria. If GE's criterion differs from the "general criterion" described in the HFE review model, the analysis of the difference is given.

18.9.2.2.1 Element 1 - Human Factors Engineering Program Management

GE addresses human factors engineering program management in its HFE program plan of the HFE ITAAC and DAC described in Table 3.1 of the CDM and in the Tier 2 description in SSAR Table 18E-1(II). GE's description of this element is substantially the same as the HFE PRM, Element 1 description. However, the staff identified and evaluated the following three exceptions. It also evaluated DSER (SECY-91-320) Issue 18.07 as part of this element:

(1) Operating Experience Review (OER)

GE incorporated OER (HFE PRM Element 2) into its HFE program plan instead of presenting it as a

separate element. The main purpose of this element is to ensure that the designer identifies HFE issues from current and past operating experience to be incorporated into the HFE issue tracking system. The merger of these elements does not compromise the contribution of the OER and is acceptable.

(2) Absence of System Safety Engineering Expertise on the Design Team

GE's design team does not include system safety engineering expertise as specified in the HFE PRM. In the May 1992 meeting, GE stated that the system safety engineers will be included as needed and not as full permanent members of the HSI design team. The staff finds GE's approach acceptable, because this area of engineering expertise is applicable to the HFE design rather than the other HFE elements of the process.

(3) Absence of Reliability, Availability, Maintainability, and Inspection Expertise on the Design Team

GE's design team does not include reliability, availability, maintainability, and inspection expertise as specified in the HFE PRM. In the May 1992 meeting, GE agreed to include this expertise in the description of the HSI design. This

Human Factors Engineering

is acceptable because this area of engineering expertise is applicable to the HSI design rather than the other HFE elements of the process.

DSER (SECY-91-320) Issue 18.07: ABWR Human Factors Program Plan (HFPP)

In the DSER (SECY-91-320), the staff stated that SSAR Section 18.3 gave an outline for systems analysis and HSI design, but because the ABWR HFPP was not provided or referenced, the section contained little detail about actual analysis steps and procedures and discussion of results. The staff needed additional detailed information to complete its review.

Evaluation: The design and implementation process described in the SSAR and in the HFE certification design material DD document adequately addresses the HFPP and the types of analyses to be performed. All relevant portions of the HFE model were incorporated into the GE documents. Therefore, DSER (SECY-91-320) Issue 18.07 was to be resolved subject to receipt of final ITAAC and DAC. This was DFSEER Confirmatory Item 18.9.2.2.1-1. GE has provided an acceptable plan as described in the ITAAC and in SSAR Appendix 18E (the following section addresses the technical justification of the plan's acceptability).

18.9.2.2.2 Element 2 - Operating Experience Review

As indicated above, GE addresses GER as part of the HFE program plan of the HFE ITAAC and DAC described in Table 3.1 of the CDM and in the SSAR Tier 2 description in SSAR Table 18E-1 (II).

In the DFSEER the staff identified an open item with regard to the OER. The OER for a specific list of issues still needed to be performed. The list of items had not been identified when the DFSEER review was conducted. The list of issues was DFSEER Open Item 18.9.2.2.1-1. GE has provided an acceptable approach to OER in Table 18E-1(II) in SSAR Appendix 18E. The main treatment of OER is included as Article e in that SSAR section. For the first ABWR implementation, a list of issues is identified and organized into topical areas such as CR design, computers, CRT displays, and anthropometrics. These issues were identified through a preliminary OER performed by GE and are required, according to the design and implementation process, to be included in the COL HFE issues tracking system. The issues were identified on the basis of a review of pertinent industry experience literature and detailed control room design reviews of predecessor plants. The basis for the identification of operating experience is consistent with the requirements of the PRM. In addition, experience reviews

are required in six selected areas, which GE has identified as ones for which further industry development is anticipated; thus, new issues are likely to emerge. These areas include on-screen controls, wide display panels, alarm prioritization systems, automation, VDU design, and workstation integration. The staff agrees that these are significant areas of HSI design and areas where significant technology development and operating experience are likely to occur. COL reviews in these areas include source material consistent with the PRM. New issues identified will be incorporated into the HSI tracking system. Subsequent ABWR COL applications would use the OER first implemented if no design changes were made. When changes are to be made, an OER is to be conducted using operator interviews and LERs of previous ABWR implementations. The staff interprets this requirement as also including a review of the documentation identified under Article 2e.(i)(b) of SSAR Appendix 18E; that is, the reviews will include industry experience, design, and research reports applicable to the areas of the design being modified. Where the changes represent a significant departure from the previous ABWR implementation, the staff expects applicable elements of the certified design and implementation process to be followed.

OER also is included as part of Article d of Appendix 18E for applications of HSI technologies that are different from those specified in SSAR Section 18.4.3.

On the basis of GE's commitment in SSAR Table 18E-1(II) for an OER to be conducted, this item is resolved.

18.9.2.2.3 Element 3 - System Functional Requirements Analysis

GE addresses system functional requirements analysis as part of the HFE ITAAC and DAC described in Table 3.1 of the CDM and in the Tier 2 description in Table 18E-1(III). GE's description of this element is substantially the same as the HFE PRM description for Element 3. However, the staff identified and evaluated the following three exceptions:

(1) Modification of General Criterion 2

This criterion defined critical functions as "those functions required to achieve major system performance requirements; or those functions which, if failed, could degrade system or equipment performance or pose a safety hazard to plant personnel or to the general public." GE has deleted from the definition the words "degrade system or equipment performance." The staff agrees with this

change because the initial model definition was so broad that, if applied, it could define all functions as critical. GE's later definition of critical functions is acceptable.

(2) Elimination of General Criterion 8

This criterion stated: "The function analysis shall be kept current over the life cycle of design development." GE justified the deletion of this criterion by stating that an iterative approach to design development does not affect the review process. The staff review focuses on the acceptability of the end products of the design process as defined by the ITAAC requirements in Table 3.1 of the CDM, regardless of the number of iterations the designers went through to derive the requirements. Therefore, GE's deletion of this criterion is acceptable.

(3) Elimination of General Criterion 9

This criterion addressed verification that "all the functions necessary for the achievement of safe operation are identified" and that "all requirements of each function are identified." GE's justification for eliminating this criterion was that it was covered by the quality assurance (QA) requirements of Appendix B to 10 CFR Part 50. The staff finds this change acceptable because (1) general verification is covered by QA and is addressed in the Tier 2 analysis report criteria for this element and (2) the HFE design team evaluation report will adequately address the verification aspects of this element. Therefore, GE's deletion of this criterion is acceptable.

18.9.2.2.4 Element 4 - Allocation of Function

GE addresses function allocation as part of the HFE ITAAC and DAC described in Table 3.1 of the CDM and in the SSAR Tier 2 description in SSAR Table 18E-1(IV).

GE's description of this element is substantially the same as the HFE PRM description for Element 4. However, the staff identified and evaluated the following two exceptions:

(1) Elimination of General Criterion 5

This criterion stated: "Functions shall be re-allocated in an iterative manner, in response to developing design specifics and the outcomes of ongoing analyses and trade studies." As in the case of the systems requirements analysis, as the design is modified as a result of design tradeoffs and

analyses, it is important to reevaluate the function assignments; however, an iterative approach to design does not affect the review process since the staff review focuses on the acceptability of the end products of the design process regardless of the number of iterations the designers went through to derive the results. Therefore, GE's deletion of this criterion is acceptable.

(2) Elimination of General Criterion 6

This criterion stated: "Function assignment shall be evaluated." Although this was eliminated as a general criterion (Tier 1) it was not removed from the Tier 2 requirements that elaborate on the evaluation specification. Because the general criterion did not provide any specific information and the evaluation of function assignment is maintained in the Tier 2 description, GE's deletion of this criterion is acceptable.

18.9.2.2.5 Element 5 - Task Analysis

GE addresses task analysis as part of the HFE ITAAC and DAC described in Table 3.1 of the CDM and in the SSAR Tier 2 description in SSAR Table 18E-1 (V). GE's description of this element is substantially the same as the HFE PRM description for Element 5. However, the staff identified and evaluated the following exceptions; it also evaluated DSER (SECY-91-320) Issue 18.10 as part of this element.

(1) Elimination of Part of General Criterion 4

This criterion stated: "The task analysis shall be iterative and become progressively more detailed over the design cycle. The task analysis shall be detailed enough to identify information and control requirements to enable specification of detailed requirements for alarms, displays, data processing, and controls for human task accomplishment." GE deleted the first sentence dealing with the iterative aspects of the analysis. This change is acceptable because an iterative approach to design does not affect the review process since the staff review focuses on the acceptability of the end products of the design process regardless of the number of iterations the designers went through to derive the results. Therefore, GE's partial deletion of this criterion is acceptable.

(2) Definition of PRA Critical Tasks

In SSAR Table 18E-1(V), "Task Analysis Implementation Plan," Item (1)(c) states: "Human

Human Factors Engineering

actions which are identified through PRA sensitivity analyses to have significant impact on safety shall also be considered critical tasks." The staff expects that the definition of critical tasks will include all PRA-defined human actions that are critical and that the definitions will not be limited to sensitivity analyses alone. SSAR Section 19D.7.6 defines important operator actions from the ABWR PRA as derived from the ABWR Level 1, Level 2, fire, flood, seismic, and shutdown analyses. These actions are to be included in the definition of critical tasks for task analysis and the HFE efforts associated with subsequent elements - HSI design, procedure development, and V&V. Therefore, the definition of critical tasks is acceptable.

DSER (SECY-91-320) Issue 18.10: Detailed Task Analyses

In the DSER (SECY-91-320), the staff stated that detailed task analyses, which should cover the full range of normal and off-normal plant operations, had not been performed. GE stated that the task analysis will be performed as part of the hardware and software procurement and design implementation activities.

Evaluation: Although the design commitment, ITAAC, and general criteria for task analysis in Table 3.1 of the CDM and in the SSAR Tier 2 description adequately address detailed task analyses, DSER (SECY-91-320) Issue 18.10 was to be resolved subject to receipt of the final ITAAC and DAC. This was DFSE Confirmatory Item 18.9.2.2.5-1. Task analysis is described in Table 3.1 of the CDM, Item 4. The analyses will address "the range of plant operating modes, including startup, normal operations, abnormal operations, transient conditions, low power and shutdown operations." This approach is acceptable and this item is resolved.

18.9.2.2.6 Element 6 - Human-System Interface Design

GE addresses HSI design as part of the HFE ITAAC and DAC described in Table 3.1 of the CDM and in the Tier 2 description in SSAR Table 18E-1 (Item VI). The description of this element is different from the HFE PRM Element 6 description. The staff identified and evaluated the following two exceptions. In addition, it evaluated DSER (SECY-91-320) Issues 18.13, 18.15, and 18.17 as part of this element.

(1) Elimination of General Criterion 6

This criterion stated: "The selection and design of HSI hardware and software approaches shall be

based upon demonstrated criteria that support the achievement of human task performance requirements. Criteria can be based upon test results, demonstrated experience, and trade studies of identified options." GE eliminated this element from its process description. It states that the list for Element 6 of specific documents to serve as guidance will be used by the COL applicant for the selection and design of HSI hardware and software approaches. The staff finds this to be an acceptable approach; therefore, GE's deletion of this criterion is acceptable.

(2) Elimination of General Criterion 7

This criterion stated: "HFE standards shall be employed in HSI selection and design. Human engineering guidance regarding the design of particular features, shall be developed by the HSI designer to (1) insure that the HSIs are designed to currently accepted guidelines and (2) insure proper consideration of human capabilities and limitations in the developing system. This guidance shall be derived from sources such as expert judgement, design guidelines and standards, and quantitative (e.g., anthropometric) and qualitative (e.g., relative effectiveness of differing types of displays for different conditions) data. Procedures shall be employed to ensure HSI adherence with standards." GE states that the specific documents listed in SSAR Appendix 18E will be used by the COL applicant to comply with this general criterion. The staff finds that the information in SSAR Appendix 18E related to this issue and GE's deletion of this criterion is acceptable.

DSER (SECY-91-320) Issue 18.13: HSI Design Requirements for Cathode Ray Tube (CRT), Flat Panel, and Large-Screen Displays

In the DSER (SECY-91-320), the staff stated that additional detailed information was needed on the ABWR HSI design requirements for the control station CRT, flat panel, and large-screen displays.

Evaluation: Although the design commitment, ITAAC, and general criteria for HSI design described in the CDM and in the SSAR Tier 2 description adequately address the detailed design of the HSI, DSER (SECY-91-320) Issue 18.13 was to be resolved subject to receipt of the final ITAAC and DAC. This was DFSE Confirmatory Item 18.9.2.2.6-1. Design Commitment 5 of Table 3.1 of the CDM and SSAR Appendix 18E states that the HSI will be based on requirements derived from task analyses and

designed using HFE criteria and guidance. The staff has compared GE's Design Commitment 5 and SSAR Appendix 18E with the HFE PRM and found GE's description to be acceptable. Therefore, this item is resolved.

DSER (SECY-91-320) Issue 18.15: CRT Display Information

In the DSER (SECY-91-320), the staff stated that no details of the CRT displays were provided to permit visualization of the actual information available to the operator. The staff needed this information to complete its review.

Evaluation: The design commitment, ITAAC, and general criteria for HSI design described in the DD of the CDM and in the SSAR Tier 2 description adequately address the detailed design of the HSI. DSER (SECY-91-320) Issue 18.15 was to be resolved subject to receipt of the final ITAAC and DAC. This was DFSER Confirmatory Item 18.9.2.2.6-2. Design Commitment 5 of Table 3.1 of the CDM and SSAR Appendix 18E state that the HSI will be based on requirements derived from task analyses and designed by the COL applicant using HFE criteria and guidance. This is acceptable based on a comparison of the design commitments with the HFE PRM. Therefore, this item is resolved.

DSER (SECY-91-320) Issue 18.17: Alarm Suppression Criteria, Alarm Points

In the DSER (SECY-91-320), the staff stated that additional detailed information about the ABWR alarm suppression criteria and rationale used to determine the limit number of alarm points that operators can simultaneously recognize was necessary for the staff to complete its review.

Evaluation: Although the design commitment, ITAAC, and general criteria for HSI design described in the CDM and in the SSAR Tier 2 description adequately address the detailed design of the HSI, DSER (SECY-91-320) Issue 18.17 was to be resolved subject to receipt of the final ITAAC and DAC. This was DFSER Confirmatory Item 18.9.2.2.6-3. Design Commitment 5 of Table 3.1 of the CDM and SSAR Appendix 18E state that the HSI will be based on requirements derived from task analyses and designed using HFE criteria and guidance. This is acceptable based on a comparison of Design Commitment 5 with the HFE PRM. Therefore, this item is resolved.

18.9.2.2.7 Element 7 - Plant and Emergency Operating Procedure Development

As stated in Section 18.9.1 above, GE has not included procedure development in the scope of its HFE design and implementation plan. Since procedure development is addressed as a COL applicant responsibility in SSAR Section 13.5, it is evaluated in Chapter 13 of this report.

18.9.2.2.8 Element 8 - Human Factors Verification and Validation (V&V)

GE addresses human factors V&V as part of the HFE ITAAC and DAC described in Table 3.1 of the CDM and in the SSAR Tier 2 description in Tables 18E-1(VII). GE's description of this element is the same as the HFE PRM description for Element 8. However, the staff identified and evaluated the following exception. It also evaluated DSER (SECY-91-320) Issues 18.04, 18.05, 18.06, 18.08, and 18.16 as part of this element.

Elimination of PRA/HRA-Defined Critical Actions From General Criterion 8

This criterion stated: "A verification shall be made that all critical human actions as defined by the task analysis and PRA/HRA have been adequately supported in the design. The design of tests and evaluations to be performed as part of HFE V&V activities shall specifically examine these actions." Risk-critical human actions (those to which the plant design is especially sensitive in a risk model) should receive special attention in the V&V process. GE committed to having all critical tasks confirmed as part of the V&V process and defined critical tasks under the task analysis element to include all PRA/HRA items included in SSAR Appendix 19D, Section D.7. These would include those operator actions that had significant safety impact. The staff considered this approach acceptable. The incorporation of the commitment was DFSER Confirmatory Item 18.9.2.2.8-1. SSAR Table 18E-1 states that critical tasks will include those human actions identified through PRA sensitivity analyses to be critical. Table 3.1 of the CDM, Item 6a, (4)(a), states that one V&V objective will be the "confirmation that the identified critical functions can be achieved using the integrated HSI design." GE's deletion of PRA/HRA-defined critical actions from this criterion is acceptable; therefore, this item is resolved.

Human Factors Engineering

DSER (SECY-91-320) Issue 18.04: Evaluation of Operator and System Reliability During Shift From Normal to Abnormal Operations

In the DSER (SECY-91-320), the staff stated that the net effect on operator and system reliability should be evaluated for normal operations and for the shift, or transition, from normal to emergency operations.

Evaluation: Incorporation of this analysis into the V&V ITAAC was DFSER Confirmatory Item 18.9.2.2.8-2. Table 3.1 of the CDM, Item 6a, states that the dynamic performance evaluations conducted as part of validation "shall be conducted over the range of operational conditions and upsets." The staff interprets this as including transitions from normal to emergency operations to permit the evaluation of the crew's ability to assume plant control under abnormal conditions, and notes that a detailed examination of operator transition from normal to abnormal operations will be performed during the V&V of the main CR and RSS by the COL applicant. Therefore, this item is resolved.

DSER (SECY-91-320) Issue 18.05: Operator Workload Analysis

In the DSER (SECY-91-320), the staff stated that GE did not make it clear what analyses had been performed in support of the design and development of the ABWR and what tests and analyses are yet to be done by the COL applicant.

Evaluation: In the DFSER, the staff stated that the V&V analyses conducted by the COL applicant as part of the HFE ITAAC and DAC described in Table 3.1 of the CDM specifically would address this issue, which would be resolved subject to receipt of the final ITAAC and DAC. This was DFSER Confirmatory Item 18.9.2.2.8-3. Design Commitment 6, Acceptance Criteria 6a(6)(d) of Table 3.1 of the CDM, and SSAR Appendix 18E state that workload will be used as a performance measure in dynamic performance tests. On the basis of a comparison of Design Commitment 6 and SSAR Appendix 18E with the HFE PRM, this item is resolved.

DSER (SECY-91-320) Issue 18.06: Tests and Analysis To Support Design Implementation

In the DSER (SECY-91-320), the staff stated that additional detailed information was needed about the methods, criteria, and results of analyses that support the level and type of staffing, automation, and function allocation to achieve the goals of safe and reliable performance of the operating crew and overall system. The staff further stated that the design bases in SSAR Section 18.2 would be more

appropriate as design requirements if they had been derived and justified on the basis of the systems analysis. The staff considered it more appropriate to develop design bases that are stated in terms that would help achieve the primary goal of developing interfaces (and a system) that make possible safe, efficient, and reliable operator performance. The bases could be described in "operator-centered" terms that can objectively be linked with achieving the design goals and serve as criteria for test and evaluation activities. There were two aspects of this issue to consider: (1) the analyses conducted to date and (2) the analyses that will be done in the future.

Evaluation: The analyses conducted to date were included in SSAR Appendix 18G and evaluated by the staff as acceptable (see detailed discussion of this review in Section 18.3.2.1 above). Although the V&V analyses to be conducted by the COL applicant as part of the HFE ITAAC and DAC described in Table 3.1 of the CDM specifically address this issue, it was to be resolved subject to receipt of the final ITAAC and DAC. This was DFSER Confirmatory Item 18.9.2.2.8-4. Design Commitment 6 of Table 3.1 of the CDM and SSAR Appendix 18E include a commitment that the V&V analyses will be performed. These analyses are acceptable (as stated above) on the basis of a comparison of Design Commitment 6 and SSAR Appendix 18E with the HFE PRM. Therefore, this item is resolved.

DSER (SECY-91-320) Issue 18.08: CR Prototype

In the DSER (SECY-91-320), the staff stated that development of a fully functional CR prototype of the standard design was appropriate to demonstrate acceptable human performance.

Evaluation: Although V&V analyses conducted by the COL applicant as part of the HFE ITAAC and DAC described in Table 3.1 of the CDM specifically require prototype evaluation, this issue was to be resolved subject to receipt of the final ITAAC and was DFSER Confirmatory Item 18.9.2.2.8-5. Design Commitment 6 of Table 3.1 of the CDM requires dynamic task performance test evaluations to be performed for HSI validation. SSAR Table 18E-1(VII), Item (1)(d), states that the dynamic task performance evaluations will be performed using "dynamic HSI prototypes, i.e., prototypical HSI equipment which is dynamically-driven by real time plant simulation computer models." On the basis of a comparison of Design Commitment 6 and SSAR Appendix 18E with the HFE PRM, the staff finds that GE's proposal is acceptable and, therefore, this item is resolved.

DSER (SECY-91-320) Issue 18.16: Power Generation Control System (PGCS) Reliability

In the DSER (SECY-91-320), that staff stated that additional detailed information was necessary regarding the reliability of the PGCS and the effect on operator performance and workload should it malfunction.

Evaluation: Since system malfunctions are required to be analyzed as part of the V&V effort under HFE model Element 8, the COL applicant will analyze malfunctions of the PGCS. This issue was resolved as DFSER COL Action Item 18.9.2.2.8-1 to be addressed as part of the HFE issue tracking system. GE has included this analysis in Item 18.8.10 of SSAR Section 18.8, which the staff finds acceptable. Therefore, this item is resolved.

18.9.3 Design Description Tier 1 ITAAC and DAC Review

While the above review was directed toward the acceptability of the design and implementation process as a whole, the staff performed a separate evaluation of the HFE ITAAC and DAC provided in Table 3.1 of the CDM to ensure that significant features of the design certification application contained in the SSAR were captured by Table 3.1 of the CDM. It should be noted that the materials reviewed in SSAR Appendix 18E were used to support the safety finding with regard to the design and implementation process. Thus, any change to the commitments in SSAR Appendix 18E would involve an unreviewed safety question and, therefore, require NRC review and approval before implementation. Any requested change to commitments in SSAR Appendix 18E shall either be specifically described in the COL application or be submitted for license amendment after COL issuance.

The review of the SSAR using the HFE PRM led to the staff's conclusion that the design and implementation process contained the necessary and sufficient aspect of an HFE program that were sufficient to result in an acceptable HSI design. The general guidance in SECY-92-287 was used to support the review of Table 3.1 of the CDM.

Table 3.1 of the CDM was compared to the major PRM elements to determine whether they were captured. No omissions were identified. Table 3.1 of the CDM was then evaluated to ensure that they accurately reflected the design and implementation process and that they were at a level of detail consistent with the staff's intent to not constrain the use of state-of-the-art, proven technology at the time the HSI is designed (one of the stated intents of the DAC process). All necessary and sufficient ITAAC were identified based upon comparison with the HFE PRM, and no concerns were identified.

Therefore, the staff concludes that the design commitments in the HFE ITAAC and DAC accurately summarize the DD for HFE; that the inspections, tests, and analyses identified are acceptable methods for determining whether the design commitments have been met; and that the acceptance criteria are sufficient to establish, if they are met, that the design commitments have been met.

18.10 Conclusion

The staff has reviewed the HSI design development and implementation process presented by GE in SSAR Sections 18.0 through 18.8 and Appendices 18A, 18B, 18D, 18E, 18F, 18G, and 18H, up through SSAR Amendment 34. SSAR Appendix 18C represents only one illustration of a possible CR design for the ABWR; therefore, it is NOT a part of the certified design and was NOT subject to the review process. The staff concludes that the design and implementation process discussed in the SSAR describes an acceptable HFE program, and if applied, will result in an acceptable HSI designs for the main CR and RSS. In addition, the design commitments and ITAAC in Table 3.1 of the CDM accurately summarize the minimum HFE requirements for an acceptable design, development, implementation, and V&V process for the main CR and RSS. All previously identified DSER (SECY-91-320) and DFSER issues are resolved.

19 SEVERE ACCIDENTS

19.1 Probabilistic Safety Assessment

19.1.1 Executive Summary

As part of its advanced boiling water reactor (ABWR) design certification application, GE Nuclear Energy (GE) has performed a design-specific probabilistic risk assessment (PRA) as required by 10 CFR 52.47(a)(1)(v). GE submitted a Level 3 PRA (i.e., the PRA calculated core damage frequencies, conditional containment failure probabilities, and conditional offsite consequences) that addresses internal initiating events. The PRA also evaluates seismic, internal flood, and fire-initiating events.

The staff reviewed the ABWR PRA to investigate design insights and to determine its quality. The staff concluded that the quality of the ABWR PRA is adequate for supporting and improving the ABWR design process; providing relative importance of sequences (as well as identifying important structures, systems, and components (SSCs)) leading to core damage or containment failure; and searching for design and procedure vulnerabilities that could be eliminated on a cost-benefit basis.

The draft safety evaluation report (DSER) (SECY-91-309) and the draft final safety evaluation report (DFSER) included a number of unresolved issues. Since both evaluations were too detailed and extensive to be repeated in this safety evaluation, the staff totally revised its evaluation as reflected in this report. As stated herein, those issues that were individually listed in the previous two evaluations have been adequately addressed by GE in its application.

The staff concludes, based on its review of the ABWR PRA, that the ABWR is of a robust design, that the design is an improvement over existing designs, and that the design meets the Commission's safety goals described in 51 FR 28044 and 51 FR 30028 published August 21, 1986, for internal events (see FSER Section 19.1.3.8.3). The Commission has determined that it is acceptable for GE to submit external event analyses that provide insights

needed to identify design and procedure vulnerabilities; provide insights needed for inclusion in areas such as the reliability assurance program (RAP); and inspections, tests, analyses, and acceptance criteria (ITAAC); but do not provide core damage frequency estimates suitable for use in comparison to the Commission's safety goals or in comparison to the Electric Power Research Institute's (EPRI's) Public Safety Requirement, which states

"The design is considered to have met the EPRI risk requirement if the mean complementary cumulative distribution function (CCDF) for whole-body dose developed for a 0.8 km (one-half mile) radius falls outside the region bounded by a lower limit for frequency at 1E-6 per year and has a lower limit for consequences of 25 rem whole-body dose at 0.8 km (one-half mile). The EPRI goal is based on consideration of both internal and external initiators"

Although direct comparison of external-event results to these goals is not possible, the ABWR design has significant margins above the design bases for seismic, fire, and internal flood-initiating events and, where computed, has low estimated core damage frequencies from these bounding analyses. The staff believes that the ABWR design meets the Commission's safety goals.

The staff finds that there is an acceptable balance of preventative and mitigative features in the ABWR design. The core damage frequency estimates for internal events reported in the ABWR PRA are on the order of 1E-7 per year. Table 19.1-1 lists the most important internal initiating events and Table 19.1-2 lists the top 20 internal event sequences leading to core damage. Station blackout (SBO) contributes about 70 percent of the internal events core damage frequency (i.e., its absolute value is about 1E-7 per year, which is low when compared to the figures in most recent boiling water reactor (BWR) PRA studies). Table 19.1-3 lists the most important sequences leading to core damage from seismic, internal flood, and fire event initiators.

Table 19.1-1 ABWR PRA initiating event contributors to CDF (Level 1, internal events)

Initiating Event	Events Per Yr.	CDF x 1E-8	Percent CDF
Station blackout < 2 hrs	1.2E-6	6.7	43 %
Station blackout 2 < X < 8 hrs	4.5E-7	2.6	16 %
Station blackout > 8 hrs	1.6E-8	1.7	11 %
Isolation/loss of feedwater	0.18	1.7	11 %
Unplanned manual reactor shutdown	1.0	1.2	7 %

Severe Accidents

Table 19.1-2 Important sequences leading to core damage (internal events)

Sequence Description (Top 20 sequences)	CDF (Per Year)	Percent CDF	Plant Damage Class
SBO from 0.5 to 2 hrs, RCIC unavail. because of test or maintenance (T/M)	2.4E-8	15.6	ID
SBO more than 8 hrs	1.6E-8	10.4	IB-2
SBO from 0.5 to 2 hrs, RCIC turbine mech. failure	1.3E-8	8.6	ID
SBO from 2 to 8 hrs, RCIC unavail. because of T/M	8.9E-9	5.7	IB-1
SBO from 0.5 to 2 hrs, RCIC pump fails	8.3E-9	5.3	ID
SBO from 0.5 to 2 hrs, RCIC lubrication system fails	5.1E-9	3.3	ID
SBO from 2 to 8 hrs, RCIC turbine mech. fails	4.9E-9	3.1	IB-1
Loss of feedwater/isolation, failure to inject with feedwater, conditional containment failure (CCF) of MUX, operator fails to manually initiate feedwater after 30 min.	4.6E-9	2.9	IA
Reactor trip, failure to inject feedwater, CCF of MUX, operator fails to manually initiate feedwater after 30 min.	3.1E-9	2.0	IA
SBO from 2 to 8 hrs, RCIC pump fails	3.0E-9	1.9	IB-1
Loss of feedwater/isolation, failure to inject feedwater, CCF of system logic unit, operator fails to manually initiate feedwater after 30 min.	2.3E-9	1.5	IA
Loss of feedwater/isolation, failure to inject feedwater, operator fails to manually initiate feedwater after 30 min., CCF of remote MUX	2.3E-9	1.5	IA
SBO from 0.5 to 2 hrs, valve E51-F011 fails to close after RCIC pump has started	2.2E-9	1.4	ID
SBO from 0.5 to 2 hrs, valve F037 fails closed (NCFC)	2.1E-9	1.3	ID
SBO from 0.5 to 2 hrs, valve F004 fails closed (NCFC)	2.1E-9	1.3	ID
SBO from 0.5 to 2 hrs, valve E51-F011 fails to open when RCIC pump starts	2.1E-9	1.3	ID
Loss of feedwater/isolation, failure to inject feedwater, battery CCF, loss-of-offsite line 1 power	2.0E-9	1.3	IA
SBO from 2 to 8 hrs, RCIC lubrication fails	1.9E-9	1.2	IB-1
Turbine trip, failure to inject feedwater, CCF of MUX, operator fails to manually initiate feedwater after 30 min.	1.8E-9	1.2	IA
SBO from 0.5 to 2 hrs, isolation signal logic fails	1.7E-9	1.1	ID
Totals	1.1E-7	71.9	

Table 19.1-3 Important sequences for external events (seismic, fire, and internal floods)

Seismic Events	
Sequences chosen for having low HCLPF ¹ values or needing few SSC failures	Sequence HCLPF
Failure of emergency dc power	0.74 g
Emergency ac/emergency SW and fire water	0.62 g
Emergency ac/emergency SW and scram	0.62 g
Reactor or control building, containment, RPV pedestal, or RPV supports	1.11 g
ATWS and SLCs failure	0.62 g
ATWS and SLCs and high-pressure core flood	0.62 g
Internal Flood Events	
Sequences chosen as being most challenging and having worst consequences	Estimated CDF
Control building: large pipe break in RSW, operator fails to isolate flooding, auto RSW pump trip fails, water flows to remaining RSW pump rooms, operator fails to respond to flooding alarm, RSW fails.	2E-9 per year
Reactor building: break in fire water standpipe or line from CST, operator does not respond to alarm to isolate flood, overflow lines to corridor are clogged, all three electrical rooms on floor B1F flood, ac power is lost to all make-up systems.	2E-10 per year
Turbine building: break in CWS system, isolation valves in CWS lines fail to close, water fills up and runs out of the condenser pit, fire door between the turbine building and the service building is either open or fails open allowing water into service building, service building floods and a door between the service building and the control building fails open or is open, water entering the control building causes electrical power supplies and all three divisions of RCW to fail.	3E-9 per year
Fire	
Sequences chosen that had a core damage frequency of 1E-6 or higher before the plant design was improved	Estimated CDF (considered conservative)
Fire in the control room causes its evacuation, feedwater fails, RCIC or one high-pressure core flood train fails, either one train of low-pressure core flood train or manual depressurization fails.	less than 1E-6 per year

- 1 - High confidence with low probability of failure that the structure, system, or component will fail at the given peak ground acceleration.

Note: Because of the assumptions and methods used in the ABWR shutdown risk evaluation, no dominant sequences leading to core damage could be determined. NUREG-1449 did identify various important scenarios that were potential contributors to core damage in BWRs during shutdown. Conclusions from NUREG-1449 are based on actual nuclear power plant operating experience.

Severe Accidents

The internal events core damage frequency estimate is very low and is a reflection of GE's efforts to systematically minimize the effect of sequences or initiators that have been important contributors to core damage frequency in previous BWR PRAs. A brief discussion is necessary on the implications of the low estimated core damage frequency for internal events for the ABWR. Estimated core damage frequency values with absolute values less than 1 in a million years should not be taken literally as the expectation of the "true" core damage frequency of the design. Rather, this value should be taken as a reflection of the conscientious engineering and design effort to reduce or eliminate the contributors to core damage frequency found in previous PRAs. When core damage frequencies of one in a hundred thousand or a million years are estimated in a PRA, it is the areas of the PRA where modeling is least complete, supporting data are sparse, or even nonexistent that actually could be the more important contributors to risk. Areas not modeled or incompletely modeled include errors of commission, sabotage, rare initiating events, construction errors, design errors, control systems, ageing, systems interactions, human interaction with smart control rooms, and human errors.

For seismic initiating events, GE submitted a PRA-based seismic margins analysis. This method eliminates the uncertainties associated with picking an appropriate seismic hazard curve, while still providing the insights needed to judge the ability of the design to withstand beyond-design-basis earthquakes. With a PRA-based seismic margins analysis, rather than developing an estimated core damage frequency, the method estimates the margin the design has beyond the design basis safe shutdown earthquake (SSE) (which is 0.3g for the ABWR) and identifies any weak links in the design. GE reported that all sequences leading to core damage from a purely seismic event were found to have a high confidence with low probability of failure (HCLPF) value of 0.6g or higher. An HCLPF roughly represents the g-level acceleration at which a SSC is believed to fail 5 percent or less of the time with a 95 percent confidence level. The staff finds that the ABWR's HCLPF ($\geq 0.5g$) demonstrates that a significant margin exists beyond the design basis earthquake level.

For internal floods that occur at power, GE performed a PRA internal flood analysis that assumed that once flood water reaches a level high enough to cause the failure of any piece of equipment in an area, then all the equipment in that area instantly fails and is unrecoverable. This analysis estimated that the core damage frequency from internal floods was on the order of $1E-8$ per year. This number was particularly low because the ABWR design has three safety divisions that are physically separated. For internal floods during shutdown, GE developed

guidelines for the COL applicant for configuring shutdown cooling divisions such that one division would be in operation, one isolated and in standby, and one in maintenance. It is believed that the core damage frequency from this configuration should not exceed $1E-6$ per year and probably can be at least an order of magnitude lower, given the conservatism of the assumptions in the analysis.

For fires, GE submitted a fire analysis that was a combination of the Fire Induced Vulnerability Evaluation (FIVE) methodology developed by EPRI and the internal events PRA. This analysis assumed that, if a fire occurred in any portion of a fire area, all equipment in the area failed instantly. The GE fire analysis estimated the core damage frequency from fires to be on order of $1E-6$ per year.

The design basis analysis for tornados in the ABWR is such that the ABWR is designed to be able to withstand tornados that occur with a frequency in the range of $1E-7$ per year. Since the plant is designed to handle these low frequency tornados and is already analyzed for loss-of-offsite power events, the staff does not consider it necessary to analyze design basis tornados probabilistically.

The staff finds that the ABWR design is adequate to limit exposure to risk when the plant is operated in Modes 3, 4, and 5 (hot shutdown, cold shutdown, and refueling, respectively). The staff finds that the ABWR design includes enhanced features that reduce risk during shutdown operations when compared to operating BWRs. These features specifically address the more risk-significant operations during shutdown identified in NUREG-1449 including three independent residual heat removal (RHR) divisions, three emergency diesel generators (EDGs), an ac-independent water addition (ACIWA) system, an alternate onsite combustion turbine generator (CTG), and proper plant electrical and physical separation and layout.

The results of the Levels 2 and 3 portion of the ABWR PRA indicate that the ABWR containment is quite robust and able to accommodate severe accidents with a low attendant probability of containment failure. Both GE and staff estimates of the ABWR conditional containment failure probability (CCFP) are within the Commission's containment performance goal (0.10). Using the structural integrity definition, GE's estimate of CCFP is 0.005, whereas the staff's estimate is 0.026.

Based on the Level 3 PRA, the estimated total risk to the public for the ABWR is extremely small. GE's analysis indicates a total dose of about 0.3 person-rem over the

60-year plant life. The staff's estimate is about 1 person-rem. The staff estimated that total risk is dominated by events that lead to early containment failure and containment bypass. This is consistent with results from PRAs for operating plants.

GE made a number of design modifications to the ABWR both early in the design and later during the staff's review of the ABWR PRA that were motivated by the results of the PRA. Table 19.1-4 describes some of the modifications made to the design by GE.

Table 19.1-4 Examples of cost-effective PSA-inspired design/procedure modifications to the ABWR design

Area Modified	Modification
Core cooling systems	GE found it only needed three, not four, ECCS divisions.
Reactivity control	GE found alternate rod insertion reliability was such that less expensive ATWS mitigation system was acceptable.
Instrumentation	GE found that it could eliminate 60 percent of sensor instrumentation in the reactor safety systems without affecting plant safety.
AC-independent water addition (ACTWA) system	System added to ABWR design. Staff believes that it is the most important system for helping to prevent severe accidents.
Combustion turbine	System added to ABWR design. In combination with ACTWA, it virtually eliminates SBO as a consideration.
Lower drywell flooders (LDF)	System added to ABWR design. Floods lower drywell in event that vessel fails and corium enters lower drywell.
Containment Overpressure Protection System (COPS)	System added to ABWR design. If an accident occurs that pressurizes containment, the COPS allows for release of the pressure (90 psig set point) with the capability to reclose vent path.
Control for fourth SRV on remote shutdown panel	Extra SRV control added based on ABWR fire analysis. Mitigates control room fire.
Water-level sensors, pump trip and valve isolation circuits	Pump trips added for floods in turbine and control building as well as valve isolation signals on high water level. Pipe length between first RSW isolation valve and control building limited to help assure pipe break will not cause unacceptable results.
Containment concrete	Basaltic concrete used rather than limestone concrete to limit production of noncondensable gases from a core on floor event.

Severe Accidents

GE and the staff have drawn a substantial number of significant safety insights from the ABWR PRA that have or will affect the design, construction, operation, maintenance, and regulation of the ABWR. These insights are discussed in more detail in Section 19.1.3 in the final safety evaluation report (FSER) and in Section 19.8 in the SSAR. Appendix 19K in the SSAR lists those SSCs that are to be included in a COL action item that lists proposed inputs for the COL applicant's operational reliability assurance process (O-RAP) and design reliability assurance program (DRAP) based on the ABWR PRA. Appendix K in the FSER lists those safety insights that were motivated by the PRA. The disposition of these insights is documented in the appendix and indicates if the insights are in ITAAC, Tier 2 information, technical specifications (TS), COL action items, Interface Items, or RAP.

It is the staff's view that the mean core damage frequency for the ABWR from internal, external, and shutdown events is probably on the order of $1\text{E-}6$ or less assuming the plant is constructed, maintained, and operated in accordance with the SSAR. This judgment is based on the staff's understanding of operating plant experience and PRA results, design improvements in the ABWR relative to operating reactors, and insights from the staff's review of the ABWR PRA. Furthermore, it should be emphasized that there are large uncertainties in internal event core damage frequency estimates. The external event analyses for the ABWR were quasi-probabilistic and were designed to uncover vulnerabilities in the design rather than generate specific core damage frequency estimates. Also, the shutdown risk evaluation was quasi-probabilistic, using PRA-based techniques to determine the reliability of shutdown cooling and to examine the possibility of determining plant configurations that would limit exposure to risk. The staff's review of the shutdown risk evaluation reflected the insight from previous PRAs that human error was the greatest contributor to shutdown risk. However, human error analysis methods still cannot accurately predict human response to various circumstances. This fact increases the uncertainty of the estimated bottom-line core damage frequency numbers for shutdown events.

GE conducted uncertainty analyses for the Level 1 portion of the ABWR PRA. GE reported that the internal event core damage frequency distribution had a mean value of $1.6\text{E-}7$ per year and an error factor (EF) of about 4.2 (where the EF is the ratio of the 95th percentile to the median of the lognormal distribution). The 95th and 5th percentiles reported by GE were $4.5\text{E-}7$ per year and $3.8\text{E-}8$ per year, respectively. GE used a lognormal distribution for most random variables in the ABWR PRA, which is a mathematical simplification and assumption that is commonly used in PRA evaluations. The actual distribution for most variables is not known. Use of the

lognormal distribution in lieu of the "actual" distribution adds an unquantifiable uncertainty to the evaluation, particularly in the bottom-line numbers.

GE performed importance analyses in order to determine the most important structures, systems, and components (SSCs) to be added to the RAP. GE used two importance measures: risk achievement worth ratio and fussell-vesely importance. The analysis identified scram function and its attendant equipment as very important. Station batteries are similarly important, since dc power controls the automatic depressurization system (ADS) as well as many pumps and valves. Another insight is the importance of reactor core isolation cooling (RCIC). This is because it is ac-independent, reliable, and provides high-pressure injection. These attributes are important for mitigating SBOs.

The probabilistic shutdown risk evaluation performed by GE concluded that the ABWR design could be maintained in configurations during Modes 3, 4, and 5 so that the estimated conditional core damage frequency was very low. The staff noted that estimates of core damage frequency and risk at shutdown for the ABWR design have much larger uncertainties than do such estimates for internal events in Modes 1 and 2. In the SSAR, GE presented sample plant configurations that help limit risk when shutdown occurs. The staff finds that the ABWR design, through appropriate shutdown planning, contingency planning, and operator instrumentation, can be configured and maintained in Modes 3, 4, and 5 in a manner that helps to reduce the chances of core damage or releases to the environment to a point where shutdown risk does not represent a disproportionate risk to the public.

19.1.2 Introduction

As part of the ABWR design certification application, GE submitted the ABWR PRA in response to 10 CFR 52.47, the Commission's Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants, described in ER Vol. 50, No. 153, dated August 8, 1988, p. 32138 dated August 8, 1991, and the ABWR Licensing Review Bases. The staff's assessment included the traditional evaluation of events that could lead to core damage and offsite consequences as well as an evaluation of what the ABWR PRA revealed about the ABWR design.

The general objectives of the staff's review of the ABWR PRA were (a) to identify safety insights based on the performance of systematic risk-based evaluations of the ABWR design; (b) to determine in a quantitative manner whether the ABWR design represents a reduction in risk over existing plants; (c) to examine the balance of preventive and mitigative features of the design; (d) to

assess the reasonableness of the risk estimates documented in the PRA and other risk-related documents submitted as part of the FDA application package, and (e) to support pre- and postcertification activities such as ITAAC, RAP, TS, and completion of site-specific design details (e.g., ultimate heat sink). In addition, the ABWR PRA was used to both determine how the ABWR design related to various safety goals and discover design and procedural vulnerabilities.

The objectives are drawn from 10 CFR 52.47, the Commission's Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants, the Commission's Safety Goal Policy Statement, the Commission approved positions concerning the analyses of external events contained in SECY-93-087, and the Commission's interest in the use of PRA to help improve future reactor designs. In general, these objectives have been achieved by the ABWR PRA and the staff's review. The staff's proposed applicable regulation for the analysis of external events for the ABWR PRA is as follows:

The application for design certification must contain a probabilistic risk assessment that includes an assessment of internal and external events. Simplified probabilistic methods and margins methods may be used to assess the capacity of the standard design to withstand the effects of external events such as fires and earthquakes. Seismic margin analysis must consider the effects of earthquakes with accelerations approximately one and two-thirds the acceleration of the safe-shutdown earthquake.

The staff believes that if its review were to concentrate on bottom-line numbers or merely on the quality of a PRA, the most important insights from a PRA could receive inadequate attention. In Section 19.1.3, the staff reported on its investigation of those safety insights that are revealed by the ABWR PRA about the ABWR design. These insights include those that are to be passed on to the COL applicant, insights into the balance of prevention and mitigation, design vulnerabilities, and aspects of the design that tend to reduce or exacerbate risk estimates. The results of this broadened perspective are multifold. GE's systematic evaluation of the ABWR design has accomplished the following:

- Identified important areas where minor design modifications will help confirm that the potential for severe accidents is maintained at a low level
- Helped GE to identify the most safety-important equipment to be included in the RAP

- Provided the COL applicant with information about what is most important about the ABWR design and operation from a safety standpoint
- Helped confirm for the staff that the ABWR design is robust against internal and external events.

GE provided a list of important safety insights from the ABWR PRA including internal and external events and events during all modes of operation. This list, which was developed from a systematic process, is described in detail in SSAR Section 19.9 and is detailed in Appendix K in the FSER. As indicated in this table, GE recommended and staff agreed that a subset of these insights be included in the design control document (DCD) as ITAAC, Interface Items, or Tier 1 or Tier 2 material.

During the construction stage, the COL applicant will be able to consider as-built information. The staff concludes that updated PRA insights, if properly evaluated and utilized, could strengthen programs and activities in areas such as training, development of emergency operating procedures, reliability assurance, maintenance, and 10 CFR 50.59 evaluations. The staff recommends that the design-specific PRA developed to meet 10 CFR 52.47 be revised to account for site-specific information, as-built (plant-specific) information refinements in the level of design detail, and design changes. These updates are the responsibility of the COL applicant. As plant experience data accumulate, failure rates (taken from generic data bases) and human errors assumed in the design PRA are to be updated and incorporated, as appropriate, into ORAPs.

19.1.3 Advanced Boiling Water Reactor Probabilistic Risk Assessment-Based Safety Insights

Insights gleaned from a PRA can provide significant perception into the design and operation of a nuclear power plant. This section documents the insights derived by GE and the staff about the ABWR design based on the ABWR PRA.

19.1.3.1 Technical Insights Summary

In developing the ABWR design, GE significantly reduced the dominant contributors to core damage frequency found in most current BWR plant-specific design PRAs. The success of this attempt is substantiated by the low estimated core damage frequency for internal events recorded in the ABWR PRA (1.6E-7 per year) and the staff's conclusion documented in Section 19.1.4 that, on balance, the ABWR PRA was performed in an acceptable manner. The estimated core damage frequency from internal floods is about 7E-9 per year. The fire analysis performed by GE for the ABWR produced a core damage

Severe Accidents

frequency estimate of about $1\text{E-}6$ per year. Because of conservatism in the analyses, the staff does not believe that the fire and internal flood core damage frequency estimates should be compared to those of internal events. The seismic analysis performed by GE was a PRA-based margins analysis that does not provide core damage frequency estimates. The margins analysis demonstrated that the ABWR design is robust for seismic events well beyond the design basis. The staff finds that the ABWR design, if built, maintained, and operated as assumed in the ABWR PRA, represents a reduction in risk when compared to the current design of BWRs.

It is important to understand what contributes to core damage frequency and risk in the ABWR design. Internal initiating events that are dominant contributors to core damage frequency include SBO, loss of offsite power, and vessel isolation or loss of feedwater. Failure of RCIC, the multiplexing transmission network, the station batteries, and the trip logic units are the most important contributing failures. SSAR Section 19K discusses the SSCs found to be most important in the PRA for internal and external events and when the plant is in modes other than full power. Table 19.1-2 provides a list of important sequences leading to core damage or risk for internal events. Table 19.1-3 provides a list for external events sequences.

For events occurring during modes other than full power, the ABWR design provides enhanced protection over the designs of many operating plants in that it has three fully separated safety divisions. To make use of this redundancy for maintenance purposes and still maintain an appropriate level of protection when in modes other than full power, GE has developed tables (See SSAR Tables 19Q.7-2 to 19Q.7-4) that list combinations of equipment that, if kept operable while in Modes 3, 4, and 5, will help ensure that a severe accident does not occur. These tables have the goal of maintaining a conditional core damage frequency of less than $1\text{E-}5$ per year should the operating train of RHR cooling become unavailable. Since GE assumed that a loss of RHR cooling (i.e., of the operating train) has an occurrence rate of once per 10 years (GE's assumption is that during a 10-year period with all its included shutdowns there will be one loss of shutdown cooling event — 0.1 per year), this gives a frequency of core damage of less than $1\text{E-}6$ per year when operating in Modes 3, 4, and 5. In the shutdown analysis, it is conservatively assumed that all equipment not specifically referred to in the tables as "operable" is "inoperable." During shutdown, it is conservatively assumed that any core damage would result in a large release, since the containment would be open much of the time. The staff and GE noted that separation of safety divisions is not always maintained during maintenance

outages. To help assure that fires and floods cannot become common cause failures of all three divisions when in modes other than full power, GE has developed guidelines for plant operation. These guidelines would have one division isolated and in standby, another division operating in the shutdown cooling mode but not necessarily isolated, and another in maintenance. The staff believes that a safety-oriented approach to planning and controlling an outage is needed and that such an approach will reduce risk during Modes 3, 4, and 5. It recommends that COL applicants make use of the guidance provided by GE in Section 19Q of the SSAR regarding outage planning and control.

For seismic events, fires, and internal floods, the ABWR design has specific advantages over many current designs. The seismic design bases for the ABWR is a 0.3g SSE. In simple terms, the ABWR design can be built at any site that has its site-specific spectrum bounded by the design bases spectrum. Such a site might normally have an SSE of 0.2g assigned to it. However, the ABWR will be built to the 0.3g SSE standard, regardless. This creates an additional explicit seismic robustness at most potential sites east of the Rocky Mountains. For fires and internal floods, the existence of three separated safety divisions along with a diesel-driven fire water pump and an alternative water supply (that will remain functional following a design bases earthquake) external to the reactor building provide design improvements that significantly reduce potential core damage.

External events such as external flooding and hurricanes may be analyzed by a designer in a bounding manner in order to minimize the chances that the design has vulnerabilities to site-specific external events. GE did not chose to evaluate such external events at the Design Certification stage. The staff worked with GE to assure that GE was aware of the potential for vulnerabilities to such external events. It is possible that some sites may not be appropriate for the ABWR design because they could introduce vulnerabilities into the ABWR design that were not taken into account in the Design Certification process. For each site, the COL applicant must provide a site-specific PRA-based analysis to help determine if the ABWR design has any vulnerabilities to previously unanalyzed external events applicable to the site (e.g., river flooding or soil liquefaction from seismic events).

Human reliability analyses (HRAs) of the ABWR design show that it is not particularly sensitive to operator errors during operation in Modes 1 and 2. This is true, in part, because of the multitude of paths by which water can be provided to the core and the inherent attributes of the design (in most sequences of actions, it takes a long time before an operator needs to make a critical decision, if

such a decision is needed at all). GE conducted a sensitivity analysis to investigate the influence of variations in the PRA-modeled human error rates of the ABWR core damage frequency estimate. The analysis suggests that, while there is little room for reducing core damage frequency by improving human performance, core damage frequency can increase if human performance significantly degrades. The potential increase in core damage frequency appears to be about two orders of magnitude above the base case frequency under an assumption that all human error probabilities are simultaneously increased by a factor of 30. However, such a magnitude of increase is unrealistic because the human error probabilities used in the ABWR human performance analyses were either screening values or reasonable values for the actions defined based on previous HRAs for existing PRAs. Even though some recovery actions (such as offsite power and emergency ac) were not included in this analysis, the results of the sensitivity analysis provide a proper indication of the ABWR tolerance of human errors.

Since the ABWR design was not detailed in a number of areas important to evaluating human actions and potential errors (e.g., control room design or plant-specific emergency procedures), GE's HRA was essentially a scoping analysis, based largely on a generalization of results from previous HRAs (which reflect conventional BWR human-machine interface designs) and the use of screening-type human error probabilities collected from various sources. GE contends that this treatment is conservative for the ABWR because of the significant improvements envisioned for the ABWR human-machine interface design relative to earlier designs. However, the validity of the scoping analysis for the as-built ABWR design will need to be confirmed as part of the implementation of the detailed control room design process. This process and associated ITAAC/DAC are described in Chapter 18 of this report. The focus of that effort with regard to the HRA will be on confirming that the final control room design has not introduced any human engineering deficiencies that would significantly increase the error rates for human actions modelled in the HRA or the potential for additional, risk-significant errors not modelled in the HRA.

Every nuclear power plant PRA is incomplete to some extent. Ordinarily, a PRA is performed on a plant for which the site is known, the equipment has been procured, and the plant is nearly or completely built. For the ABWR PRA, the PRA practitioners had to develop their models using a design that lacks many of the details available for an existing plant. The lack of detail was recognized by the staff as well as by GE. For this reason, the staff developed several processes to which the COL applicant must comply, including (a) the ITAAC process, (b) the

reliability assurance process and program (O-RAP and DRAP) that the COL applicant should implement to confirm that the as-built plant conforms to the assumptions of the ABWR PRA, and (c) interface items that the COL applicant must address in its application for a COL. To help confirm that the assumptions in the PRA are realized in the as-built design, GE provided a systematic list of SSCs that are to be included in the COL applicant's RAP, PRA-based insights into the ITAAC, and a systematic list of safety insights derived from the ABWR PRA that will be passed on to future COL applicants. One of the significant benefits of having a PRA at early stages in the design is that GE took advantage of PRA insights to improve the final design and provide guidance to COL applicants or holders.

Based on the information provided in the SSAR, this design has achieved a significant reduction in expected core damage frequency and risk compared to operating plant PRA results. The staff compared the numerical results of internal events in the ABWR PRA to the Commission's safety goals and found that the ABWR design meets each safety goal.

As part of its investigation of the ABWR design, the staff endeavored to determine if a balance had been achieved between the prevention of accidents and accident mitigation capabilities. The staff concludes that the design has an appropriate balance of prevention and mitigation. Details of this discussion are provided in Section 19.1.3.9.2 of this report.

GE searched for design and procedure improvements that were prudent to include in the ABWR. The staff concludes that the search made by GE was adequate to identify design and procedure vulnerabilities for the ABWR design. Details of the design improvements motivated by the ABWR PRA are discussed in Section 19.1.3.2.3 of this report.

The staff believes that the ABWR PRA is capable of supporting pre- and post-certification activities such as ITAAC, RAP, and TS. The PRA can be modified to include site-specific design details for areas outside the design certification such as the ultimate heat sink.

19.1.3.2 Level 1 — Internal Events

19.1.3.2.1 Dominant Accident Sequences

GE estimated the total core damage frequency from internal events for the ABWR design to be $1.6E-7$ per year. The internal events initiators that contribute the most to core damage frequency are loss of offsite power, SBO, and loss of feedwater/isolation of the vessel (See

Severe Accidents

Table 19.1-1). Of these, SBO is the largest contributor to core damage frequency. This is consistent with risk profile estimates for many other BWR PRAs that have identified SBO as one of, if not the leading, contributor to core damage frequency. For the ABWR, SBO and loss of offsite power sequences have an estimated core damage frequency of about $1\text{E-}7$ per year.

The top seven sequences that in aggregate contribute 52 percent of the internal event core damage frequency are SBO events in which RCIC fails or is unavailable or where the blackout outlasts the capacity of the emergency batteries. Of the top 20 sequences, 14 involve SBO. The reason SBO events in the ABWR have a very low estimated absolute core damage frequency value is the low overall core damage frequency estimate for the ABWR design. This is primarily because the ABWR design has three independent EDGs and because of the addition of the onsite CTG. With respect to SBO events, the staff noted that the removal of a steam-driven high-pressure system (such as the high-pressure core injection system (HPCI) in earlier designs) is well compensated for by a substantial improvement in the reliability of backup power resulting from the addition of a CTG in the ABWR design.

Anticipated transients without scram (ATWS) has a very low absolute value for the ABWR design for several reasons, including the following: (1) the ABWR design has diverse means of inserting control rods into the core (by hydraulic or electric means), (2) initiation of the standby liquid control system (SLCS) is automated, (3) the hydraulic system that inserts control rods includes additional backup scram valves to relieve scram air header pressure, and (4) the ABWR design does not have a scram discharge volume.

19.1.3.2.2 PRA as a Design Tool

The ABWR design was influenced by PRA insights. Table 19.1-4 summarizes design changes that GE made to the ABWR that were motivated by PRA insights. GE used PRA insights and mini-PRA studies to help decide on a number of important design options. The performance of the ABWR PRA and several mini-PRAs early in the ABWR design process made important contributions to improving the ABWR design. The PRA influenced the design positively, not only early in the design process, but also during the design certification.

GE indicated that the use of PRA evaluations in the early stages of the ABWR design helped GE to simplify the design in a manner that maintains or improves core damage frequency estimates compared to estimates for operating plants. Examples of the design simplification by GE include the following:

- The reduction in the number of emergency core cooling system (ECCS) divisions from four to three by upgrading the RCIC system and by modifying the automatic depressurization system (ADS) logic to begin operation when a low water level is reached (improved design for transient response)
- The elimination of 60 percent of the sensor instrumentation in the reactor safety systems without its affecting plant safety.

In its initial ABWR PRA submittal, GE concluded that the Commission's safety goals and goals proposed by EPRI for evolutionary reactors (core damage frequency, CCFP, and dose at the boundary of the plant following an accident) were met and that the design had many means of severe accident prevention and mitigation. After discussions with the staff and in order to come to more complete agreement with EPRI evolutionary plant design guidelines, GE decided to make several design improvements that significantly added to defense in depth and reduced uncertainty that the design would meet its intended goals. GE made the following design changes:

- Added an AC-independent water addition (ACIWA) system. This system provided benefits for SBO, fires, internal floods, and seismic events. It has both preventative and mitigative capabilities in that it can either inject to the vessel or spray the drywell. It can provide water from a seismically robust diesel-driven pump or a fire truck.
- Added a non-safety grade combustion turbine generator that starts automatically with safety grade loads added manually. This system used in conjunction with the ACIWA system significantly reduces the estimated core damage frequency from SBO.
- Made recommendations for the improvement of emergency procedures following examination of the dominant severe accident sequences.
- Changed the lower drywell basemat concrete composition from limestone to basaltic concrete to limit production of noncondensable gases.
- Added a containment overpressure protection system (COPS).
- Increased the pressure capacity of the drywell head.
- Surveillance testing of the microprocessor-based controllers was increased to quarterly to improve the ability to detect failures left undetected by the continuous self-test feature. This action was taken

based on a study of potential failures of the safety system logic and control.

19.1.3.2.3 Plant Features and Operator Actions Important to Risk

Insights as to what features, procedures, or operator actions are important to risk have been gathered from several areas of Chapter 19 of the SSAR, including Appendices 19K, 19L, 19M, 19Q, and 19R as well as from discussions with GE. The staff considers the following plant features, procedures, and operator actions to be the most important to risk reduction or prevention for Level 1 internal events.

Plant Features Important to Risk

Combustion Gas Turbine Generator — The combustion gas turbine generator (CTG) provides non-safety grade power that is diverse and independent from the normal and emergency ac power sources. It is capable of providing ac power to any of the three safety divisions or to a condensate pump. The CTG requires no plant support systems to start or run. The gas turbine, in conjunction with the ACIWA system, is a significant factor in reducing the likelihood of an SBO (a leading contributor to core damage).

ACIWA System — The ACIWA system has two pumping sources available: a diesel-driven pump and a fire truck. The diesel-driven pump is designed to survive a design bases earthquake. Its pumping sources are located outside the reactor building. Manual valves to direct the flow from the ACIWA system into the RHR system and then on to either the core or to the drywell or wetwell spray are located in the reactor building and can be operated successfully following an internal event.

Lack of Recirculation Piping — The ABWR design has done away with external reactor recirculation piping and pumps. This means that there are no large pipes that penetrate the vessel below the core. This design detail has significantly reduced the already low chances of a loss-of-coolant accident (LOCA) leading to core damage.

Three ECCS Trains — The redundancy of the divisions improves the chances of the design preventing transients from developing into core damage events.

Reactor Core Isolation Cooling (RCIC) — RCIC is a safety-grade system that provides the ABWR with a diverse high-pressure system that can delay the onset of core damage following an SBO event to about 8 hours. Importance analyses from the ABWR PRA indicate that

RCIC is one of the most important systems in preventing core damage accidents.

Multiplexing System — The high reliability of the multiplexing system is very important in the ABWR PRA. Significant degradation of this function could result in a large increase in the likelihood of the plant's having an event that leads to core damage.

High-pressure Core Flooder (HPCF) Pump — One of the high-pressure core flood pumps can be operated independently of the essential multiplexing system. This design feature is an important factor in reducing the chances of the plant experiencing core damage, since this design should reduce the chance of a common cause control system failure, that would disable all ECCS pumps.

Electrically Driven Control Rod Insertion — The diverse ability to drive in rods with electrical motive power as well as hydraulic power significantly reduces the chances of an ATWS occurring.

Automatic Depressurization System — The ADS is needed to enable the use of low-pressure core flooder pumps and the ACIWA system. ADS is important in SBO, small LOCAs, and transients.

Reactor Water Cleanup (RWCU) Isolation Valves — The isolation valves in the RWCU system must be capable of isolating against a differential pressure equal to the operating pressure of the reactor coolant system (RCS) in the event that there is a LOCA in the RWCU.

Operator Actions Important to Risk

- Five specific human actions have been identified as most important to the Level 1 PRA analysis because of their impact on core damage frequency. All five of these relate to makeup of reactor inventory — the first four with the reactor at high pressure and the last with the reactor at low pressure (the acronyms in parentheses correspond to the basic event identifier in the ABWR PRA):

- Backup manual initiation of HPCF (HOOBOPHL)
- Recovery of feedwater in events without MSIV closure (Q)
- Recovery of feedwater in events with MSIV closure (Q2)
- Reopening of HPCF injection valves following maintenance (HBMAER1)

Severe Accidents

- Use of condensate injection following scram with reactor depressurized (COND).

A sixth human action — control of reactor water level in an ATWS (LPL) — was also identified as critical to the Level 1 analysis based on its impact on offsite consequences rather than on its impact on core damage frequency. Because of their importance, these human actions are included as "critical tasks" that will be evaluated further by the COL applicant as part of the advanced plant design (see SSAR Section 19.9).

- Although miscalibration of sensors is not a significant contributor to core damage frequency (partly because of the low assigned probabilities), instrument calibration is an important maintenance action requiring that a special maintenance procedure be developed by the COL applicant.
- A limited number of human actions were identified as having a significant impact on containment performance and the results of the Level 2 analysis. These involve the following:
 - Emergency depressurization of the reactor to mitigate core damage in-vessel and reduce the potential for containment failure from direct containment heating (OP)
 - Alignment and initiation of firewater (ACIWA mode of RHR) for injection into the depressurized reactor pressure vessel (ARV)
 - Alignment and initiation of firewater for injection to the drywell sprays to prevent drywell overtemperature (HTF)
 - Initiation of drywell sprays in response to suppression pool bypass
 - Use of a fire truck as a backup to the diesel-driven firewater pump.

Because of their importance, the actions are included in COL Action Items (See SSAR Section 19.9). Some of these actions would also be addressed by the COL applicant as part of its Accident Management Program discussed in Section 19.2.

Appendix 19K in the FSER lists those PRA-based safety insights motivated by the PRA. The disposition of these insights is documented in the table and indicates if the insights are in ITAAC, Tier 2 information, TS, COL action items, Interface Items, or the RAP.

19.1.3.2.4 Interface and COL Applicant Action Items

The Design Certification Rule 10 CFR Part 52 does not require a vendor to provide complete design details, especially in areas where there is evolving technology (e.g., the control room) or where it is very difficult to bound the possible design options (e.g., the ultimate heat sink and the service water pump house). Other areas such as accident management are the responsibility of the COL applicant. GE identified Interface Items (hardware) and COL Action Items (primarily procedural or analytical) that define the areas derived from performance of the ABWR PRA that the COL applicant needs to perform or complete. The following COL action item addressed in Section 19.9 in the SSAR is of particular interest to the staff:

- The COL applicant should update the design-specific ABWR PRA to include site-specific information (i.e., a site-specific PKA) and additional design details (e.g., once the COL applicant designs the structures, systems, and components that were not part of the design certification). The PRA is to be maintained by the COL holder (living PRA) so that the PRA can be useful in helping to make 50.59-like decisions as well as helping to determine the safety significance of operational events or data from ABWR operation or other nuclear power plants.

These actions are detailed in Section 19.9 and Table 1.9.1 in the SSAR. The staff finds the Interface Items in Section 1.9 in the SSAR to be adequate for performing the ABWR design-specific PRA. The staff finds the COL action items in Section 19.9 of the SSAR to be acceptable and finds that GE identified those actions that the COL applicant must take in order to help make the PRA assumptions come true in the as-built, as-operated plant.

19.1.3.2.5 Insights From Uncertainty, Importance, and Sensitivity Analyses

GE has conducted a traditional uncertainty analyses for the Level 1 portion of the ABWR PRA. GE calculated importance analyses in order to determine the most important SSCs to be added to the RAP. GE also conducted selected sensitivity analyses to determine the robustness of the design to biases in numerical assumptions.

The uncertainty analyses started with GE's assuming that all basic events (in the fault trees) have lognormal distributions. These distributions were propagated using Monte Carlo techniques. As shown in Figure 19.1-1, the core damage frequency distribution had a mean value of $1.6E-7$ per year and an error factor (EF) of about 4.2

AIM

Association for Information and Image Management

1100 Wayne Avenue, Suite 1100
Silver Spring, Maryland 20910
301/587-8202

Centimeter

Inches

MANUFACTURED TO AIM STANDARDS
BY APPLIED IMAGE, INC.

8 of 10

(where the EF is the ratio of the 95th percentile to the median of the lognormal distribution). The 95th and 5th percentiles reported by GE were $4.5\text{E-}7$ per year and $3.8\text{E-}8$ per year, respectively.

The uncertainty studies revealed that the majority of basic events that were part of the dominant core damage sequences (i.e., involved in the failure of SSCs needed to prevent core damage) were also identified by the importance analyses as the top contributors to core damage frequency. Similarly, the relative importance of the top-ranked sequences and top-ranked basic events was shown to change only slightly when the input data were biased by a factor of two.

GE performed importance analyses on internal events using "change in core damage" as the underlying variable. GE used the importance analyses results to help determine SSCs to be included in Appendix 19K of the SSAR that deals with SSCs that are to be included in the COL

applicant's RAPs. GE used two importance measures: risk achievement worth ratio and Fussell-vesely importance. The risk achievement worth ratio is calculated by taking the ratio of (a) the core damage frequency with the particular SSC always failed to (b) the base core damage frequency. Risk achievement identifies those SSCs for which it is particularly important to do good maintenance, since poor reliability or availability of this equipment would increase estimated core damage frequency significantly. The Fussell-Vesely importance measure is calculated by (a) finding the difference between the base case core damage frequency and the core damage frequency with the SSCs operating perfectly, and (b) dividing by the base case core damage frequency. The Fussell-Vesely importance measure identifies which SSCs would benefit the most from improved testing and maintenance that would minimize equipment unavailability and failures. The staff believes that the risk achievement worth is a particularly important measure when deciding which SSCs to include in a RAP.

Figure 19.1-1 ABWR core damage frequency distribution

Severe Accidents

Table 19.1-5 in this report provides a listing of the most important SSCs and their importance. The table highlights the importance of the scram function and its attendant equipment. Similarly important are the station batteries, since dc power controls ADS as well as many pumps and valves. The table also points out the importance of the CTG and RCIC. This is because SBO is the largest contributor to internal event core damage frequency estimates.

Table 19.1-6 in this report shows the sensitivity of core damage frequency to test and maintenance outage assumptions for ECCS equipment. Increasing RCIC unavailability was found to cause the greatest increase in estimated core damage frequency. This results in large part from RCIC's contribution to mitigation of SBO sequences. This outcome supports the results of importance analyses discussed above. GE found that increasing high-pressure core flooders B (HPCFB) unavailability resulted in the second highest core damage frequency increase. This is because HPCFB is important in mitigating a common-cause failure of the essential multiplexing system.

GE performed a sensitivity study on the effect of increasing the test and maintenance outage times for ECCS equipment (i.e., the period each piece of ECCS equipment is assumed to be unavailable during the year because of testing or maintenance). RCIC was found to be the ECCS system most sensitive to increased outage time. This would be expected based on the Fussell-Vesely importance measure results. Second in importance is high-pressure core flooders "B" that includes a diverse (resulting from the multiplexing system), hard-wired manual-initiation backup in the control room. The sensitivity study indicated that other individual systems are not sensitive to test and maintenance outage time. GE noted that even with RCIC out of service all the time, the ABWR core damage frequency estimate is still below $1\text{E-}5$ per year. It should be noted that the CTG and the EDGs have the largest Fussell-Vesely importance measure value, since 100-percent availability of the CTG or an EDG would eliminate SBO as an estimated core damage contributor.

19.1.3.3 Level 1 — External Events

Three external events were analyzed in the ABWR PRA: seismic, fire, and internal flood. In many PRAs performed to date, these external events have had combined core damage frequencies that are the same magnitude as internal events. It is not unusual to see the

combined core damage frequencies for these events in the area of $1\text{E-}4$ per year. The varying methods used in the ABWR PRA to evaluate external events are acceptable to the staff because they provide the insights necessary to determine if any design or procedural vulnerabilities exist for these external events and because the methods provide insights needed for the RAP, ITAAC, TS, and other important programs.

To help confirm that no vulnerabilities are introduced when a site is chosen for an ABWR (e.g., site-specific external floods can flood higher than assumed in the design basis so a vulnerability, such as an external, nonwatertight door, could exist), the following steps need to be taken. The staff requires COL applicants to perform a site-specific PRA-based analysis that searches for vulnerabilities from site-specific external events not evaluated in the ABWR PRA and that are not enveloped by the ABWR PRA assumptions (e.g., seismic-induced soil liquefaction). This PRA-based analyses must also search for vulnerabilities in the parts of the ABWR design that were not part of the design certification (e.g., the service water pump house).

The ACIWA system is very important to preventing and mitigating severe accident external events. GE and the staff understand the importance of the ACIWA system being able to function following various external initiators. For this reason, GE developed a COL action item (detailed in Section 19.9 and Table 1.9 in the SSAR) that outlines the following:

- The COL applicant should design the building to house the ac-independent water addition (ACIWA) pumps in such a manner that the building is capable of withstanding high seismic events, river flooding, and other site-specific external events such as high winds (e.g., hurricanes). The capability of the building housing the ACIWA pumps should be evaluated in the site-specific PRA to assure that vulnerabilities do not exist for the specific site.

19.1.3.3.1 Seismic Events

A PRA-based margins analysis systematically evaluates the capability of SSCs in the design to withstand an earthquake, but does not estimate the core damage frequency from seismic events. The margins analysis is a way of estimating how much larger an earthquake than the SSE the design should be able to withstand without sustaining core damage.

Table 19.1-5 SSCs identified by importance measure values

Structure, System, or Component (SSC)	Risk Achievement Worth	Fussell-Vesely Importance Measure (%)
Multiplex Transmission Network (CCF)	204,400	12.1
Trip Logic Units	204,300	6.0
Remote Multiplexing Units	204,300	6.0
Station Batteries	13,160	3.3
Digital Trip Modules (CCF)	281	0.1
Level 2 Sensors (CCF)	273	0.1
SRV (CCF)	189	0.1
RHR Flow Transmitters (CCF Miscalibration)	32	0.2
Level 8 Sensors (CCF Miscalibration)	28	0.1
Combustion Turbine Generator (CTG)	14	69.6
Both Offsite Power Sources	14	1.3
Division 1 Transmission Network Failure (EMS)	13	0.70
1st ESF RMU Division 1 Fails	13	0.34
2nd ESF RMU Division 1 Fails	13	0.34
RCIC Turbine	12	12.0
RCIC Pump	12	7.4
RCIC System (Unavailable, Test or Maintenance)	12	21.8
RCIC Flow Sensor E51-FT007-2 Fails	12	0.32
RCIC Isolation Valve F036 Fails (NOFC)	12	0.18
RCIC Isolation Valve F035 Fails (NOFC)	12	0.18
RCIC Isolation Valve F039 Fails (NOFC)	12	0.18
RCIC Check Valve E51-F003 Fails to Open	12	0.15
RCIC Check Valve F038 Fails to Open	12	0.15
RCIC Outboard Check Valve F005 Fails to Open	12	0.15
NBS Isolation Check Valve B21-F003B (FW Isolation) Fails Closed	12	0.15
NBS Isolation Check Valve B21-F004B (FW Isolation) Fails Closed	12	0.15
NBS Manual Valve B21-F005B (FW Isolation) Fails Closed (NOFC)	12	0.14
RCIC Pres Sensor PIS-Z605 Miscalibrated	12	0.054
RCIC Flow Sensor FT-007-2 Miscalibrated	12	0.054
RCIC Pressure Sensor E51-PIS-Z605 Fails	12	0.013
Failure of Division I Distribution Panel	12	0.0064
SLU/EMS Link for Division I SLU 1 Fails (RCIC Fails)	12	0.00046
SLU/EMS Link for Division I SLU 2 Fails (RCIC Fails)	12	0.00046
RCIC Turbine Exhaust Isolation Valve F039 Limit Switch Fails	12	0.74
RCIC Steam Supply Bypass Valve F045 Limit Switch Fails	12	0.74
RCIC Turbine Lubrication System	12	4.6
RCIC Min. Flow Bypass Valve E51-F011 (NOFO)	12	2.0
RCIC Min. Flow Bypass Valve E51-F011 (NCFC)	12	1.9
RCIC Injection Valve E51-F004 (NCFC)	12	1.9
RCIC Steam Supply Valve E51-F037 (NCFC)	12	1.9
RCIC Isolation Signal Logic	12	1.5
All 3 Diesel Generators, CCF	11	unknown
SP Temp High (Loss of Pump Head)	6.6	0.00055
SRVs	4.3	1.0
Single Offsite Power Line	4.1	3.1
HPCF Maintenance Valve E22-F005B	2.7	1.7
HPCF Pump	2.6	1.1

Severe Accidents

Table 19.1-6 Sensitivity to test or maintenance (T/M) outages

Single system perturbations (factor of five) to base case T/M of two percent unavailability

<u>System</u>							
RCIC	0.02	0.1*	0.02	0.02	0.02	0.02	0.02
HPCFB	0.02	0.02	0.1	0.02	0.02	0.02	0.02
HPCFC	0.02	0.02	0.02	0.1	0.02	0.02	0.02
RJRA	0.02	0.02	0.02	0.02	0.1	0.02	0.02
RHRB	0.02	0.02	0.02	0.02	0.02	0.1	0.02
RHRC	0.02	0.02	0.02	0.02	0.02	0.02	0.1
Core Damage Frequency	1.6E-7 ¹	2.9E-7	1.8E-7	1.6E-7	1.6E-7	1.6E-7	1.6E-7
Percent Increase	N.A.	86	13	< 1	< 1	< 1	< 1

* Unavailabilities in bold type represent perturbations to the base case in the first column.

Effect of single systems completely removed from service

<u>System</u>							
RCIC	0.02	1.0*	0.02	0.02	0.02	0.02	0.02
HCPFB	0.02	0.02	1.0	0.02	0.02	0.02	0.02
HPCFC	0.02	0.02	0.02	1.0	0.02	0.02	0.02
RHRA	0.02	0.02	0.02	0.02	1.0	0.02	0.02
RHRB	0.02	0.02	0.02	0.02	0.02	1.0	0.02
RHRC	0.02	0.02	0.02	0.02	0.02	0.02	1.0
Core Damage Frequency	1.6E-7	1.8E-6	4.1E-7	1.6E-7	1.6E-7	1.6E-7	1.6E-7
Percent Increase	N/A	1073	162	4	5	2	< 1

* Unavailabilities in bold type represent single systems that have been removed from service.

Table 19.1-6 Sensitivity to test or maintenance (T/M) outages (continued)

Multiple system perturbations (factor of five) to base case

System							
RCIC	0.1*	0.02	0.02	0.1	0.02	0.1	0.1
HPCFB	0.02	0.1	0.02	0.1	0.1	0.1	0.1
HPCFC	0.02	0.02	0.1	0.02	0.1	0.1	0.1
RHRA	0.1	0.02	0.02	0.02	0.02	0.02	0.1
RHRB	0.02	0.1	0.02	0.02	0.02	0.02	0.1
RHRC	0.02	0.02	0.1	0.02	0.02	0.02	0.1
Core Damage Frequency	2.9E-7	1.8E-7	1.6E-7	3.1E-7	1.8E-7	3.2E-7	3.2E-7
Percent Increase	88	14	< 1	101	13	102	104

* Unavailabilities in bold type represent perturbations to a base case value of 0.02.

The capability of a particular SSC to withstand beyond design bases earthquakes is measured in terms of an HCLPF given a certain input acceleration. The HCLPF has units of acceleration. An HCLPF value represents the acceleration that approximates the concept of having about 95 percent confidence that the SSC will fail less than about 5 percent of the time. The ABWR is designed to withstand a 0.3g SSE. Since the analysis used in evaluating the capability of SSCs to withstand the SSE has significant margin in it, the staff expects that a plant built to withstand a 0.3g SSE actually will be able to withstand a much larger earthquake. The staff indicated that it expects that a plant truly designed to withstand a 0.3g SSE should have a plant HCLPF at least 1.67 times the SSE. In SSAR Section 19H.5, GE provided a COL action item to assess how the as-built facility corresponds to the assumptions in the seismic margins analysis. As discussed below, the ABWR design as analyzed meets the 1.67 times the SSE expectation. Thus, GE and the staff believe that the ABWR can be designed and built in a reasonable, cost-effective manner to achieve a plant HCLPF of 0.5g or higher. The staff notes that the ABWR seismic margins analysis assumes that no soil liquefaction will occur (regardless of the g-level) or that liquefaction would not affect the plant HCLPF. COL applicants will need to confirm (with their site-specific seismic risk assessment) that soil liquefaction does not introduce any vulnerabilities. The staff finds this acceptable.

19.1.3.3.1.1 Dominant Accident Sequences

In the PRA-based margins method, the event trees and fault trees for internal events are modified to accommodate

seismic events. In this way, the random failures and human errors modeled in the internal events portion of the PRA are captured in the seismic analysis.

In Table 19.1-7 in this report, 25 sequences are identified that lead to core damage in the ABWR PRA-based seismic margins analysis, listed by plant damage class. The first HCLPF value given for each sequence is the HCLPF assuming that only seismic failures can occur (i.e., without any random failures or human errors). Additional HCLPF entries under a sequence include various combinations of random failures and human errors. The underlying assumption that earthquakes exceeding the SSE will happen less frequently than once in a 1,000 years allows us to exclude random failures or human errors less than 1E-3. This is because the combination of seismic events more than 1.67 times the SSE with random failures lower than 1E-3 would result in core damage frequency estimates much less than 1E-6 per year. We also exclude seismic-random combinations where the seismic portion has an HCLPF at least as high as the seismic-only HCLPF.

Further details are provided in Table 19.1-8 in this report, which provides the "dominant" cut sets for each plant damage state. The word "dominant" appears in quotes to emphasize that the use of this terminology in the context of a margins study should not be taken in the same way as it would be for a conventional PRA. While these sequences and cut sets dominate the HCLPF values for the plant, the margins approach does not permit a determination that these are the dominant contributors to seismic risk in a probabilistic sense.

Severe Accidents

Table 19.1-7 Results of GE HCLPF quantification using the MIN/MAX approach (by plant damage class)

Damage Class	Sequence Number	Sequence Description	Seismic HCLPF	Seismic/Random HCLPF
IA	5	/SI*LOP*APW*/DP*/C*/UR*X*/HX	0.74 g	None
	9	/SI*LOP*APW*/DP*/C*UR*X*/HX	0.74 g	None
	13	/SI*LOP*AW*DP*/HX	0.74 g	None
	18	/SI*LOP*/APW*/C*/PC*UR*UH*X	0.74 g	None
	20	/SI*LOP*/APW*/C*PC*UH*X	0.74 g	None
	Total		0.74 g	None
IB-2	3	/SI*LOP*APW*/DP*/C*/UR*/X*FA*/HX	0.62 g	None
	Total		0.62 g	None
IC	21	/SI*LOP*/APW*C*/C4*/LPL*/PC*/PA*UH*UR	0.70 g	0.62g * 6E-2
	23	/SI*LOP*/APW*C*/C4*/LPL*PC*/PA*UH	0.74 g	0.62g * 1E-1
	27	/SI*LOP*/APW*C*C4*UH	0.62 g	None
	Total		0.62 g	None
ID	7	/SI*LOP*APW*/DP*/C*UR*/X*FA*/HX	0.70 g	0.62g * 6E-2
	17	/SI*LOP*/APW*/C*/PC*UR*UH*/X*V1*V2	0.70 g	0.62g * 6E-2
	19	/SI*LOP*/APW*/C*PC*UH*/X*V1*V2	0.74 g	0.62g * 2E-3
	Total		0.70 g	0.62g * 6E-2
IE	15	SI*/HX	1.11 g	None
	Total		1.11 g	None
IV (incl. IV-P)	11	/SI*LOP*APW*/DP*C*/HX	0.62 g	None
	12	/SI*LOP*APW*/DP*C*HX	0.70 g	None
	22	/SI*LOP*/APW*C*/C4*/LPL*/PC*PA	0.74 g	0.62g * 2.4E-3
	24	/SI*LOP*/APW*C*/C4*/LPL*PC*PA	0.74 g	None
	25	/SI*LOP*/APW*C*/C4*LPL	0.74 g	0.62g * 1E-2
	26	/SI*LOP*/APW*C*C4*/UH	0.62 g	None
	Total		0.62 g	None

Table 19.1-7 Results of GE HCLPF quantification using the MIN/MAX approach (by plant damage class) (continued)

Damage Class	Sequence Number	Sequence Description	Seismic HCLPF	Seismic/Random HCLPF
IA-P through IE-P	4 (IB2-P)	/SI*LOP*/APW*DP*/C*/UR*/X*FA*HX	0.70 g	None
	6 (IA-P)	/SI*LOP*APW*/DP*/C*/UR*X*HX	0.74 g	None
	8 (ID-P)	/SI*LOP*APW*/DP*/C*UR*/X*FA*HX	0.70 g	None
	10 (IA-P)	/SI*LOP*APW*/DP*/C*UR*X*HX	0.74 g	None
	14 (IA-P)	/SI*LOP*AW*DP*HX	0.74 g	None
	16 (IE-P)	SI*HX	1.11 g	None
	Total		0.70 g	None

NOTE: An entry of "none" in the seismic/random HCLPF column means that either (1) there were no combinations in which the random portion was greater than 1E-3 or (2) there were no combinations in which the HCLPF of the seismic portion of the combination was less than the HCLPF from seismic only.

LEGEND:

APW = Failure of Emergency AC Power or Service Water
 C = Failure of Reactivity Control System
 C4 = Failure of Standby Liquid Control System
 DP = Failure of dc Power
 FA = Failure of Fire Water System
 HX = Rupture of RHR Heat Exchanger
 LOP = Loss of Offsite Power
 LPL = Failure of Primary Level and Pressure Control
 PA = Failure to Inhibit ADS Actuation
 PC = Failure of SRVs to Close
 SI = Collapse of Plant Essential Structures
 UH = Failure of High Pressure Core Flooder
 UR = Failure of Reactor Core Isolation Cooling System
 V1 = Failure of Low Pressure Core Flooder
 V2 = Failure of Condensate Injection
 X = Failure of Primary Depressurization

A slash (/) appearing before a designator means the occurrence of the opposite condition (i.e., success rather than failure).

Severe Accidents

Table 19.1-8 "Dominant" contributors to HCLPF using the MIN/MAX approach (by plant damage class)

DAMAGE CLASS	SEISMIC ONLY "DOMINANT CUT SETS"				SEISMIC/RANDOM "DOMINANT CUT SETS"
IA	Do Cable Trays (0.74 g)				None
	SW HVAC (0.63 g) or SW Pump (0.62 g) or SW Pump House (0.60 g) or Diesel Gen. (0.62 g) or Transformer (0.62 g) or MCCs (0.62 g) or Switch (0.63 g) or HPCF Pump (0.62 g)		S/Rvs (0.74 g)		
IB-2	SW HVAC (0.63 g) or SW Pump (0.62 g) or Diesel Gen. (0.62 g) or Transformer (0.62 g) or MCCs (0.62 g) or Switch (0.63 g)		Fire Pump (0.62 g)		None
IC	Fuel Ass. (0.62 g) or HCU (0.63 g)	SLC Tank (0.62 g) or SLC Pump (0.62 g)	HPCF Pump (0.62 g)		None

How to read this table:

The columns for seismic only and seismic random are independent. Within each of those columns, the or function is obvious, and the and function is represented by a horizontal dotted line. Thus, for Damage Class IA, the seismic only column represents the Boolean expression [(SW HVAC + SW Pump + Diesel Gen. + Transformer + MCCs + Switch + HPCF Pump) * S/Rvs] and the same column for Damage Class IC represents the Boolean expression [Fuel Ass. * (SLC Tank + SLC Pump) * HPCF Pump].

Table 19.1-8 "Dominant" contributors to HCLPF using the MIN/MAX approach (by plant damage class) (continued)

DAMAGE CLASS	SEISMIC ONLY "DOMINANT CUT SETS"				SEISMIC/RANDOM "DOMINANT CUT SETS"		
ID	SW HVAC (0.63 g) or SW Pump (0.62 g) or SW Pump House (0.60 g) or Diesel Gen. (0.62 g) or Transformer (0.62 g) or MCCs (0.62 g) or Switch (0.63 g) or HPCF Pump (0.62 g)	RCIC Pump (0.70 g)	Fire Pump (0.62 g)		SW HVAC (0.63 g) or SW Pump (0.62 g) or SW Pump House (0.60 g) or Diesel Gen. (0.62 g) or Transformer (0.62 g) or MCCs (0.62 g) or Switch (0.63 g) or HPCF Pump (0.62 g)	Fire Pump (0.62 g)	RCIC (6E-2)
	RCIC Pump (0.70 g)	HPCF Pump (0.62 g)	LPCF Pump (0.56 g)	V2 (0.62 g)			
IE	Reactor Building (1.12 g) or Control Building (1.11 g)				None		
IV (incl. IV-P)	SW HVAC (0.63 g) or SW Pump (0.62 g) or SW Pump House (0.60 g) or Diesel Gen. (0.62 g) or Transformer (0.62 g) or MCCs (0.62 g) or Switch (0.63 g) or SLC Tank (0.62 g) or SLC Pump (0.62 g)	Fuel Assemblies (0.62 g) or HCU (0.63 g)			None		
IA-P through IE-P	SW HVAC (0.63 g) or SW Pump (0.62 g) or SW Pump House (0.60 g) or Diesel Gen. (0.62 g) or Transformer (0.62 g) or MCCs (0.62 g) or Switch (0.63 g)	RHR HtEx (0.70 g)	Fire Pump (0.62 g)		None		

How to read this table:

The columns for seismic only and seismic random are independent. Within each of those columns, the or function is obvious, and the and function is represented by a horizontal dotted line. Thus, for Damage Class IA, the seismic only column represents the Boolean expression [(SW HVAC + SW Pump + Diesel Gen. + Transformer + MCCs + Switch + HPCF Pump) * S/Rvs] and the same column for Damage Class IC represents the Boolean expression [Fuel Ass. * (SLC Tank + SLC Pump) * HPCF Pump].

Severe Accidents

GE used two methods to calculate its HCLPFs: min/max and convolution. The min/max method provides more safety insights. The convolution method may more appropriately (in a pure mathematical sense when a margins analysis is performed) take into account the fact that when one combines HCLPFs, one is working with the tails of probability distributions. If random failures and human errors are ignored when the min/max method is used, the plant HCLPF is 0.6g. With the min/max method, if random failures and human errors are included, the most significant combination of HCLPF, human errors, and random failures for the plant is also 0.6g. When the convolution method is used, the plant HCLPF is about 0.69g if random failures and human errors are ignored. All of these values exceed the 1.67 times SSE that represents the seismic robustness the staff expects to find.

For earthquakes that generate higher accelerations than the plant HCLPF, the staff no longer has the same high degree of confidence that core damage will not occur. However, the staff does not believe that a cliff-effect will exist for the ABWR design at or near the plant HCLPF and expects that the plant will have margin (perhaps quite a bit) above the HCLPF value.

The staff defines the most important seismic core damage sequences as those that have the lowest HCLPFs (seismic only) or the lowest combination of HCLPF with random failure or human error. Using these as the selecting criteria, none of the most important seismic core damage sequences involve random failures or human errors. The following are the most important sequences:

- (loss of emergency ac power or loss of emergency service water) and (fire water injection fails)
- (loss of emergency ac power or loss of emergency service water) and (scram fails)
- (scram fails) and (standby liquid control fails)
- (scram fails) and (standby liquid control fails) and (high-pressure core flood fails).

These results are similar to those from seismic margins analyses and PRAs performed previously.

The staff focused on the performance of and insights drawn from the ABWR seismic margins analysis. Because of the large uncertainty in hazard curves and the fact that seismic PRA results are dominated by the tails of the site hazard and SSC's fragility curves, it is expected that if a seismic PRA had been performed, it would have been one of the largest contributors to core damage frequency (though still a low absolute value). This is particularly

true for the ABWR design since design changes or improvements have greatly reduced the estimated core damage frequency from internal events. As it is, the ABWR design should be better able to resist seismic events than most, if not all, existing nuclear power plants east of the Rocky Mountains because of its built-in safety margins.

The staff did not require GE to examine the HCLPFs of paths by which the containment could be bypassed, fail to isolate, or fail, since the containment structure was considered to be very rugged and none of the sequences that led to core damage had HCLPFs less than 1.67 times the SSE. Nevertheless, GE performed an evaluation of beyond-design-basis earthquakes to see how the containment would perform under high g-levels. This evaluation showed that no bypass paths are expected to occur with an HCLPF of less than 1.67 times the SSE. The lowest min/max HCLPF for bypass reported by GE was 0.74g.

19.1.3.3.1.2 PRA as a Design Tool

The following are examples of ways in which the ABWR design or procedures were modified by GE, based on the ABWR PRA-based seismic margins analysis:

- GE switched the seismic qualifications of the fire water pumps (i.e., the diesel-driven fire water pump took the place of the motor-driven pump as the pump capable of surviving an SSE) because failure of the diesel generators following a seismic event (by seismic or random failure) would otherwise have left the plant with no fire water addition capability.
- GE chose to lower the capacities of a few SSCs since it might have been difficult for a COL applicant to achieve the capacities in a cost-effective manner. The staff found this acceptable, since the lowered capacities did not affect the overall HCLPF of the plant and the original capacity assumptions seemed to be slightly higher than normally assumed in eastern U.S. seismic fragility analyses.
- GE developed recommendations for improving emergency operating procedures by instructing the operator to manually operate heat removal system valves if seismic-induced transformer loss should make power operation of these valves impossible.

19.1.3.3.1.3 Plant Features and Operator Actions Important to Risk

The margins approach does not allow a determination of which plant features are most important to risk. It does allow one to determine which plant features are important

to the plant-level HCLPF and the redundancy or diversity available in achieving that HCLPF. In order to make this determination, the staff examined each sequence that led to core damage on the seismic event trees. None of the sequences has a seismic-only HCLPF of less than 0.5g. The sequences were examined to determine if lowering the HCLPF value of a single SSC (to a much lower HCLPF value) or increasing the demand failure rate of a single system (to a much higher demand failure rate) would result in a plant HCLPF of less than 0.5g. A review of the cut sets in Table 19.1-8 shows that most of the important sequences require at least two failures of SSC with HCLPFs above 0.5g. The two cases where this is not true are discussed below:

- **Structural integrity** — Most of the safety-significant structures (except for the service water pump house) are assumed to have an HCLPF of 1.11g or higher. If any of these structures were to be built with an HCLPF much lower than 0.5g, it would result in a much lower plant HCLPF. These structures include the control building, reactor building, containment, service water pump house, reactor pressure vessel (RPV) supports, and RPV pedestal.
- **Batteries** — In a seismic event, it is assumed that offsite power will be lost and ac power must be supplied by the EDGs. For the EDGs to start and load, they require dc power. The battery chargers or inverters, which provide dc power when ac power is available, are lost on loss of offsite power, until the EDGs start and load. However, the EDGs will not start and load unless dc power is available. Therefore the EDGs rely on the batteries to start and load in these circumstances. If the batteries or the dc cable trays were to have an HCLPF of less than 0.5g, it would lower the plant HCLPF accordingly.

All other seismic sequences require multiple failures of SSC the HCLPF of which is greater than 0.5g in order to cause the plant to experience core damage. As noted in SSAR Sections 19.9.5 and 19.H.5, a check of the capacity of as-built SSCs to meet the HCLPFs assumed in the ABWR PRA will be provided by a seismic walkdown. Details are to be developed by the COL applicant.

19.1.3.3.1.4 Human Reliability Insights and Important Human Actions

In the margins analysis, GE used the same human error rates and random failure rates that were used in the ABWR internal events analysis. The PRA-based seismic margins analysis did not identify any human reliability insights that were not already identified in the internal events analyses.

There were no human actions or random failures that contributed to the plant HCLPF.

19.1.3.3.1.5 COL Action Items

GE identified COL actions that define the areas derived from performing the ABWR PRA-based seismic margins analysis that the COL applicant needs to perform or complete. These actions, including plant walkdowns, are detailed in SSAR Section 19.9 and are acceptable.

19.1.3.3.1.6 Insights from Uncertainty, Importance, and Sensitivity Analyses

One of the reasons for performing an uncertainty analysis is to help to display the range of values within which the results of an analysis could reasonably be expected to fall. The use of a PRA-based seismic margins analysis inherently makes use of the breadth of information being considered. This is because HCLPF values can be thought of as the g-level at which one has a 95 percent confidence that less than 5 percent of the time the equipment will fail (i.e., we are dealing with the tails of the curves). In addition, the staff does not require that a seismic hazards analysis be convoluted with equipment fragilities, since hazard curves have a large uncertainty that reduces their value in helping to make judgements about the seismic risk. From seismic PRA analyses, it is clear that uncertainties in the hazard curves would dominate the uncertainties in equipment or structure fragilities. For the ABWR PRA-based seismic margins analysis, no uncertainty analysis was performed because uncertainty is directly reflected in the margins method. Similarly, sensitivity studies were not performed on the margins analysis. Finally, the margins method does not result in either core damage frequency or risk results. Therefore, importance analyses were not performed.

19.1.3.3.2 Fire

In a number of PRAs for operating plants, fires have shown up as important contributors to core damage frequency and risk. GE has taken a unique and effective approach to analyzing beyond-design-basis fires. In performing its fire analysis (a combination of the FIVE methodology and the ABWR internal events PRA), GE chose to simplify its analysis by assuming that any fire that started in a divisional fire area while the plant was at power would immediately cause all equipment to fail in that divisional fire area. While this assumption greatly simplifies the performance of a fire analysis, it biases the numerical results in a conservative manner. GE suggests and the staff agrees that it would be inappropriate to compare the numerical results from the ABWR fire analysis to the results of the more realistic internal events

Severe Accidents

analyses. Analysis of the potential consequences of a fire while in Modes 3, 4, and 5 are investigated in the shutdown risk analysis discussed in Section 19.1.3.4.

The ABWR design has a number of attributes that have significantly improved the detection, suppression, and confinement potential of fires. The most important improvement is that the ABWR was designed with fire prevention and mitigation in mind. The ABWR does not need to rely on spatial separation as a barrier between safety divisions. The three safety divisions are separated by 3-hour fire barriers throughout the plant, with few exceptions (See SSAR Section 9A.5). Detection systems and suppression systems will be placed for optimal advantage.

19.1.3.3.2.1 Dominant Accident Sequences

Because GE did not perform a fire PRA, but rather used a combination of the FIVE methodology and the ABWR internal events PRA, one cannot determine dominant sequences in the sense normally used when working with a PRA. For the fire analysis, the word "dominant" appears in quotes to emphasize that the use of this terminology in the context of a fire analysis should not be taken in the same way as it would be for a PRA (i.e., no cut sets were developed for the fire analysis and therefore importance analyses cannot be performed). The staff identified the "dominant" sequences leading to core damage in the systematic fire analysis performed by GE on the basis of engineering judgement, core damage frequency point estimates, and descriptions provided by GE of what it considers the most important sequences. The staff believes this approach is adequate to identify fire vulnerabilities.

GE identified the most important sequences as those that begin with a fire in the control room. In these "dominant" sequences, a control room fire starts, feedwater is lost, RCIC or one high-pressure core flood train is lost, and either one train of low-pressure core flood or manual depressurization fails. The fire analysis screened out all additional sequences and fire areas because the estimated screening core damage frequency associated with each fire area was less than $1\text{E-}6$ per year.

The staff finds $1\text{E-}6$ per year to be an acceptable screening level for the ABWR since the fire analysis methodology used by GE assumes that any fire that starts in a physically separated divisional area instantaneously causes all the equipment in that division to fail. Although this assumption is conservative, the separation of divisions in the ABWR design still results in low core damage frequency estimates.

19.1.3.3.2.2 PRA as a Design Tool

On the basis of the ABWR fire analysis, GE made a design change to assure that certain fire protection capabilities existed in the as-built plant. The change was made because GE determined that unless the capability to control ADS valves from the remote shutdown control panel was improved (controls for a fourth safety relief valve (SRV) were added) and unless it was possible to control the RCIC system locally (RCIC could not be controlled from the remote shutdown panel), the control room would not pass the $1\text{E-}6$ per year screening value.

19.1.3.3.2.3 Plant Features and Operator Actions Important to Risk

Several plant features and operator actions are important to reducing the consequences of a fire in an ABWR.

Design Features

- Separation of the three safety divisions — The most important is the 3-hour fire barrier that is to be in place among the three safety divisions. The fire analysis assumes that there will be no spread of fire or suppressants among the divisions; once a fire begins.
- Smoke control system — This system helps prevent the migration of smoke to other divisions if a fire occurs. This is accomplished by pressurizing the surrounding areas so that the smoke will be contained in the fire zone. Smoke control is particularly important in secondary containment.
- Inerting of containment at power — The inerting of containment precludes the need to analyze fires in containment while at power. Containment is one of the few areas in the plant where all four divisions (three safety, one non-safety) are in the same fire area. The control room is another area where multiple divisions coexist.
- Capability to operate RCIC from outside the control room — The largest contributor to core damage frequency among fire-initiating events is a control room fire. For control room fires, it is important to be able to control RCIC locally, since it cannot be controlled directly from the remote shutdown panel. This is both a design feature and an important operator action.
- Capability to operate four SRVs from the remote shutdown panels — As with RCIC, for control room fires, it is important to be able to operate four SRVs from the remote shutdown panel. Successful

depressurization is achieved by manual operation of three of the four SRVs. If only three SRVs were on the panel, a single failure of one of the SRVs would prevent successful mitigation of the control room fire. This is both a design feature and an important operator action.

- Remote shutdown panel — The panel is required for shutdown of the plant in the event that the control room needs to be evacuated.

19.1.3.3.2.4 Human Reliability Insights and Important Human Actions

The ABWR fire analysis considered random failures and human errors modeled in the internal events analysis. Operator control of the RCIC system from outside the control room was found to be an important human action, should the control room need to be evacuated.

19.1.3.3.2.5 Insights From Uncertainty, Importance, and Sensitivity Analyses

Because the methodology used in performing the ABWR severe accident fire analysis (a combination of the FIVE analysis, which is primarily a deterministic evaluation of fires, and portions of the ABWR PRA internal events analysis) is bounding in nature, the results of uncertainty, sensitivity, or importance analyses would be biased. Since the purpose of performing an uncertainty analysis is to better understand the subject being investigated and since in the case of the fire analysis it is unclear what the results of an uncertainty analysis would represent physically given the bounding nature of the assumptions in the analysis, the staff finds that an uncertainty analysis was not required. The staff concludes that the fire methodology used by GE provides the insights needed to determine whether fire vulnerabilities exist for severe accident fires and whether fires represent a disproportionate risk.

19.1.3.3.3 Internal Floods

At some plants, PRAs have shown that internal floods are leading contributors to core damage frequency. Utilities that have performed a systematic internal floods analysis have generally found design modifications that they deemed prudent to implement in order to maintain adequate prevention and mitigation capability for internal floods. GE performed a comprehensive internal flood PSA for ABWR power operation. In the shutdown heat removal analysis, GE performed a separate flooding analysis that evaluated the flood protection provided by the plant during shutdown by controlling barriers between divisions and by controlling equipment configurations.

19.1.3.3.3.1 Dominant Accident Sequences

The ABWR Probabilistic Flooding Analysis (flood analysis) in Appendix 19R of the SSAR estimates that the chances of an internal flood causing core damage is very small (about $7E-9$ per year).

For the internal flooding analysis, the word "dominant" appears in quotes to emphasize that the use of this terminology in the context of the flooding analysis should not be taken in the same way as it would be for a PRA performed using more realistic assumptions or for an analysis that has cutsets on which importance analyses can be performed. The ABWR internal flooding analysis made several conservative assumptions that could bias the results. GE has identified the "dominant" sequences leading to core damage in the internal flooding analysis on the basis of sequences with the largest estimated core damage frequency and on engineering judgment.

The "dominant" sequences that are initiated in the turbine building are those associated with the circulating water system (CWS) or the turbine service water (TSW) system. The following sequence is "dominant" for the turbine building: a large pipe breaks in the CWS system, the isolation valves in the CWS lines fail to close, water fills up and runs out of the condenser pit, and the fire door between the turbine building and the service building is either open or fails open allowing water into the service building. The service building floods and a door between the service building and the control building fails open or is open. Water enters the control building and causes electrical power supplies and all three divisions of reactor building cooling water (RCW) to fail. The estimated frequency of this event is $3E-9$ per year.

The most important internal floods initiated in the control building are those associated with the reactor service water (RSW) system. This sequence involves the following: a large pipe breaks in the RSW piping in the RSW/RCW room and the operator fails to isolate the flooding. The automatic RSW pump trip fails and the water flows into the remaining RSW pump rooms. The operator fails to respond to the flooding alarm and the RSW fails. The estimated frequency of this event is $2E-9$ per year.

Floods that begin inside the reactor building are divided into two parts: those inside and those outside secondary containment. The most important internal flood sources that initiate inside secondary containment are the suppression pool, condensate make-up, and fire water. The lowest floor of the reactor building is entirely within the secondary containment. On this level, each of the three safety divisions has a separate ECCS room that has an alarmed watertight door and a sump pump. Floods

Severe Accidents

inside these rooms would cause the HPCF, RCIC, RHR, and control rod drive (CRD) pumps, and the CRD hydraulic control unit to fail. A common hallway into which the watertight doors open runs around the perimeter of the division rooms. There are three worst-case flood sequences for the reactor building. The three flood sequences are developed in event trees in SSAR Appendix R, Figures 19R.5-4, -5, and -6. The largest estimated contributor to core damage among these three flood sequences occurs inside an ECCS room resulting from a leak in the suppression pool suction line upstream of the isolation valve. This is a nonisolable break.

In the review of the risk resulting from internal floods, several concerns were identified. The main concerns identified for turbine building floods were (1) the assumptions regarding the likelihood of the truck door failing and allowing turbine building flood waters to exit the turbine building without challenging the fire door between the turbine building and the service building and (2) the justification for the assumed reliability for the fire door. GE satisfactorily addressed these staff concerns in Section 19R of the SSAR. In the reactor building, the chief concern identified was the lack of any common-cause failure analysis for the failure of the isolation measures credited in the analysis. GE performed an improved common-cause failure analysis that resulted in no change in internal flood insights. For control building floods, the staff was concerned that failure of the discharge or intake valves following a service water system pipe break in an RSW or RCW room would result in an unisolable flood that could affect all three divisions of the RSW system. GE added design requirements for antisiphon capability to prevent continued flooding in the event the RSW pump is tripped but the isolation valves do not close. In addition, GE stated that the ABWR would be designed so that the ABWR ultimate heat sink (UHS) cannot gravity drain into the control building.

19.1.3.3.2 Comparison of Dominant Sequences

Conservatism in the ABWR flooding analysis biases the results and may make it unsuitable for comparison to the more realistic internal events results. Conservatism includes the assumptions that all piping breaks are double-ended shears and once a flood in a division causes one piece of equipment to fail, all equipment in that division is assumed to fail.

19.1.3.3.3 PRA as a Design Tool

In performing its internal flooding analysis, GE determined that several areas of the design needed to be modified or strengthened to help make sure that internal floods do not

present an unacceptable level of risk for the ABWR design. These modifications include the following:

- GE modified the motor control centers (MCCs) to have NEMA Type 4 enclosures to protect the MCCs from water spray from a pipe break or leak.
- GE added four water-level sensors in the condenser pit. When actuated, these sensors send an alarm to the control room, trip the CWS and TSW pumps, and close isolation valves in both systems. This isolates floods in the turbine building caused by pipe breaks in the TSW and CWS.
- GE added four room-floor water-level sensors, which send an alarm to the control room when water is first detected in the room, and four diverse sensors, which automatically trip the reactor water service water/RCW pumps at a higher water level in the RCW/RSW room. The sensors will alert the operators to RCW leakage and will isolate the flood.
- GE added to the Tier 1 description of the control building in the design control document (DCD) the design requirement that no more than 4,000 meters (4,374 yds) of pipe exist between the isolation valves at the RSW pump house and the control building (2,000 meters each, for supply and return lines). This limits the amount of water that can be drained into the RCW/RSW room following RSW pump trip during flooding.
- GE added antisiphon capabilities to the RSW system to end a flood if the RSW pump trips but the isolation valves do not close.

19.1.3.3.4 Plant Features and Operator Actions Important to Risk

Several plant features are important in reducing the chances or consequences of an internal flood. These features are discussed below:

- Separation of the three safety divisions — Whenever the three safety divisions are in a building, they are separated by barriers. The most important barrier is the 3-hour fire barrier that must be in place between the three safety divisions. The flooding analysis assumes that there is a limited spread of water among the divisions, once a flood begins. The primary assurances that flooding will not get high enough to affect SSCs in multiple divisions are that (1) the 3-hour barriers are in place, (2) there are elevator shafts with no sills that carry water to lower reactor building levels, (3) stairwells on all floors have doors that

provide a leak path (up to a 1.9 cm (0.75 in.) gap at bottom of fire door) down the stairs to the lowest building level, (4) there is at least one drain in every room, (5) all pipes penetrating floors have 200 mm (8 in.) sills that help prevent sneak-path leaks, (6) all hatches and electrical penetrations through floors are to be able to withstand 200 mm (8 in.) of standing water without leaking, and (7) except as noted in SSAR Section 19R.4.2.3, equipment from multiple divisions does not occupy the same divisional area. Barriers include concrete fire barrier floors, ceilings, and walls; partitions; rated watertight doors; penetration seals for process pipes and cable trays; and special assemblies and constructions.

- Auto trip of the CWS and TSW pumps — The three CWS pumps and the two TSW pumps provide unlimited supplies of water to the turbine building. If a flood were to occur, level sensors would alert the control room operators, trip all five pumps, and close all isolation valves in both systems.
- Maximum length of RSW pipe — The COL applicant will have to limit the RSW piping distance between the control building and the RSW isolation valves to 4000 meters (4,374 yds)(2,000 meters each, for supply and return lines). A longer length would provide enough water to flood one RCW room, overflow, and start to flood another RCW room.
- Auto trip of the RSW pumps — The RSW pumps trip and the RSW isolation valves close in the affected division should flooding occur in the control building. The trips limit the volume of water that can be added to a control building flood. Trip sensors are diverse from sensors that alert the operators early on (i.e., at a lower level) to a flood in the room.
- Antisiphon capability — If the RSW pumps trip on a high water level caused by a line break, but the isolation valves fail to close, antisiphon capabilities will be needed to prevent further flooding because otherwise the siphon effect will continue to draw water from the ultimate heat sink into the rooms.
- ECCS room watertight doors — The doors between the ECCS rooms and the outside corridor in the reactor building are watertight. These doors have dogs and alarms. The doors do not send an alarm if they are physically closed, but not "dogged." The doors are designed to limit flooding in a division to the particular ECCS divisional area.

- Reactor building corridor volume — The volume of the reactor building corridor surrounding the three ECCS divisions at the lowest level of the reactor building is sufficiently large to handle large breaks. The corridor can hold water from a suppression-pool-driven flood at the equilibrium water level with the suppression pool and have the water level not exceed the ceiling of the corridor. This limits the flood potential to one ECCS division and the corridor.
- Drip-proof designs and NEMA Type 4 enclosures — All electric motors have drip-proof designs and all motor control centers have NEMA Type 4 enclosures that protect electrical equipment from water spray from above.

Further details of the ABWR design features that help to prevent or mitigate internal flooding are given in SSAR Appendix 19R, Table 19R.6-2. In addition, the following are important insights into the ABWR internal flooding analysis:

- The only buildings modeled in the PRA for flooding where internal flooding could damage safety-related equipment or cause plant transients are the turbine building, control building, and reactor building.
- The service water pump house, which is outside the ABWR certification scope, is a building that must be designed to prevent internal floods from impairing multiple safety trains. Flooding of the pump house would cause all ECCS pumps to fail except for RCIC. For injection to the vessel, the operators also would have the fire water pumps and the condensate pumps.
- Secondary containment in the reactor building is designed to mitigate internal floods that begin at elevations above the lowest elevations by directing flood water to floor drains and stairwells that are routed to the lowest elevation. It is important that the drains be sized to drain at a rate that does not permit a flood on a particular level to rise high enough to damage safety-related equipment, pressurize the volume within which it is contained, or spill over to other divisions.
- From SSAR Appendix 19Q, the recommended shutdown configuration is as follows: one RHR division and its support systems should be operating, the second safety division should be administratively controlled to not be in maintenance and its barriers should be intact, and the third safety division may be undergoing maintenance.

Severe Accidents

- Fire doors generally are not considered to be capable of holding back a large head of water.
- If a watertight door is physically shut but not "dogged," it will not alarm to indicate it is incapable of holding back a flood inside the door.
- The service water system and the CWS are assumed to be designed and located so that they cannot gravity feed to the plant.
- The ACIWA system can provide water to the core or spray the drywell in the event of a catastrophic internal flood. Manual valves to direct the flow to either the core or to the drywell are located in the reactor building and can be operated successfully following an internal flood.

19.1.3.3.3.5 Insights Into Human Reliability and Important Human Actions

Although postulated floods can be mitigated from a risk perspective with few operator actions, because of the inherent ABWR flooding capability (the frequency of internal floods leading to core damage without taking any credit for operation action is still quite low), timely implementation of the following operator actions can limit potential flood damage:

- Isolation of flood sources following detection by sump pump operation and alarms or floor water-level detectors (for floods in the turbine building, the operator should attempt to isolate the leak and shut down the plant without losing condenser vacuum to avoid a "turbine trip without bypass" scenario)
- Closure of watertight doors to prevent damage to equipment in more than one safety division
- Opening of certain nonwatertight doors or hatches to divert water from safety-related equipment.

19.1.3.3.3.6 Combined License Applicant Action Items

GE has identified COL Action Items that define the actions derived from performing the ABWR internal flood analysis that the COL applicant needs to perform or complete. These actions are detailed in SSAR Section 19.9. The staff finds these COL Action Items to be acceptable.

19.1.3.3.3.7 Insights From Uncertainty, Importance, and Sensitivity Analyses

The ABWR internal flooding analysis methodology made bounding assumptions to simplify the task of evaluating

floods. This resulted in an evaluation that cannot readily be manipulated to provide uncertainty, importance, or sensitivity insights because of its internal biases. Therefore, GE did not perform any uncertainty, importance, or sensitivity analyses on its internal flooding analysis nor did the staff require these analyses to be performed. GE identified important design features through engineering judgement. The staff finds this approach to be acceptable.

19.1.3.3.4 External Floods

In SECY-93-087, the staff identified the need for a site-specific probabilistic safety analysis and analysis of external events. GE did not perform an analysis (PRA or bounding) of the capability of the ABWR design to withstand external flooding. Instead, GE assumed that the ABWR standard design would be sited such that its grade level would be 30.5 cm (1 ft) higher than the probable maximum flood level as stated in Section 2.6.2 of the SSAR. However, estimates of the return periods of river floods at various nuclear power plant sites that would exceed the probable maximum flood level range from probable to very improbable. For some sites where the return period of large floods is high, the ABWR design may have vulnerabilities to external flooding.

Therefore, the staff will require, where applicable to the site, that the COL applicant perform a site-specific PRA-based analysis for external flooding to search for site-specific vulnerabilities.

19.1.3.4 Operation in Modes Other Than Full Power and Startup — Level 1

This section details the staff's safety insights drawn from the review of the ABWR shutdown heat-removal reliability study performed by GE. Although the staff found that the most significant events to date have occurred at pressurized water reactors (PWRs), the potential vulnerability of BWR plants to shutdown and low-power events cannot be ignored. GE submitted a shutdown risk evaluation of the ABWR design (SSAR Appendix 19Q). The evaluation covered Modes 3 (hot shutdown), 4 (cold shutdown), and 5 (refueling). It included all aspects of the nuclear steam supply system (NSSS), containment, and all systems that support the NSSS and containment. It did not address fuel handling outside the primary containment or fuel storage in the spent fuel pool.

The evaluation covered important aspects of draft NUREG-1449, "NRC Staff Evaluation of Shutdown and Low Power Operation," such as decay heat removal (DHR), inventory control, containment integrity, electrical power, reactivity control, and instrumentation. The

analysis examined 70 loss-of-power and more than 100 loss-of-DHR precursor events at operating BWRs. For new features of the ABWR design, GE developed a failure-modes-and-effects-analysis to determine if the new features would introduce additional shutdown vulnerabilities. GE performed a detailed reliability study of the ABWR DHR function. This study included fault and event trees for all DHR and make-up systems. The analysis separately investigated fire and flooding at shutdown, along with the potential effect of increased maintenance of important equipment.

The staff finds that the ABWR design includes enhanced features that reduce risk during shutdown operations when compared with operating BWRs. These features, including three independent RHR divisions, three EDGs, an ACIWA system, an alternate onsite CTG, and proper plant electrical and physical separation and layout, specifically address the more risk-significant operations during shutdown identified in NUREG-1449. GE provided appropriate safety guidance for effective outage planning and control and provided TS to ensure adequate systems are available to respond to events that may occur during shutdown. Implementation of these recommendations by a COL applicant will be reviewed by the staff.

The staff concludes, based on previous shutdown analyses (both international and U.S. operating reactors) and the ABWR Shutdown Risk Evaluation, that the chances of a core damage event occurring when in Modes 3, 4, or 5 is probably on the same order of magnitude as that of internal events occurring in Modes 1 and 2.

19.1.3.4.1 Dominant Accident Sequences

GE and the staff determined that it would not be useful to attempt to identify dominant accident sequences during Modes 3, 4, and 5. This is because, among all possible plant shutdowns and even on a minute-by-minute basis during any shutdown, the plant configuration could and probably would change. During each of these different configurations, one could conceivably have different dominant sequences and therefore different insights. Because of this, no dominant sequences based on the ABWR Shutdown Risk Evaluation were identified. NUREG-1449 identified that the majority of shutdown precursors and actual shutdown events were caused by human error. Operating, administrative, and emergency procedures as well as aspects of the ABWR design that minimize the chances of human error are discussed below.

19.1.3.4.2 PRA as a Design Tool

GE made no design changes specifically based on the ABWR shutdown risk evaluation. However, it did develop procedural guidance for outage planning and did develop for COL applicants a short list of acceptable system configurations, minimum combinations of systems to ensure adequate shutdown safety margins, when in Modes 3, 4, and 5. This guidance is located in SSAR Section 19.Q.7.9, Tables 19Q-3 through 19Q-5 and is repeated in Table 19.1-9 in this report. The tables were developed based on a single initiating event during shutdown (the failure of the operating DHR division), but are valid for initiating events during shutdown including loss of offsite power and the loss of the operating service water division.

19.1.3.4.3 Vulnerabilities

The staff believes that the results of the ABWR shutdown risk evaluation provide the information needed to decide if there are shutdown vulnerabilities and whether operation in modes other than full power represent a disproportionate risk in the ABWR. GE did not identify any shutdown vulnerabilities. With the use by a COL applicant of the tables (and accompanying or similarly acceptable methodology) that define sets of equipment that should remain operable for the ABWR to meet GE's proposed goal for conditional core damage frequency, $1E-5$ given the loss of a DHR train, when in Modes 3, 4, and 5, the staff finds that operation of the ABWR in Modes 3, 4, and 5 does not represent a disproportionate risk. In SSAR Appendix 19Q.9, GE has investigated whether the new features in the ABWR design might have introduced vulnerabilities when in modes other than full power. GE concluded that they did not introduce new vulnerabilities, and the staff concurs with GE's conclusion. The staff reviewed the ABWR shutdown evaluation and found no unreported shutdown vulnerabilities.

19.1.3.4.4 Plant Features and Operator Actions Important to Risk

A detailed list of ABWR features that are important to risk in Modes 3, 4, and 5 are provided in Tables 19Q.4-1 in the SSAR. The following list outlines those areas of the ABWR design that are important to maintaining risk during shutdown operations at a low level.

Features and Actions Minimizing Loss of DHR

- Having three divisions of RHR that are physically separated lowers the frequency of loss of DHR.

Severe Accidents

Table 19.1-9 Examples of configurations that meet 1E-5 CCDF goal in Modes 3, 4, and 5

Example Minimum Sets of Systems for Modes 3 and 4

	RHRB	Main Cond.	CUW	HPCFB	CRD	ADS	RHRB (CF)	Condensate	Fire Water
1	X				X	X	X		X
2	X		X			X	X		X
3	X					X	X	X	X
4	X	X				X	X		X
5	X			X	X	X		X	

Examples of Minimum Sets of Systems for Mode 5 (Unflooded) (2 to 3 Days After Shutdown)

	RHRB	HPCFB	CRD	Condensate	Fire Water	RHRB (CF)
1	X		X		X	X
2	X			X	X	
3	X	X	X	X		

Examples of Minimum Sets of Systems for Mode 5 (Flooded) (3 or More Days After Shutdown)

	RHRB	FPC	CUW	HPCFB	CRD	RHR (CF)	Condensate	Fire Water
1	X						X	X
2	X				X	X		
3	X			X	X		X	
4	X					X	X	X
5*			X		X			X
6*			X				X	X
7**		X			X		X	
8**		X					X	X
9**		X			X			X

* After 8 days

** After 10 days

Severe Accidents

- If all RHR loops were unavailable, (a) steam from the reactor could be directed to the main condenser (if available) and make-up could be supplied to the vessel by many sources discussed in Inventory Control below, or (b) the suppression pool could be used as a heat sink via the SRVs, or (c) the reactor water cleanup system could be a heat sink, or (d) the spent fuel pool inventory could be a heat sink if the reactor water level were raised to the refueling level, or (e) if the vessel head were removed, bulk boiling of reactor coolant in the vessel with adequate make-up would prevent fuel damage.
- There is no isolation of shutdown cooling on the loss of reactor protection system (RPS) logic power.

Features and Actions Minimizing Loss of Inventory

- The vessel level is displayed for the operator in the control room during all shutdown configurations, including refueling.
- Multiple sources of make-up exist such as the suppression pool, condensate storage tank, main condenser hotwell, and ac-independent water addition system.
- RHR system valves are interlocked with reactor system pressure to help ensure that RHR system low-pressure piping is not subject to full system pressure. However, even if it were exposed to full system pressure, the RHR piping should withstand full reactor pressure without rupture.
- Make-up can be provided by the CRD hydraulic system, the reactor water cleanup system, the condensate pumps in conjunction with the hotwell, the ACIWA system, and the RHR system.
- The mode selector switch automatically realigns the valves, as required, for the RHR mode selected. In the past, operator errors led to sending water to the wrong place. Now, all the operator has to do is change the mode switch to realign the system automatically.
- The shutdown cooling piping connects to the nozzle in the vessel above the level of the active fuel, so fuel cannot be uncovered by a siphoning effect.
- Suppression pool drain down has been identified as a period during which there is a diminished level of protection against core damage. It is important that a COL applicant properly coordinate suppression pool drain down with TS equipment configuration

requirements and the DHR core damage frequency goal (1E-5 conditional core damage frequency).

Features Minimizing Loss of Containment Integrity

- During shutdown with the drywell head removed, the ABWR has a secondary containment that will automatically be isolated on high radiation from a radiological boundary breach or fuel-handling accident. If the radiological accident is one that does not pressurize secondary containment, the filtering function of standby gas treatment system (SGTS) will be a benefit. If the radiological breach is caused by boiling in Mode 5, secondary containment will overpressurize and fail. In this circumstance, the SGTS filtering function will be lost; however, plate out via a tortuous path will tend to reduce consequences. If the core can continue to be covered by water, then boiling is a relatively benign event (with respect to radiological consequences). If, however, the core subsequently becomes uncovered for an extended period with the head off and secondary containment breached, there would be a massive radiological release. It takes approximately 3 hours to boil down the water above the core and longer to heat up the core. This should be an extremely low probability event since the ABWR design includes multiple means to provide water to the core in all modes of operation.

ABWR Features Minimizing Loss of Electrical Power

- There are three physically independent 1E diesel generators.
- There is a CTG that can be used to power any of the Class 1E or non-Class 1E buses. This generator can start a feedwater or other pump for DHR or inventory make-up if required.
- There are four divisions of dc power.

Other Design Features and Operator Actions Minimizing the Chances of Core Damage When Not at Power

- Fires and floods during shutdown can be mitigated by ensuring, through administrative procedures, that at least one safety division is not in maintenance and its physical boundaries remain intact. If it is decided to breach the boundaries of two safety divisions to complete maintenance tasks, an evaluation could indicate if the minimum set of systems capable of meeting the shutdown safety criteria would be available if a fire or flood were to occur. The analysis would indicate if a minimum set were available should a flood

Severe Accidents

or fire occur in the intact division or the breached divisions.

- The staff believes that the most important element in control of shutdown risk is adequate planning of maintenance on systems and support systems that can be used to remove decay heat or supply inventory make-up to the vessel. Maintenance planning is the responsibility of the COL applicant, but GE has provided guidance in its shutdown risk evaluation (SSAR Appendix 19Q.10) to help ensure that the planning will not place the plant in an unfavorable configuration from the standpoint of expected core damage frequency.

Appendix K of this report lists those PRA-based safety insights that were drawn from the PRA. The appendix documents the disposition of these insights and indicates if the insights are in ITAAC, Tier 2 information, TS, COL Action Items, Interface Items, or the RAP.

19.1.3.4.5 Insights Into Human Reliability and Important Human Actions

A total of eight human actions were identified as important for controlling risk during shutdown. Five actions were treated probabilistically in the evaluation, and three were treated deterministically (i.e., they were assumed to be taken if needed):

- Recognition of failure of an operating RHR system
- Initiation of standby RHR following loss of the operating division
- Use of non-safety grade equipment for DHR (e.g., reactor water cleanup (CUW) and main condenser)
- Use of non-safety grade equipment for inventory make-up (e.g., CRD, feedwater, condensate)
- Use of boiloff for DHR with the RPV head removed
- Implementation of fire or flood watches during periods of degraded safety equipment integrity
- Firefighting during shutdown operations, possibly with part of the fire protection system in maintenance
- Use of the remote shutdown panel during shutdown operation.

Because these actions and instrument requirements are considered significant, they are included in COL Action Items (See SSAR Section 19.9).

19.1.3.4.6 Combined License Applicant Action Items

COL Action Items based on the shutdown evaluation are identified in SSAR Appendix 19Q.12.3. These actions are contained in the detailed discussion of Chapter 19 COL Action Items given in SSAR Section 19.9.11. The staff finds these COL Action Items to be acceptable.

19.1.3.4.7 Insights From Uncertainty, Importance, and Sensitivity Analyses

The greatest uncertainty in a shutdown evaluation is the actual configuration of the plant. This is particularly true since there is no agreed-upon method of modeling plant configurations during shutdown. To bound this concern, the ABWR shutdown evaluation made a conservative assumption that all equipment not included in the minimum combinations of equipment needed to meet the goal of conditional core damage frequency was unavailable. GE and the staff believe that this conservatism would overshadow any uncertainty analysis or sensitivity analysis that might be performed. No importance analyses were performed since the importance of an SSC will vary depending on the particular configuration the plant is in.

19.1.3.5 Level 2 Analysis

19.1.3.5.1 Containment Performance and CCFP

The results of the Level 2 and 3 portions of the ABWR PRA indicate that the ABWR containment is quite robust and able to accommodate severe accidents with a low attendant probability of containment failure. In assessing the probability of containment failure, two alternative definitions of containment failure were considered: (1) loss of containment structural integrity, and (2) releases that result in doses of 25 rem or greater at a distance of 0.8 km (0.5 mile) from the reactor. The GE and staff estimates of CCFP are presented in Table 19.1-10. The staff estimates are based on a "staff-adjusted" ABWR risk profile that reflects staff views on selected issues. The most significant staff adjustment was to increase the frequency of early containment failures to account for uncertainty in the magnitude of direct containment heating loads and the contribution to CDF from unisolated LOCAs outside containment and ATWS events. Adjustments to account for uncertainties in source terms and consequence modelling were also made but affect only the dose definition of CCFP.

Table 19.1-10 GE's point estimates and the staff's mean estimates of the internal events risk

Performance Measure	GE Updated PRA		Staff-Adjusted Result	
	CFP	CCFP	CFP	CCFP
Structural Integrity	7.7E-10 ^{2/}	0.005 ^{2/}	4.1E-9 ^{1/}	0.026 ^{1/}
Dose Definition	3E-10	0.002	1.6E-8 ^{3/}	0.10 ^{3/}

Notes:

- ^{1/} Staff estimates reflect: (1) contribution from unisolated LOCAs outside containment and (2) increased probability of containment failure because of DCH.
- ^{2/} Based on GE's estimates of early and late containment failure frequency.
- ^{3/} Staff estimates include all frequency from Cases 7, 8, and 9, plus 58 percent of the frequency of Case 1 based on staff calculations using MACCS.

Both the GE and the staff estimates of the ABWR CCFP are within the Commission's containment performance goal (0.10) using either containment failure definition. Combined with the core damage frequency estimate from the Level 1 portion of the PRA (1.6E-7), this results in an extremely low likelihood of containment failure in absolute terms (i.e., on the order of 1E-8 to 1E-9 per year). GE's estimate of CCFP using the structural integrity definition is 0.005. The staff's estimate is 0.026, and is higher due to the treatment of LOCAs outside containment as containment failures, and the higher loads associated with DCH in the staff's assessment. GE's estimate of CCFP based on the dose definition is 0.002. In contrast, the staff's estimate of CCFP using the dose definition of containment failure is about 0.1. The staff's estimate is higher because of primarily (1) a higher source term used by the staff to represent scenarios with vented containment and suppression pool scrubbing, and (2) the use of the Melcor Accident Consequence Code System (MACCS) code for the consequence in calculations. However, the staff concludes that CCFP does not exceed 0.1, even under conservative staff assumptions, and therefore, meets the Commission's safety goal.

19.1.3.5.2 Leading Sequences for Containment Failure

The breakdown of contributors to containment failure for internal events is presented in Figure 19.1-2 in the form of

a pie chart. For internal events, the bulk (about 85 percent) of the core melt events in the ABWR are successfully contained, with the releases to the environment limited to leakage on the order of the design-basis containment leak rate. A small fraction (about 13 percent) of core damage events result in actuation of the COPS and releases to the environment via the stack. While this is the dominant release mode for the ABWR, the consequences of these releases are significantly reduced because of the relatively late time of the release (about 20 hours from accident initiation), and fission product removal by the suppression pool. As a result, these releases do not dominate the ABWR risk profile.

Based on the staff assessment, only about 3 percent of core melt accidents result in containment failure. The bulk of these failures (about 90 percent of the 3 percent) are classified as early failures, relative to the time of core melt. Major contributors are DCH, LOCAs outside containment that proceed to core melt, and overpressure as a result of ATWS (The DCH frequency is based on a GE sensitivity study, as discussed in FSER Section 19.1.3.5.4; the LOCA and ATWS frequencies are based on GE's baseline Level I PRA). Late failures constitute only about 10 percent of the containment failure frequency. Drywell head or penetration overtemperature and rupture disk failure (i.e., failure of the drywell before rupture disk actuation) are the major contributors.

Figure 19.1-2 Breakdown of ABWR containment release frequency from CET analysis

* Based on GE results before Amendment 32. GE values for unisolated LOCAs outside containment reported in Amendment 32 are about a factor of 10 lower, however this does not affect the overall staff findings.

19.1.3.5.3 Important Design Features for Containment Performance

GE added a passive floodor system to the lower drywell to improve the chances that a core-on-the-floor (following a postulated reactor vessel failure) would be covered by water. This is designed to help quench the corium and reduce the drywell temperature and pressure from noncondensable gas generation.

GE added a COPS to protect the integrity of the containment (i.e., to assure that fission products must pass through the pool rather than bypass it by preventing failure of the drywell head) in slow containment overpressure events. The COPS allows isolation of the containment after the opening of the COPS rupture disks.

Suppression Pool — The suppression pool is an important containment feature to prevent severe accident progression and to promote fission product removal, since releases from the reactor vessel are either directly routed to the pool (e.g., transients with actuation of ADS) or pass through the pool via the drywell-wetwell connecting vents. However, the suppression pool function can be compromised in the ABWR design by a single failure of a wetwell or drywell vacuum breaker (i.e., a stuck open vacuum breaker) or by excessive leakage of one or more vacuum breakers. While a similar containment challenge exists for all operating BWRs, the frequency of suppression pool bypass is expected to be somewhat greater for the ABWR since the ABWR has only a single vacuum breaker in each wetwell-drywell connecting path, in contrast to two valves in series in each path in operating BWRs. The impact of suppression pool bypass is minimized in the ABWR by assuring: (1) a low probability of vacuum breaker leakage and failure through periodic surveillance, and (2) availability of drywell or wetwell sprays to condense steam that bypasses the suppression pool.

Vacuum Breaker Position Indication — Each vacuum breaker will be equipped with a position indication switch that will be sensitive enough to detect the allowable suppression pool bypass capability of the containment. Vacuum breaker position indication and associated alarms will be provided in the main control room. This reduces the potential for suppression pool bypass by assuring that the plant is not operated with a stuck open vacuum breaker, and that preexisting leakage paths will be limited to small flow areas.

Drywell Sprays — The drywell spray system is critical for mitigating the consequences of severe accidents in the ABWR. The drywell spray system serves to: (1) reduce containment overpressure and delay the time to actuation

of COPS, (2) eliminate the potential for drywell failure resulting from overheating in those events in which debris may be dispersed to the upper drywell, and (3) mitigate the consequences of suppression pool bypass by condensing steam produced in the drywell. The ACIWA mode of RHR as a backup source of water to the sprays is perhaps the single-most important feature for reducing the consequences of severe accidents in the ABWR.

Lower Drywell Design — The design of the ABWR lower drywell or reactor cavity is such that there is a low probability that the cavity will be flooded at the time of reactor vessel failure, but a high probability that the cavity will be flooded after vessel failure. A dry cavity at the time of vessel failure reduces the potential for large ex-vessel steam explosions, whereas the subsequent flooding of the cavity helps minimize the impact of core concrete interactions (CCIs). The following ABWR design features are important to assuring a dry cavity at the time of vessel failure: (1) lack of any direct pathways by which water from the upper drywell (e.g., from drywell sprays) can drain to the lower drywell, other than by overflow of the suppression pool, (2) negligible probability of premature or spurious actuation of the passive floodor valves at temperatures lower than 260 °C (500 °F) or under differential pressures associated with reactor blowdown and pool hydrodynamic loads, and (3) a capability to accommodate approximately 7.2E5 kg of water in the suppression pool from external sources before the pool overflows into the lower drywell. Additional design features are included to increase the chances of a flooding a cavity following vessel failure, as discussed below.

Lower Drywell Floodor (LDF) System — The LDF system in the ABWR provides a passive means of adding water to the lower drywell following reactor vessel breach. This water would cover the core debris, thereby enhancing debris coolability, cooling the drywell, and providing fission product scrubbing. The passive floodor system is a backup to other means of lower drywell water addition in the ABWR, including: (1) continued water addition through the breached reactor vessel, (2) suppression pool overflow as a result of water addition from water sources outside containment, and (3) ingress of suppression pool water after the core debris has penetrated the wetwell-drywell connecting vents (DCVs). PRA-based sensitivity studies indicate that the incremental risk reduction offered by the passive floodor system is minimal. This is because of credit taken in the ABWR for continued water addition using the ACIWA mode of RHR.

Reactor Pedestal and Drywell Floor — The effect of CCIs is minimized in the ABWR by the use of a robust reactor pedestal and the use of basaltic concrete in the floor of the lower drywell. The reactor pedestal is 1.7 m (5.6 ft) thick

Severe Accidents

and capable of withstanding 1.55 m (5 ft) of core-concrete erosion without loss of structural integrity. The use of basaltic concrete in the floor minimizes the production of noncondensable gases, thereby delaying the time of COPS actuation.

Containment Ultimate Pressure Capacity — The ultimate pressure capacity of the ABWR containment is limited by the drywell head, the failure mode of which is plastic yield of the torispherical dome. After the original SSAR submittal, GE increased the pressure capability of the drywell head from 100 pounds per square inch (psig) to 134 psig, and increased the COPS setpoint from the original value of 80 psig to the final value of 90 psig. The strengthening of the drywell head increases the ability of the containment to withstand rapid pressurization events, such as DCH, without loss of structural integrity, and provides additional margin between the COPS setpoint and the drywell failure pressure, thereby reducing the potential for drywell failure before COPS actuation.

COPS — COPS is part of the atmospheric control system in the ABWR and consists of a pair of rupture disks installed in a 10-in. diameter line, which connects the wetwell airspace to the stack. COPS provides for a scrubbed release path in the event that containment pressure cannot be maintained below 90 psig. Without this system, late containment overpressure failures would be expected to occur in the drywell, resulting in unscrubbed releases. COPS provides a significant benefit by reducing the source terms for late releases, and minimizing the potential for containment-failure-induced loss of core cooling (e.g., in Class II sequences).

Containment Inerting — The ABWR containment will be made inert during power operation. As a result, the threat of containment failure as a result of hydrogen combustion is essentially eliminated for power operation.

Containment Sump Protection — To preclude significant debris from entering the containment sump following a severe accident, a protective barrier was added around the sumps to prevent the entrance of molten debris, while allowing water to enter the sumps during normal operations.

19.1.3.5.4 Impact of Severe Accident Phenomena on Containment Performance

DCH occurs only in sequences with reactor vessel failure at high pressure. For the ABWR, reactor vessel failures at high pressure constitute about 30 percent of the reactor vessel failures (about $4.3\text{E-}8$ per year). In view of the large uncertainties inherent in estimating the pressure loads associated with DCH, the staff conservatively based its

findings on the results of a GE sensitivity analysis for DCH which reflected an increased contribution to pressurization due to higher baseline containment pressure, and combustion/recombination of hydrogen with residual oxygen. As such, DCH is the principal contributor to containment failure for the ABWR. The staff has estimated the containment failure probability for DCH to be about 5 percent, conditional upon reactor vessel failure at high pressure. This results in a very low frequency of containment failure from DCH ($2.3\text{E-}9$ per year). The low frequency of reactor vessel failure at high pressure is a result of the highly reliable depressurization system. There are no specific ABWR containment design features to deal with DCH loads other than the general arrangement of the drywell, wetwell, and connecting vents, which provides for a series of 90-degree bends that debris must traverse in order to reach the upper drywell.

Fuel-coolant interactions (FCI) or steam explosions are considered negligible in the ABWR design because of the very low probability that the lower drywell will be flooded at the time of reactor vessel failure (0.3 percent). In addition, the ABWR reactor pedestal is capable of withstanding the best-estimate loads associated with an ex-vessel steam explosion as predicted by GE and staff calculations. The design features that contribute to the low probability of a flooded lower drywell at the time of vessel failure were discussed above in the context of the lower drywell design. The structural capability of the reactor pedestal is discussed further in Section 19.2 of this report.

CCIs have a minimal impact on ABWR containment structural integrity because of the inclusion of: (1) a thick reactor pedestal, (2) the use of basaltic concrete in the floor of the lower drywell, and (3) a sump shield to prevent core debris from entering the lower drywell sump. These features provide significant confidence in reactor pedestal and containment integrity for well beyond 24 hours following reactor vessel failure and render CCI-induced containment failure a relatively insignificant contributor to risk.

Hydrogen combustion is not an important containment challenge in the ABWR since the atmosphere is made inert during normal operation.

19.1.3.6 Level 3 Analysis Insights

19.1.3.6.1 Risk to the Public

Based on the Level 3 PRA, the estimated total risk to the public from the ABWR is extremely small. GE's analysis indicates a total dose of about 0.2 person-rem over the 60-year period. The staff's estimate is about 1 person-rem. The difference is largely a result of the contribution

from unisolated LOCAs outside containment, and an increased probability of early containment failure from DCH. It should be noted that while vented scenarios are the dominant contributor to the staff's estimate of containment failure using the dose definition of containment failure, these sequences do not contribute significantly to total risk when measured in terms of person-rem exposure. Rather, total risk is dominated by events that lead to early containment structural failure and containment bypass. This is consistent with results from PRAs for operating plants.

19.1.3.6.2 Leading Contributors to Risk

Despite their small contribution to total core damage frequency (0.8 percent based on the staff's assessment), unisolated LOCAs outside containment dominate the ABWR risk profile. The frequency of these sequences is extremely low in absolute terms ($1.2\text{E-}9$ per year). In the staff's assessment, these sequences dominate risk because of the severe releases associated with complete bypass of containment. These sequences also dominate risk because of the fact that the baseline core damage frequency and the contribution of more familiar sequences and containment challenges that dominate PRAs for operating plants have been reduced in the ABWR design. The fact that these sequences dominate risk is, in a way, a reflection of the low estimated risk of the ABWR.

Other early containment failures (DCH and ATWS) are the second most dominant contributor to risk for the ABWR. Although the frequency of these early failure mechanisms ($2.5\text{E-}9$) is higher than unisolated LOCAs outside containment, fission product releases are reduced somewhat because of holdup in containment.

It should be noted that the reported frequency of unisolated LOCAs reported in Amendment 32 of the SSAR is about a factor of 10 lower than the above value on which the staff based its finding. This would result in the total risk's contribution, as well as the contribution of bypass scenarios to risk, being lower than described above.

19.1.3.7 PRA-Based Input to the Certified Design

19.1.3.7.1 Reliability Assurance Program

The ABWR is the lead plant for development of a RAP for advanced reactors as required by 10 CFR Part 52. GE made a particularly strong effort in identifying important SSC for inclusion in RAP based on insights from the ABWR PRA. In Appendix 19K of the SSAR, GE has listed the reliability and maintenance actions that it believes should be considered throughout the life of the plant so that the PRA remains an adequate basis for quantifying

plant safety and determining safety insights. It is anticipated that a COL applicant will make these insights (given in Table 19K.11-1 in the SSAR) the cornerstone of its programs (DRAP and ORAP).

Internal and external event sequences, both for Levels 1 and 2, were considered in drawing up the list of reliability and maintenance actions. In developing the list, GE performed a systematic search that involved both quantitative and qualitative considerations. For Level 1 internal events, the key considerations were the results of risk achievement worth and Fussell-Vesely importance measures. These results are reported in Tables 19K.3-1 and 19K.3-2 in the SSAR and Appendix K in this report. From risk achievement worth measures, one gets a list of SSCs where maintenance or testing resources should be focused to help assure high levels of availability. A Fussell-Vesely importance measure answers the question, "For which SSCs would improvement of current unavailabilities be most beneficial in order to best lower the estimated core damage frequency?"

19.1.3.7.2 Tiers 1 and 2 Information or Requirements

In its review, the staff believed that it was important to systematically search a PRA for safety insights. As part of this search for insights, GE identified important safety insights that need to be passed on to a COL applicant. Appendix K of this report provides a cross-reference between the PRA insights and the ITAAC, Tier 1 design descriptions, and Tier 2 material in the DCD. Appendix K explains the disposition of these ABWR PRA safety insights. The staff has reviewed the COL action items proposed by GE and finds them to be appropriate dispositions of these insights, and finds they are acceptable.

19.1.3.8 PRA Insight Conclusions and Safety Findings

This section documents the overall conclusions about the insights drawn from the ABWR PRA about the ABWR design.

19.1.3.8.1 Vulnerabilities

In its performance of the ABWR PRA, GE did not specifically identify any design features or procedures that constituted a vulnerability to severe accidents. GE did search for and identify many cost-effective design and procedure improvements that it included in the ABWR design. Details of the design improvements motivated by the ABWR PRA are discussed previously in Section 19.1.3 and listed in Table 19.1-4. The search by the staff and GE for design and procedure vulnerabilities (or areas of

Severe Accidents

potential improvement) included internal and external events (seismic, fire, and internal floods) as well as events in Modes 3, 4, and 5.

19.1.3.8.2 Balance of Prevention and Mitigation

The staff examined the ABWR design to determine if the design has an inappropriate reliance on either prevention of severe accidents or mitigation of severe accidents to the exclusion or detriment of the other. The staff finds the balance to be acceptable for the following reasons:

In the area of prevention of core damage, GE has reported that the ABWR PRA estimates internal event core damage frequency to be in the range of $1\text{E-}7$ per year. This estimate reflects the design modifications incorporated into the ABWR design that have eliminated or reduced most of the important sequences that have led to core damage in other BWR PRAs. The NRC does not endorse the absolute value of the core damage frequency reported by GE, but does endorse the insights drawn from the PRA by GE and the NRC, which are based on relative and other considerations. For the external events seismic, fire, and internal floods, the ABWR design is an improvement over those of existing operating plants. The seismic design is built to an SSE of 0.3 g while most sites in the United States east of the Rocky Mountains would have lower SSEs when determined on a site-specific basis. In addition, the ABWR PRA-based seismic margins analysis demonstrated that the design has significant margin to earthquakes well above the SSE. For fires and floods, the ABWR design has three physically separated safety divisions that limit the chances of a fire or flood affecting more than one division. This capability is a design enhancement for advanced reactors. For mitigation of a severe accident in which core damage has occurred, the ability of the containment design to withstand severe accidents is at least as robust as that of operating BWRs today and has some additional features not in current designs. Additional features include the ac-independent water addition system, an increased drywell head ultimate pressure capacity, and the passive flooders system.

19.1.3.8.3 Comparison With NRC and EPRI Safety Goals

In the DSER (SECY-91-309), the staff compared the integrated risk results for the ABWR with the Commission's quantitative health objectives and safety goals, NRC and EPRI requirements or goals for advanced light-water reactors (ALWRs), and GE's design goal. GE and staff estimates of various risk measures based on the original GE analysis (Amendment 8 to the SSAR), are reported in Figures 19.1-1 and 19.1-2 of the DSER. Based on these comparisons, the staff reached tentative

conclusions on how the various objectives and goals were met. After the DSER, GE modified the ABWR design and submitted the results of the updated ABWR PRA (Levels 1, 2, and 3) reflecting modifications to the plant design, as well as modeling enhancements and corrections identified by GE and staff since the original PRA. The staff committed itself to reporting the results of its review of the integrated risk estimates in this report.

The staff's evaluation of GE's updated PRA analyses is presented in the preceding portions of Section 19.1 of this report. Based on a review of the updated PRA analysis, the staff performed a limited update of the comparisons with the safety goals. The updated comparisons with safety goals are not as exhaustive as the one provided in the DSER since: (1) the staff did not attempt to requantify uncertainties in core damage frequency and containment performance as part of the final evaluation, and (2) only doses at the site boundary were recalculated as part of the final evaluation, rather than the full set of offsite consequence measures. However, for certain performance measures, such as individual risk of early fatality, the original comparisons indicated significant margins between the PRA results and the health objectives or safety goals, even when the uncertainty associated with various phenomenological issues and other issues raised in the staff review were factored into the assessment. In those cases, the staff's tentative conclusions presented in the DSER are still considered valid for the updated ABWR PRA and have been used as the basis for the staff's final conclusions. Limited reliance on the original analyses is considered acceptable on the basis that the updated core damage frequency and containment release characteristics are not significantly different from those on which this original assessment was based. Furthermore, the analyses and plant modifications performed after Amendment 8 to the SSAR did not identify any plant vulnerabilities and are expected to further reduce risk.

The comparison of the ABWR PRA results with the various health objectives and safety goals follows. As discussed in Section 19.1.1 of this report, the staff does not believe that it is appropriate, or possible, to directly compare the results of the updated ABWR external events analyses with the safety goals because in the updated PRA these events were analyzed using either a margins approach (seismic) or bounding analyses (internal floods and fire). Accordingly, the comparisons with safety goals are limited to internal events.

Individual risk of early fatality — Goal: $< 5\text{E-}7$

The staff concluded in the DSER that the ABWR meets the Commission's quantitative health objective for individual risk of early fatality by a wide margin. The calculated

mean estimate, as well as the upper bound of the uncertainty distribution (95th percentile), were orders of magnitude below the goal. Because the updated core damage frequency and containment release characteristics are not significantly different from those on which this original assessment was based, the staff concludes that the final ABWR design would also meet this health objective by a wide margin, even if the effects of unisolated LOCAs outside containment are taken into account. The staff further believes that the remaining margin provides reasonable assurance that consideration of the impact of those portions of the analysis for which full PRA quantification was not performed (e.g., seismic events) will not result in the mean value's exceeding the goal.

Individual risk of cancer fatality — Goal: $< 2E-6$

The staff concluded in the DSER that the ABWR meets the Commission's quantitative health objective for individual risk of cancer fatality by a wide margin. The calculated mean estimate, as well as the upper bound of the uncertainty distribution, were several orders of magnitude below the goal. Because the updated ABWR core damage frequency and containment release characteristics are not significantly different from those on which this original assessment was based, even if the effects of unisolated LOCAs outside containment are taken into account, the staff concludes that the final ABWR design would also meet this health objective by a wide margin. The staff further believes that the significant margin between the calculated risk and this goal provides reasonable assurance that consideration of the impact of those portions of the analysis for which full PRA quantification was not performed will not result in the mean value's exceeding the goal.

Core damage frequency — Commission Goal: $< 1E-4$
EPRI Goal: $< 1E-5$

The staff concluded in the DSER that the ABWR meets both the Commission and the EPRI goals for internal event core damage frequency by a wide margin. The calculated mean value was below the goal, with the upper bound of the uncertainty distribution about 1 order of magnitude below the Commission goal. Based on the updated PRA and a supporting Level 1 uncertainty analysis, GE has estimated the ABWR mean core damage frequency at $1.6E-7$ per year, with an upper bound (95th percentile) value of $4.5E-7$ per year. Based on review of the ABWR PRA, the staff concludes that the final ABWR design meets the Commission's goal for internal events, even if uncertainties in the calculated core damage frequency are taken into account. The analyses performed by GE for ABWR external and shutdown events do not lend themselves to comparison with the Commission's core

damage frequency goals, based on methods and assumptions used. The staff determined that this is acceptable, since the GE analyses of external events and shutdown events included an appropriate search for vulnerabilities.

Probability of large release (one or more early fatalities) — Goal: $< 1E-6$

The staff concluded in the DSER that the ABWR meets the Commission's goal for the probability of large release by a considerable margin, recognizing that the definition of "large" was still under Commission consideration. The calculated mean estimate was well below the goal, with the upper bound of the uncertainty distribution (95th percentile) about 2 orders of magnitude below the goal. Because the updated core damage frequency and containment release characteristics are not significantly different from those on which this original assessment was based, the staff concludes that the final ABWR design would also meet this objective by a wide margin. The staff notes that this goal can be met as a result of the very low calculated core damage frequency for internal events, even without taking credit for the containment. Compliance with the Commission's containment performance goal provides added assurance that the probability of a large release will remain below the goal.

Conditional containment failure probability (structural integrity definition) — Goal: < 0.1

The staff concluded in the DSER that the ABWR did not strictly meet the Commission's goal for CCFP using the structural integrity definition of containment failure. The staff's uncertainty distribution bridged the goal with a median value slightly above the goal, suggesting that the bulk of the distribution was slightly above the goal.

GE's updated point estimate of CCFP based on structural integrity definition is 0.005. The staff adjusted this value to account for: (1) uncertainties in DCH, and (2) treatment of unisolated LOCAs outside containment. These adjustments increased CCFP to about 0.026, which is still below the Commission goal. However, the staff did not attempt to requantify the uncertainties associated with this estimate. It believes that if these uncertainties were requantified reflecting the final ABWR design and additional severe accident analyses completed since the DSER (e.g., increased containment pressure capacity and additional information supporting a low probability of a flooded reactor cavity at the time of vessel failure), the resulting CCFP estimate and associated uncertainties would be somewhat lower than reported in the DSER, but would still be significant.

Severe Accidents

Conditional containment failure probability (dose definition) — Goal: < 0.1

The staff concluded in the DSER that the ABWR did not strictly meet the Commission's goal for CCFP using GE's definition of containment failure (dose). The staff's uncertainty distribution bridged the goal, with a mean estimate above the goal and median estimate slightly below the goal. This indicated that about half of the distribution is below the goal.

GE's updated point estimate for CCFP (based on the dose definition) is 0.002. The CCFP based on the dose definition is not significantly different than the CCFP based on the structural integrity definition (where containment venting is not considered as a containment failure). The reason for this is that in GE's analysis, doses in excess of 25 rem at 0.8 km (0.5 miles) primarily occur only when structural integrity is breached. In contrast, the staff's consequence calculations indicate a high probability (about 60 percent) of exceeding 25 rem at the boundary for vented scenarios. Thus, the staff's estimate of containment failure frequency using the dose definition of containment failure includes essentially all of the frequency of events leading to loss of structural integrity, plus about 60 percent of the frequency of sequences with COPS actuation. This results in a staff estimate of CCFP of about 0.10, which still meets the Commission's safety goal.

The staff did not attempt to requantify the uncertainties associated with this estimate. It expects that if these uncertainties were requantified reflecting the final ABWR design, additional severe accident analyses completed since the DSER, and more deterministic estimates of source terms for vented scenarios, the resulting CCFP estimate and associated uncertainties would be lower than reported in the DSER. Nevertheless, the uncertainties would still be significant and would extend above the goal. Although the uncertainty bands are different, the staff and GE both concludes that the mean values of their risk estimates meet the Commission's safety goal.

19.1.3.8.4 Comparison With Operating BWRs

The staff evaluated the ABWR design and its improved or unique features to prevent or mitigate severe accidents such as the three full (high and low pressure) ECCS divisions, the COPS, and the passive flooders. The staff evaluated the ABWR PRA and concluded that the absolute value of the estimated core damage frequency for the ABWR is lower than those of operating BWRs in the United States. Based on these reviews, the staff finds that the ABWR design represents an improvement in safety over operating BWRs in the United States.

19.1.3.8.5 ABWR Design Acceptability

Based on the staff's review of the ABWR PRA and ABWR design as set forth in this section (19.1) of this report, the staff finds that the ABWR design and the submittals made for the ABWR in the SSAR meet the intent of the Commission's Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants, dated August 8, 1985, the requirement of 10 CFR 50.34(f)(1)(i) to perform a plant-specific PRA that seeks improvement in the reliability of core and containment heat removal systems, the staff's proposed applicable regulation for analysis of external events for the ABWR PRA, and the requirement of 10 CFR 52.47(a)(1)(v) for an evolutionary plant design vendor to submit a PRA.

19.1.3.9 ACRS Concerns Related to the ABWR PRA

19.1.3.9.1 Adequacy of the ABWR PRA

The Advisory Committee on Reactor Safeguards (ACRS) asked the staff to explain how it intends to use the ABWR PRA in the design certification process and when PRA guidance will be provided.

In preparing for design certification, the staff and GE used the ABWR PRA in a number of traditional as well as new areas. The staff expects that the PRA will continue to prove to be useful throughout the entire process — from design certification through the end of the life of an operating ABWR. Examples of traditional areas in which the ABWR PRA was used include estimating core damage frequencies and risk significance and identifying design vulnerabilities. Innovative ways in which the PRA has been used include the following:

- helping to identify systems and components to be included in the RAP, increasing the safety of the design of the ABWR by examining design options
- helping to identify human errors that need to be considered when designing an advanced control room
- identifying areas that should receive special attention under ITAAC
- helping to determine the balance of prevention and mitigation capabilities
- providing a structure for helping to determine procedural and TS needs during modes other than full power.

19.1.3.9.2 Adequacy of GE's Treatment of the Reactor Water Cleanup System

The ACRS performed a review, independent from the staff, of the reactor water cleanup (RWCU) system. GE's treatment of this system in the PRA was not reviewed in detail as part of the staff's initial review. Several of the questions raised by the ACRS review applied to the ABWR PRA. First, the ACRS review concluded that the ABWR PRA did not evaluate RWCU system line breaks as initiators. Second, the ACRS review concluded that GE erroneously took credit for the RWCU system as a potential heat removal path at high pressure during non-LOCA transients. The ACRS found credit to be inappropriate, in part, because the RWCU system was designed to isolate on high temperature (to protect resin beds) and because even if the resin beds were bypassed, the outlet temperatures of the heat exchanger piping would have exceeded the design limits of the piping.

The ACRS correctly noted that GE did not treat RWCU system line breaks, or other LOCAs outside the primary containment, as initiating events in the PRA. Rather, the approach taken by GE (SSAR Section 19E.2.3.3) was to show that the risks associated with such events constitute a small fraction of the risk from all non-bypass paths, and that these ex-containment LOCAs therefore need not be included in the PRA. The ACRS questioned the validity of this analysis, and the fact that it failed to consider that the RWCU system line break could be the initiator for the core damage event. Based on concerns identified by the ACRS, the staff performed a more detailed review of GE's treatment of the RWCU system in the PRA. After further review of GE's treatment of ex-containment LOCAs, the staff identified additional concerns related to flow area assumptions on which the split fractions were based and isolation valve failure probability assumptions in the analysis (i.e., whether the assigned values adequately account for environmental effects and common cause failure).

In response to ACRS and staff concerns, GE revised its suppression pool bypass analysis and provided additional clarification regarding the effects of a break in the RWCU suction line. GE indicated that the system arrangement and emergency procedure guidelines (EPGs) provide assurance that unisolated breaks in the RWCU suction line will not result in core uncover and long-term releases. The staff required that GE develop a COL Action Item that would have a COL applicant develop postaccident recovery procedures for an unisolated CUW line break.

As part of its review, the staff required GE to provide an analysis of LOCAs outside containment that was based on

event trees and fault trees rather than on split fraction arguments.

The ACRS correctly stated that GE erroneously took credit for the RWCU system at high pressure during transients. GE has corrected this design deficiency by redesigning the isolation logic of the RWCU system, realigning the isolation configuration so that only the heat-vulnerable resin beds are isolated on high temperature, and limiting the total isolation of the RWCU to those periods when the containment isolation function is actuated. In addition, the RWCU will only be put into operation by emergency procedure after the RHR fails. Thereafter, cooling water will be diverted by procedure from the RHR heat exchangers to the RWCU heat exchanger to limit the temperature increase across the RWCU heat exchanger. GE calculates that this temperature increase is only a few degrees above the design temperature and argues that this is acceptable because the RWCU is a backup system that only will have to be used in this configuration for very low probability, beyond-design-basis events.

19.1.4 Evaluation of the Quality of the ABWR Probabilistic Risk Assessment

The staff has completed its review of the quality of the ABWR PRA. It finds that the ABWR PRA is of sufficient quality that, at a minimum, it can be used in the following ways:

- (1) to assess (within the limits of PRA methods and uncertainties) the risks associated with the ABWR design
- (2) to identify strengths and weaknesses of ABWR design features
- (3) to evaluate ABWR containment failure probabilities for early and late failure modes
- (4) to compare the ABWR risk results with the Commission's safety goal and the "safety margin basis design requirements" provided in the EPRI ALWR Requirements Document within the limitations of the latest technology in risk assessment
- (5) to provide an integrated perspective of the overall risk estimates for the design
- (6) to identify major contributors to uncertainty in estimated core damage frequency.

Severe Accidents

The staff concludes that the quality of the ABWR PRA is adequate for its intended functions such as supporting and improving the ABWR design process, providing relative importance of sequences (as well as identifying important SSC) leading to core damage or containment failure, and searching for design and procedure vulnerabilities that could be eliminated on a cost-effective basis.

19.1.5 Open Item Closure

All Open and "Outstanding" Items identified in Chapter 19 of the staff's ABWR DSER and DFSE are resolved satisfactorily. The staff is issuing a letter separate from this report documenting the resolution of these issues. All important issues related to containment performance are discussed in Section 19.2.

19.2 Severe Accident Performance

19.2.1 Introduction

The purpose of this section is (1) to describe the NRC's approach to resolution of severe accident issues for evolutionary light water reactors as specified in SECY-90-016, SECY-91-262, SECY-93-087, and the corresponding SRMs and (2) to evaluate the approach proposed by GE for resolution of severe accident issues for the ABWR design.

To provide adequate protection of the public health and safety, current NRC regulations require conservatism in design, construction, testing, operation, and maintenance of nuclear power plants. A defense-in-depth approach has been mandated in order to prevent accidents from happening and, if accidents should occur, to mitigate their consequences. Siting of nuclear power plants in less populated areas is emphasized. Furthermore, the NRC, State, and local governments mandate emergency response capabilities that provide additional defense-in-depth protection to the surrounding population.

The reactor and containment systems design provides a vital link in the defense-in-depth philosophy. Current reactors and containments are designed to withstand a LOCA and to meet the siting criteria of 10 CFR Part 100 and General Design Criteria (GDC) of 10 CFR Part 50, Appendix A. The large-break LOCA and other accidents analyzed in accordance with the NRC's Standard Review Plan (NUREG-0800) and documented in Chapter 15 of the SSAR are commonly referred to as "design-basis accidents" (DBAs) for nuclear power plants. This high level of confidence in a defense-in-depth approach results, in part, from stringent requirements for meeting single-failure criterion, redundancy, diversity, quality assurance, and utilization of conservative models.

The NRC also has requirements to mitigate adverse conditions associated with transients or events considered outside the design basis such as ATWS (10 CFR 50.62), station blackout (SBO) (10 CFR 50.63), and combustible gas control (10 CFR 50.44); however, a definitive set of regulatory requirements for addressing specific severe accident phenomenon does not exist. Existing regulations that require conservative analyses and inclusion of mitigative features for design-basis events, provide margin for severe accident challenges. In addition, the staff, in keeping with the Commission's Policy Statement on Severe Accidents that future designs for nuclear power plants achieve a higher standard of severe accident safety performance, concluded that severe accidents should be considered in the design of future nuclear power plants.

In an SRM, dated January 28, 1992, on SECY-91-262, the Commission approved the staff's recommendation to proceed with design-specific rulemakings through individual design certifications to resolve selected technical and severe accident issues. The effect of these actions on the ABWR is that the criteria specified for resolution of severe accident issues in SECY-90-016 and SECY-93-087 will be incorporated into the ABWR design certification rulemaking as applicable regulations. The following discussion describes the criteria that were used for the deterministic evaluation of severe accident issues.

19.2.2 Deterministic Assessment of Severe Accident Prevention

19.2.2.1 Severe Accident Preventative Features

Accident initiators can be separated into two general groups: transients and LOCAs. Transients include planned reactor shutdowns and transients that result in reactor scrams. Examples of transients include manual shutdown, main steam isolation valve closure, loss of condenser vacuum, loss of feedwater, nonisolation event (trip with bypass), inadvertent open relief valve, and loss of offsite power. LOCAs generally fall within three categories, small, medium, and large, based on the size of the line break.

Following the accident initiator, plant systems respond to control reactivity, reactor pressure, reactor water level, and containment parameters within the design basis spectrum. Ensuring sufficient heat removal from the core to prevent overheating and subsequent fuel damage is of paramount importance. Failure to provide this heat removal can result in fuel overheating and the potential for oxidation and melting of the reactor core.

Transient-induced accidents are usually accompanied by actuation of the safety or relief valves that transfer steam

from the reactor directly to the suppression pool, thus limiting an increase in containment pressure. LOCAs blow down into the drywell atmosphere until the suppression pool horizontal vents are uncovered, resulting in containment pressurization lower than the containment design pressure. In the event that the steam from the reactor bypasses the suppression pool and travels directly to the wetwell atmosphere, significant pressurization of the containment may result. The issue of suppression pool bypass is addressed in Sections 6.2.1.8 of this report for design-basis accident (DBA) bypass and 19.2.3.3.5 for severe accident bypass.

In response to accident initiators identified through operating reactor experience and analyses of the results of probabilistic safety assessment, the staff developed criteria for evolutionary light-water reactors (LWRs) to prevent the occurrence of such initiators from leading to a severe accident. These criteria were specified in SECY-90-016 and SECY-93-087 and include design provisions for anticipated transient without scram (ATWS), SBO, fires, and intersystem LOCAs.

19.2.2.1.1 Anticipated Transient Without Scram

An ATWS is an anticipated operational occurrence followed by the failure of the trip portion of the reactor protection system (RPS). Anticipated operational occurrences (transients) are those conditions of normal operation that are expected to occur one or more times during the life of the nuclear power plant and include, but are not limited to, loss of power to all recirculation pumps, tripping of the turbine generator set, isolation of the main condenser, and loss of all offsite power. Dependent upon the transient and its severity, the plant may recover and continue normal operation or the plant may automatically shut down (scram) via the RPS. The RPS is designed to safely shutdown the reactor to prevent core damage.

These transients when coupled with a failure of the RPS may lead to conditions beyond the design basis of the plant. In these cases, the reactor must be manually scrammed in order to avoid reactor fuel damage or coolant system damage. Subsequent failure of the manual scram system and inadequate core cooling may lead to core damage.

Transients with the greatest potential for significant damage to the reactor core and containment are those leading to an increase in reactor pressure and temperature, a loss of heat sink, or a failure of the RPS to scram the reactor. During an ATWS event, reactor power, pressure, and temperature must be controlled or the potential exists for a severe accident.

The ATWS rule (10 CFR 50.62) was promulgated to reduce the probability of an ATWS event and to enhance mitigation capability if such an event occurred. For BWRs, the ATWS rule specifies inclusion of an alternate rod insertion system, an SLCS, and equipment to trip the reactor recirculation pumps. In Section 15.5 of this report, the NRC concluded that the ABWR complies with the ATWS rule.

19.2.2.1.1.1 Preventive and/or Mitigative Features

In SECY-90-016, the staff recommended that the Commission approve its position that diverse scram systems should be provided for evolutionary LWRs. In its June 26, 1990 SRM, the Commission approved the staff's position, but directed that if the applicant can demonstrate that the consequences of an ATWS are acceptable, the staff should accept the demonstration as an alternative to the diverse scram system.

The ABWR has a number of design features that reduce the risk from an ATWS event including a diverse scram system with both hydraulic and electric run-in capabilities for the fine motion control rod drives (FMCRD), an automatic SLCS, and a reactor internal pump (RIP) trip capability. In addition, the scram discharge volume has been removed from the ABWR, eliminating some of the potential problems that could affect the scram function associated with older BWR designs.

The ABWR has an alternate rod insertion system that is diverse, when compared with that of the RPS, from sensor output to the final actuation device. The ARI system has redundant scram air header exhaust valves and is designed to perform its function in a reliable manner.

19.2.2.1.1.2 Basis for Acceptability

In SECY-90-016, the NRC concluded that evolutionary LWR designs should provide diverse methods of inserting control rods to mitigate a potential ATWS and to ensure a safe reactor shutdown. The ABWR incorporates a diverse method for inserting control rods. The ABWR complies with the ATWS rule, as concluded in Section 15.5 of this report, and the design is capable of satisfactorily mitigating the effects of an ATWS and preventing an ATWS event from evolving into a severe accident with core damage. The staff concludes that the ABWR meets the criteria specified in SECY-90-016 through incorporation of the features discussed above.

19.2.2.1.2 Station Blackout

An SBO involves the complete loss of ac electrical power to the essential and nonessential switchgear busses in a

Severe Accidents

nuclear power plant (i.e., loss of offsite electric power system concurrent with turbine trip and unavailability of the on-site emergency ac power system). SBO does not include the loss of available ac power to buses fed by station batteries through inverters or by alternate ac sources, nor does it assume a concurrent single failure or DBA.

During normal plant operation, power is supplied to the Class 1E distribution system from the main generator. Following plant shutdown, the preferred power source is the offsite grid, which provides a continuous source of ac electric power to equipment required to maintain core coolability. If the power from the offsite grid is not available, the on-site distribution system will sense an undervoltage condition and initiate a transfer to the emergency diesel generators (EDGs) for continued power. In the event of the loss of both the offsite grid and EDGs, an SBO has occurred. As most DHR and containment heat removal systems are dependent upon ac power for operation, failure to provide core cooling during an SBO will likely result in core temperature and pressure increases and may lead to a severe accident.

The SBO rule (10 CFR 50.63) allows several design alternatives to ensure that a plant is able to withstand an SBO for a specified duration and recover. A complete evaluation of the ABWR relative to the SBO rule is provided in Section 8.3.9 of this report.

19.2.2.1.2.1 Preventive and/or Mitigative Features

In SECY-90-016, the staff stated that the preferred method of demonstrating compliance with 10 CFR 50.63 is through the installation of a spare (full-capacity) alternate ac power source of diverse design that is consistent with the guidance in Regulatory Guide (RG) 1.155 and is capable of powering at least one complete set of normal shutdown loads. The staff recommended that the Commission approve the requirement for an alternate ac source for evolutionary LWRs. In its June 26, 1990 SRM, the Commission approved the staff's position. Therefore, the staff's proposed applicable regulation for an alternate ac source is as follows:

The standard design must provide an alternate ac power source for the purposes of dealing with station blackout.

The ABWR design includes three independent electrical divisions, each capable of providing power to a high-pressure and low-pressure water injection division, each powered by a full-capacity EDG, and each division capable of independently shutting down the reactor. In addition, the ABWR design includes an alternate ac combustion turbine to back up the diesel generators. The RCIC

system (with its supporting systems) is designed to perform its function without ac power for at least 2 hours and also have an ultimate capability to function for 8 hours without ac power. Extended blackout capabilities are also provided by the ACIWA system. This system allows for makeup to the reactor vessel following depressurization or to the containment sprays from a direct-drive diesel fire pump or by connecting an external pumping source, such as a fire truck, to a yard standpipe into the RHR system.

19.2.2.1.2.2 Basis for Acceptability

In SECY-90-016, the NRC concluded that designers should meet the SBO rule by including an alternate ac power source (i.e., CTG) of diverse design capable of powering at least one complete set of normal shutdown loads. To cope with SBOs, the ABWR has included an alternate CTG. Based on the preventive and mitigative features described above, the staff concludes that the ABWR has met the criteria of SECY-90-016 and the staff's proposed applicable regulation for station blackout.

19.2.2.1.3 Fire Protection

The Commission concluded that fire protection issues that have been raised through operating experience and the external events program must be resolved for evolutionary light water reactors. In SECY-90-016, the staff recommended that current NRC guidance to resolve fire protection issues be enhanced to minimize fire as a significant contributor to the likelihood of severe accidents and DBAs. As indicated in SECY-90-016, the ABWR design must ensure that safe shutdown can be achieved, assuming that all equipment in any area will be rendered inoperable by fire and that reentry into the fire area for repairs and operator actions will be impossible. Because of its physical configuration, the control room is excluded from this approach, provided an independent alternative shutdown capability that is physically and electrically independent of the control room is included in the design. The ABWR design must also provide fire protection for redundant shutdown systems in the reactor containment building that will ensure, to the extent practical, that one shutdown division will be free of fire damage. Additionally, the ABWR design must ensure that smoke, hot gases, or fire suppressant will not migrate into other fire areas to the extent that they could adversely affect safe-shutdown capabilities, including operator actions. These fire protection measures are for both DBAs and severe accidents.

19.2.2.1.3.1 Preventive and/or Mitigative Features

Section 9.5.1 of this report provides a description and evaluation of the ABWR features provided to prevent and

mitigate fires. In particular, this section addresses protection of safe-shutdown equipment, passive fire protection features, fire detection, fire protection water supply system, water fire suppression systems, gaseous fire suppression systems, fire extinguishers, emergency communication and lighting, emergency breathing air, curbs and drains, smoke control, access and routes, construction materials and combustible contents, and interaction with other systems.

19.2.2.1.3.2 Basis for Acceptability

Based on the evaluation in Section 9.5.1 of this report and on the discussion above, the staff concludes that the ABWR design meets the criteria identified in SECY-90-016 and is acceptable for preventing and mitigating threats from fires for DBAs and severe accidents.

19.2.2.1.4 Intersystem Loss-of-Coolant Accident

Intersystem loss-of-coolant accidents (ISLOCAs) are defined as a class of LOCAs in which the RCS pressure boundary is breached and coolant is lost through an interfacing system with a lower design pressure. The breach may occur in portions of piping located outside the primary containment, causing a direct and potentially unisolable discharge from the RCS to the environment. An ISLOCA is of concern because of potential direct releases to the environment, loss of core cooling, and loss of core makeup.

High or low pressure interfaces occur on many lines including low-pressure injection lines and the RHR heat exchangers. An ISLOCA occurs when high pressure is introduced in a low-pressure system because of valve failure or an inadvertent valve actuation. In either case, the overpressurization can cause the low-pressure system or components to fail. An ISLOCA concurrent with a loss of all core cooling may lead to core damage.

19.2.2.1.4.1 Preventive and/or Mitigative Features

In SECY-90-016, the staff recommended that evolutionary LWR designs reduce the possibility of a LOCA outside containment by designing (to the extent practicable) all systems and subsystems connected to the RCS to an ultimate rupture strength (URS) at least equal to the full RCS pressure. The "extent practicable" phrase shows a realization that all systems must eventually interface with atmospheric pressure and that for certain large tanks and heat exchangers, it would be difficult or prohibitively expensive to design such systems to a URS equal to full RCS pressure. The staff further recommended that systems that have not been designed to withstand full RCS

pressure should include (1) the capability for leak testing of the pressure isolation valves, (2) indication in the control room of valve position when isolation valve operators are deenergized, and (3) high-pressure alarms to warn control room operators when rising RCS pressure approaches the design pressure of attached low-pressure systems and both isolation valves are not closed.

In its June 26, 1990 SRM, the Commission approved the staff's position on ISLOCA provided that all elements of the low-pressure system are considered (e.g., instrument lines, pump seals, heat exchanger tubes, and valve bonnets).

The structural capability of low-pressure piping systems interfacing with the reactor coolant pressure boundary to withstand the consequences of an ISLOCA is discussed in Section 3.9.3.1.1 of this report. In addition, GE performed a systematic evaluation of interfacing systems to ensure that the SECY-90-016 requirements were satisfied. The resolution of this issue is provided in Section 20.2.19 of this report.

19.2.2.1.4.2 Basis for Acceptability

As indicated in Chapter 20 and Section 3.9.3.1.1 of this report, the staff concludes that GE has met the criteria from SECY-90-016, as approved by the Commission, regarding ISLOCA prevention and mitigation for the ABWR.

19.2.3 Deterministic Assessment of Severe Accident Mitigation

19.2.3.1 Overview of the ABWR Containment Design

The ABWR containment maintains the pressure suppression design of other BWR containments. The containment is a reinforced concrete cylindrical structure with a steel upper drywell head and internal steel liner to reduce leakage. The ABWR containment atmosphere is made inert to preclude hydrogen combustion and a slightly positive pressure is maintained to prevent air from leaking in. The containment basemat is 5.5 m (18 ft) thick with several layers of reinforcements. The top slab is an integral part of the fuel pool. The containment wall is a right circular cylinder 2 m (6.56 ft) thick with an inside radius of 14.5 m (47.57 ft) and height of 29.5 m (96.78 ft). Its internal space is divided by the diaphragm floor and the reactor pedestal into an upper drywell chamber, a lower drywell chamber, and a wetwell. The upper drywell volume surrounds the RPV and houses the steam and feedwater lines and other connections of the reactor primary coolant system and safety/relief valves,

Severe Accidents

and the lower drywell volume houses the reactor internal pumps and FMCRD.

The cylindrical RPV pedestal separates the lower drywell from the wetwell. Ten DCVs are built into the RPV pedestal and connect the upper drywell and lower drywell. The DCVs are extended downward by steel pipes, each of which has three horizontal vent outlets into the suppression pool. The wetwell consists of an air volume and suppression pool. Steam from a reactor vessel blowdown or from a break in a major pipe inside the drywell condenses in the suppression pool through the SRVs or the DCVs. A vacuum breaker system is provided between the drywell and wetwell to prevent excessive differential pressures.

The containment heat-removal system is an integral part of the RHR system. Suppression pool temperature is controlled in the suppression pool cooling mode of RHR. RHR also includes the containment spray feature to cool the containment air space. A capability also exists to directly connect the drywell spray header to the fire protection system pumps, one of which is driven by a diesel engine.

The ABWR design includes a passive floodler system. This system consists of a group of pipes that horizontally passes through the pedestal wall and connects the suppression pool to the lower drywell. These pipes terminate in the lower drywell with fusible plugs. These plugs are designed to melt and open a connection between the suppression pool and the lower drywell region when the drywell airspace temperature reaches 260 °C (500 °F).

A COPS consisting of two relief rupture disks in succession from the wetwell airspace is provided. The inner rupture disk, which controls the actual pressure, actuates at a pressure of 617.8 kPa gage (90 psig). This is above the containment design basis pressure of 309.9 kPa gage (45 psig), but below the ASME Service Level C limit of 666.9 kPa gage (97 psig).

19.2.3.2 Severe Accident Progression

This section provides a description of the processes, both physical and chemical, that may occur during the progression of a severe accident and how these phenomena affect containment performance. This description is intended to be generic in nature. However, many aspects of severe accident phenomena depend on the specific reactor type or on the containment design features. This information has been extracted from NUREG/CR-5132 Severe Accident Insights Report, NUREG/CR-5597 In-Vessel Zircaloy Oxidation/Hydrogen Generation Behavior During Severe Accidents, and NUREG/CR-5564 Core-

Concrete Interactions Using Molten UO_2 With Zirconium on a Basaltic Basemat.

Severe accident progression can be divided into two phases: an in-vessel stage and an ex-vessel stage. The in-vessel stage generally begins with insufficient DHR and can lead to melt-through of the reactor vessel. The ex-vessel stage involves the release of the core debris from the reactor vessel into the containment and resulting phenomena such as CCI, FCI, and DCH.

19.2.3.2.1 In-Vessel Melt Progression

In severe accidents that proceed to vessel failure and release of molten core material into the containment, the in-vessel melt progression establishes the initial conditions for assessment of the thermal and mechanical loads that may ultimately threaten the integrity of the containment. In-vessel melt progression encompasses the phenomena and processes involved in a severe core damage accident. These phenomena and processes start with uncovering of the core and initial heat-up, and continue until either (1) the degraded core is stabilized and cooled within the reactor vessel, or (2) the reactor vessel is breached and molten core material is released into the containment. The phenomena and processes in the ABWR that can occur during in-vessel melt progression include the following:

- Core heat-up resulting from loss of adequate cooling
- Metal-water reaction and cladding oxidation
- Eutectic interactions between core materials, e.g., control blades and fuel assembly channel boxes, resulting in relocation of molten material. Eutectics are mixtures of materials with a melting point lower than that of any other combination of the same components.
- Melting and relocation of cladding, structural materials, and fuel
- Formation of blockages near the bottom of the core resulting from the solidification of relocating molten materials (a wet-core scenario)
- Drainage of molten materials to the vessel lower head region (a dry-core scenario)
- Formation of a melt pool, natural circulation heat transfer, crust formation, and crust failure (a wet-core scenario)
- Lower head breach resulting from failure of a penetration or from local or global creep-rupture.

Adequate core cooling can be defined as providing enough cooling water flow to the reactor core to remove the decay heat produced. Or, if the decay heat is transferred to the containment, providing enough cooling water to containment heat removal systems to remove the decay heat transferred from the core. The mechanisms by which sufficient cooling is provided to the reactor core or the containment are numerous, diverse, and redundant. These mechanisms include both safety and non-safety systems. Examples of safety systems include the RCIC system, high-pressure core flooders systems, RHR system, and ADS. Examples of non-safety systems include the condensate and feedwater systems, condenser, relief valves, and the ACIWA system.

In the event of failure of all safety and non-safety systems to remove the decay heat produced, the core will heat up to the point where damage to the fuel and fuel cladding may occur. Decay heat is transferred through radiative, conductive, and convective heat transfer to the steam, other core materials, and nonfuel materials within the reactor. The insufficient cooling supply results in coolant boiloff and a decreasing level within the reactor vessel as the decay heat generation exceeds the heat removal rate. The coolant level within the core further decreases so that the fuel rods above the coolant level are only cooled by rising steam. The fuel rods begin to overheat and cladding oxidation begins in the presence of steam at high temperatures. As the cladding oxidizes in the presence of steam, hydrogen and additional heat are generated. The fuel cladding is made of a zirconium alloy called zircaloy.

The initial zircaloy oxidation involves oxygen diffusion through a ZrO_2 surface layer. As the fuel rods continue to heat up from decay heat and the exothermic zirconium oxidation reaction, the materials within the reactor with low melting points are expected to melt first and may form eutectics.

Zircaloy with a melting point of $1,757^\circ\text{C}$ ($3,194^\circ\text{F}$) begins to melt breaking down the protective ZrO_2 layer, exposing unoxidized zircaloy. Following this, local melting of the fuel rods may cause changes in the core geometry resulting in different steam flow paths. On the one hand, this can lead to an increase in the oxidation process as access to the unoxidized zircaloy is available. On the other hand, the melt formation or changes in the steam flow path could reduce the zircaloy surface available for oxidation and thereby decrease the overall reaction process. In some accident scenarios in which residual amounts of water remain in the bottom of the core and lower plenum, substantial steaming and oxidation can take place.

In addition to oxidation, the potential exists for the zircaloy to interact with the UO_2 fuel, forming eutectics. Formation of eutectics may decrease the effective surface area for oxidation and the overall oxidation rate. The melting point of zircaloy is dependent upon its oxidation state and lattice structure. It has three melting points which include $1,877^\circ\text{C}$ ($3,410^\circ\text{F}$) (beta-Zr), $1,977^\circ\text{C}$ ($3,590^\circ\text{F}$) (alpha- $Zr(O)$), and $2,677^\circ\text{C}$ ($4,850^\circ\text{F}$) (ZrO_2). When partially oxidized zircaloy is in contact with UO_2 , an alpha- $Zr(O)/UO_2$ based eutectic will form with a liquefaction temperature of approximately $1,897^\circ\text{C}$ ($3,446^\circ\text{F}$). Therefore, in the presence of good fuel or cladding contact, fuel liquefaction and melt relocation will commence around this temperature. This has the potential to affect the oxidation behavior of zircaloy-based melt.

Various severe-fuel damage (SFD) test programs discussed in the NUREG's listed above sponsored by the NRC indicate that oxidation of the zircaloy is largely controlled by the availability of a steam supply and that high rates of hydrogen generation can continue after melt formation and relocation. Some of these experiments indicate that the majority of the hydrogen generated occurs after onset of zircaloy melting and fuel dissolution. In steam-rich experiments, oxidation took place over most of the fuel bundle length, and most of the hydrogen is generated early. For steam-starved experiments, oxidation was limited to local regions of the fuel bundle and the majority of the hydrogen is generated after the onset of Zr/UO_2 liquefaction and relocation.

The ABWR contains more than 72,000 kg (158,700 lbm) of zirconium in the active fuel region that has the potential to generate more than 3,100 kg (6,834 lbm) of hydrogen. Hydrogen production and accumulation may represent challenges to the containment in numerous ways including deflagration, detonation, and pressurization. The ABWR containment will be made inert with nitrogen to prevent the occurrence of any deflagration or detonation. Pressurization of the containment from the generation of hydrogen gases will not exceed ASME Service Level C limits.

The SFD tests indicated the potential for incoherent melt-relocation due to noncoherent temperatures within the test bundles. This is because of the different core materials present with a wide range of melting points and eutectic temperatures. Formation of eutectics would result in a nonuniform melting and relocation process. Further differences in the melt-relocation process can be attributed to asymmetric bundle heating that can increase because of zircaloy oxidation. This process begins when one area of the fuel bundle is initially at a temperature higher than the other areas. The higher-temperature zircaloy will consume the available steam through oxidation at a quicker rate.

Severe Accidents

The oxidation reaction makes the hotter areas hotter still, which further increases the oxidation rate and the local temperatures. This autocatalytic nature of zircaloy oxidation appears to contribute to asymmetric bundle heatup and the potential for incoherent melt-relocation.

As the temperature of the core increases, fission products in vapor form are released. The mixing and transport of these fission products within the primary system depend upon flow paths set up by any existing steam or hydrogen and any interactions with surfaces within the reactor. Such surfaces as the upper internal structures of the reactor vessel may act as a filter where microscopic fission product aerosols suspended in the gas can settle on comparatively cool surfaces by thermophoresis and diffusiophoresis. In addition, retention mechanisms such as turbulent deposition and gravitational settling occur.

The core melt progression, including relocation and fission product release, becomes increasingly difficult to predict as it continues to degrade. The core melt could relocate into the lower reactor vessel plenum. If water is present in the lower plenum, the potential exists for in-vessel steam explosions, where molten fuel rapidly fragments and transfers its energy causing rapid steam generation and shock waves. Another possibility is that the core debris within the lower plenum may quickly melt through the reactor vessel or interact with available water before melting through and entering the lower drywell.

The in-vessel core melt progression, including core degradation, relocation, and failure of the reactor vessel, is rather uncertain. This uncertainty includes the potential for in-vessel steam explosion, the interaction of core debris with internal vessel structures, the time and mode of vessel failure, the composition of the core debris released at vessel failure, the amount of in-vessel hydrogen generation, the in-vessel fission product release, and the transport and retention of fission products and other core materials in the RCS.

19.2.3.2.2 Ex-Vessel Melt Progression

Ex-vessel severe accident progression is affected by the mode and timing of the reactor vessel failure; the primary system pressure at reactor vessel failure; the composition, amount, and character of the molten core debris expelled; the type of concrete used in containment construction; and the availability of water to the lower drywell. The initial response of the containment to ex-vessel severe accident progression is largely a function of the pressure of the RCS at reactor vessel failure and the existence of water within the reactor cavity. If not prevented through design features, risk consequences are usually dominated by early containment failure mechanisms that could result from

energetic severe accident phenomena such as HPME with DCH and ex-vessel steam explosions. The long-term response of the containment from ex-vessel severe accident progression is largely a function of the containment pressure and temperature resulting from core-concrete interaction and the availability of containment heat removal mechanisms.

At high RCS pressures, the molten core debris could be ejected from the reactor vessel in jet form causing it to fragment into small particles. The potential exists for the core debris ejected from the vessel to be swept out of the lower drywell and into the upper drywell. Finely fragmented and dispersed core debris could heat the containment atmosphere and lead to large pressure spikes. In addition, chemical reactions of the core debris particulate with oxygen and steam could add to the pressurization loads. This severe accident phenomenon is known as HPME with DCH.

To prevent this phenomenon, the ABWR has incorporated a reliable depressurization system to provide assurance, that in the event of a core melt scenario, that failure of the RPV would occur at a low pressure. Should the RPV fail at a high pressure, the design of the ABWR containment would provide an indirect pathway from the lower to the upper drywell in an effort to decrease the amount of core debris that could contribute to DCH.

The equipment tunnels are located on the periphery of the lower drywell at a midlevel elevation. Core debris exiting the reactor vessel or entrained from the lower drywell during HPME has the potential to reach the tunnels. An accumulation of core debris within the tunnels could lead to melt-through and development of a suppression pool bypass mechanism. GE had not addressed this issue in the SSAR. The staff indicated that it believed that an acceptable resolution to this issue would be for GE to provide reasonable assurance that an appreciable amount of core debris would not enter the tunnels. This could be done by showing that the existing equipment within the lower drywell provides a tortuous pathway to the lower drywell periphery or providing an additional shield structure over the tunnels. This was Open Item F19.2.3.2.2-1.

GE addressed this issue in a letter dated February 7, 1994, which proposed a new SSAR Section 19.E.2.3.6. GE indicated that the equipment tunnels will be partially covered with 1.2 meters of suppression pool water at the low water level allowed by technical specifications. In the event core debris melts through the equipment tunnel, the debris will enter the suppression pool and any additional gases from the lower drywell will pass through the indicated suppression pool level. Also, a HPME results in

core debris and elevated temperatures within the upper drywell. In order for the containment penetrations to withstand the elevated temperatures, the operator must actuate the containment spray system. The water from the containment spray system eventually accumulates in the suppression pool, raising the water level to provide additional water coverage of the equipment tunnels. Therefore, the suppression pool water level covering the lower section of the equipment tunnels is sufficient to preclude this potential suppression pool bypass pathway. The staff finds this acceptable. Therefore, Open Item F19.2.3.2.2-1 is resolved.

RPV failure at high or low pressure coincident with water present within the lower drywell could lead to FCI with the potential for rapid steam generation or steam explosions. Rapid steam generation involves the pressurization of containment compartments from nonexplosive steam generation beyond the capability of the compartment to relieve the pressure so that local overpressurization failure of the compartment occurs. Steam explosions involve the rapid mixing of finely fragmented core debris with surrounding water resulting in rapid vaporization and acceleration of surrounding water creating substantial pressure and impact loads. The ABWR is designed so that there is a very low likelihood of water within the lower drywell at the time of reactor vessel failure.

The eventual contact of molten core debris with concrete in the lower drywell will lead to core-concrete interaction (CCI). CCI involves the decomposition of concrete from core debris and can challenge the containment in various mechanisms, including (1) pressurization resulting from the production of steam and noncondensable gases to the point of containment rupture, (2) the transport of high temperature gases and aerosols into the upper drywell leading to high-temperature failure of the containment seals and penetrations, (3) liner melt-through, (4) reactor pedestal melt-through leading to relocation of the reactor vessel and tearing of containment penetrations, and (5) the production of combustible gases such as hydrogen and carbon monoxide. CCI is affected by many factors including the availability of water to the lower drywell, the containment geometry, the composition and amount of core melt, the core melt superheat, and the type of concrete involved.

The ABWR has incorporated several design features to mitigate the effects of CCI. These include an LDF system, an ACIWA system, basaltic concrete for the lower drywell floor, and the COPS. The LDF system provides suppression pool water to assist in cooling core debris once it has entered the lower drywell. The ACIWA system provides for both reactor vessel injection and drywell spray capability to cool core debris or control containment

pressurization. Basaltic concrete protects the containment liner from melt-through and decreases the amount of non-condensable gases generated during CCI when compared with limestone-based concretes. The COPS is designed to passively relieve containment pressure to prevent gross containment failure during severe accidents when the containment pressure approaches ASME Service Level C limits. This relief pathway takes advantage of the scrubbing capability of the suppression pool to limit any offsite releases.

19.2.3.3 Severe Accident Mitigative Features

19.2.3.3.1 Hydrogen Generation and Control

Generation and combustion of large quantities of hydrogen is a severe accident phenomenon that can threaten containment integrity. The major source of hydrogen generated is from the oxidation of zirconium with steam when the zirconium reaches temperatures well above normal operating levels. This reaction is commonly referred to as the metal-water reaction.

Research indicates that in-vessel hydrogen generation associated with core-damage can vary over a wide range. The specific amount of oxidation is dependent on a variety of parameters related to sequence progression. These include the RCS pressure, the timing and flow rate of reflooding if it occurs, and the temperature profile of the reactor core during the course of the accident sequence. In addition, ex-vessel hydrogen generation must be considered. Hydrogen is produced as a result of ex-vessel core debris reacting with steam or concrete.

19.2.3.3.1.1 Preventive and/or Mitigative Features

10 CFR 52.47(a)(1)(ii) requires applicants for a standard design certification to provide demonstration of compliance with any technically relevant portions of the Three Mile Island Requirements set forth in 10 CFR 50.34(f). 10 CFR 50.34(f)(2)(ix) requires a system for hydrogen control that can provide with reasonable assurance that uniformly distributed hydrogen concentrations in the containment do not exceed 10 percent during and following an accident that releases an equivalent amount of hydrogen as would be generated from a 100 percent fuel-clad metal-water reaction, or that the postaccident atmosphere will not support hydrogen combustion.

In SECY-90-016, the staff recommended that the Commission approve the staff's position that the requirements of 10 CFR 50.34(f)(2)(ix) remain unchanged for evolutionary LWRs. In its June 26, 1990 SRM, the Commission approved the staff's position.

Severe Accidents

To comply with 10 CFR 50.34(f)(2)(ix), the ABWR will have an inert atmosphere during normal operation. The inert containment prevents hydrogen combustion and/or detonations from occurring.

10 CFR 50.34(f)(3)(v) requires containment integrity to be maintained below ASME Service Level C limits for steel containments and the factored load category for concrete containments during an accident that releases hydrogen generated from 100 percent fuel-clad metal-water reaction. GE performed an analysis in Section 19E.2.3.2 of the ABWR SSAR of the capability of the containment to withstand pressurization from a 100-percent, fuel-clad, metal-water reaction coupled with a large-break LOCA. The analysis indicated a peak containment pressure of about 618 kPa absolute (75 psig).

The ABWR has a concrete containment with a steel upper drywell head. The steel upper drywell head, based on GE structural analyses, has been shown to be the most limiting structural component of the containment, with a Service Level C limit of 666.9 kPa gage (97 psig). Therefore, the containment pressurization from a 100-percent, fuel-clad, metal-water reaction coupled with a large-break LOCA is below the Service Level C limit.

19.2.3.3.1.2 Basis for Acceptability

The ABWR design meets the requirements of SECY-90-016 and 10 CFR 50.34(f)(2)(ix) by utilizing a nitrogen-inerted atmosphere within its containment. The ABWR design is capable of withstanding the pressurization loadings resulting from a large-break LOCA and hydrogen generation equivalent to a 100-percent fuel-clad metal-water reaction as required by 10 CFR 50.34(f)(3)(v).

19.2.3.3.2 Core Debris Coolability

Coolability and quenchability have been the subject of extensive research over the past decade. However, much uncertainty still exists about these phenomena, which will most likely not be resolved in the near future. Because of this uncertainty, the NRC decided not to address the question of whether coolability or quenchability has been achieved or can be achieved, but rather, what the impact on the containment design is if they are not achieved.

CCI is a severe accident phenomenon that involves the melting and decomposition of concrete in contact with molten corium. This phenomenon may occur following accident sequences that result in molten corium's breaching the reactor vessel and spreading onto the lower drywell floor. The thickness of the corium layer within the lower drywell depends upon the amount of core debris, its

spreadability, and the lower drywell floor area. Once on the drywell floor, the molten corium may react with the concrete and any available water, producing noncondensable gases, water vapor, and heat from exothermic reactions.

CCI can challenge the containment by various mechanisms including pressurization from noncondensable gas and steam generated, destruction of structural support members, and melt-through of the containment liner. Noncondensable gases, primarily carbon dioxide, carbon monoxide, and hydrogen, are released from the concrete as it decomposes and are formed from reactions between water and metals within the molten corium. The corium and concrete are heated from the combined effects of decay heat and exothermic chemical reactions.

19.2.3.3.2.1 Preventive and/or Mitigative Features

In SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs," the staff recommended that the Commission approve the position that both the evolutionary and passive LWR designs meet the following criteria: (1) provide reactor-cavity floor space to enhance debris spreading, (2) provide a means to flood the reactor cavity to assist in the cooling process, (3) protect the containment liner and other structural members with concrete, if necessary, and (4) ensure that the best-estimate environmental conditions (pressure and temperature) resulting from core-concrete interactions do not exceed Service Level C for steel containments or factored load category for concrete containments, for approximately 24 hours. In addition, they must ensure that the containment capability has margin to accommodate uncertainties in the environmental conditions from core-concrete interactions. In its July 21, 1993 SRM, the Commission approved the staff's position.

Therefore, the staff's proposed applicable regulation for core debris coolability is as follows:

The standard design must include features that reduce the potential for and effect of interactions with molten core debris by:

- (1) providing reactor cavity floor space to enhance debris spreading;
- (2) providing a means to flood the reactor cavity to assist in the cooling process;
- (3) protecting the containment liner and other structural members with concrete, if necessary; and

- (4) providing design features that ensure that the best-estimate environmental conditions (pressure and temperature) resulting from core-concrete interactions do not exceed service level C for steel containments or factored load category for concrete containments, for approximately 24 hours.

GE incorporated numerous features in the ABWR to help mitigate the effects of core-concrete interaction. The following features were judged by the staff as being most important: a large lower drywell floor area with minimal obstructions to the spreading of core debris, an LDF system, an ac-independent water addition system, use of sacrificial basaltic concrete for the lower drywell floor, a thick reactor pedestal wall, and a COPS.

19.2.3.3.2.1.1 Lower Drywell Floor Area

The lower drywell is 10.6 meters (34.8 ft) in diameter, which provides a floor area of 88 m² (947.2 ft²). The lower drywell contains embedded sump pits that could lead to the accumulation of core debris and accelerated CCI. To prevent this, the sumps are provided with protection to prevent the entrance of core debris. The sump protection is described in 19.2.3.3.8 below.

Even with the presence of the sumps, the lower drywell will have an unobstructed floor area greater than 79 m² (850.4 ft²). This is sufficient floor area to satisfy the EPRI design criterion of 0.02 m²/MWt for debris coolability. This value represents the EPRI Requirements Document estimate of what is required to adequately cool corium debris. The staff does not support or dispute the EPRI floor sizing criterion. Instead, the staff concludes that an unobstructed floor area, along with the design features mentioned above, provides measures to promote the potential for core debris coolability, but does not necessarily ensure it.

To determine whether the lower drywell meets the criteria within SECY-93-087 relative to providing reactor cavity floor space to enhance debris spreading, the staff evaluated the total size of the lower drywell, the number of obstructions present to prevent the spreading of molten core debris, and the impact on the containment design of requiring further modifications. Based on minimal obstructions on the floor area described above, the staff concludes that the design is acceptable.

19.2.3.3.2.1.2 Lower Drywell Flooder System

An LDF was incorporated into the ABWR design to supply water from the suppression pool to the lower drywell to

assist in the cooling process of the corium. The water also cools and condenses gases that evolved during CCI, thereby limiting containment temperature and pressure increases. The LDF is discussed in Sections 9.5.12 and 19E.2.8.2 of the SSAR.

The LDF consists of ten 100 mm (4 in.) stainless steel piping lines from the suppression pool to the lower drywell with thermally activated flooder valves attached to them. The thermally activated flooder valves open when the lower drywell air temperature reaches 260 °C (500 °F). Each flooder valve has a minimum flow rate of 10.8 Kg (2.77 gallons/sec) and contains four components: a stainless steel disk, a teflon disk, a fusible metal plug, and a plastic cap. The stainless steel disk prevents suppression pool water from corroding the teflon disk and fusible metal plug. The teflon disk provides an insulating barrier to prevent the suppression pool water from contacting the fusible metal plug. This insulating barrier is needed to assure that the fusible metal plug is not cooled by the suppression pool water and prevented from melting. The teflon disk will not melt or stick in the valve because its softening temperature is approximately 400 °C (769 °F) and its chemical resistance is higher. It therefore will not adhere to the stainless steel plug or to the fusible plug. The fusible metal plug has a small, raised, annular ring around its circumference approximately 2.0 mm (0.08 in.) high. It is this annular ring that has to melt in order for the LDF to actuate.

The fusible metal plug is made of an alloy mixture of two or more metals so that the plug melts when its temperature reaches 260 °C (500 °F). The end of the flooder valve line is covered with a plastic cover with a low melting point below 130 °C (266 °F). This plastic cover prevents corrosion of the fusible metal material from intrusion of moisture. The flooder valves are mounted in the vertical position so that the fusible metal faces downward to facilitate opening of the valve when the melting temperature of the fusible metal is reached. Heat transfer resulting in melting of the fusible plug occurs through a combination of conduction, convection, and radiation. Heat is conducted from the stainless steel pipe to the fusible plug causing it to melt. This heat is received from the atmosphere within the lower drywell through convection. In addition, the stainless steel pipe also receives radiative heat from the corium on the lower drywell floor. The LDF is safety-related and seismic Category I.

During each refueling outage, the 10 fusible plug flanges and outlets will be inspected to ensure there is no leakage. Once every two refueling outages, 2 of the 10 fusible plugs will be removed, inspected, and tested to confirm their function and verify the temperature setpoint.

Severe Accidents

In Section 19E.2.8.2.2, GE calculated the minimum acceptable flow rate to remove decay heat and heat generated during CCI to be a total of 0.018 m³/sec (5 gallons/sec) compared to the system flow rate of .099 m³/sec (26 gallons/sec) assuming the failure of one of the ten floodler lines to open. This indicates that only two of the floodler lines are needed to remove decay heat and exothermic heat from zirconium oxidation. Opening of additional floodler lines contributes to the flooding within the lower drywell.

In Section 19E.2.8.2.4, GE calculated approximate values for the minimum time (21 minutes) and maximum time (1.3 hours) to fill the lower drywell. These times are very sensitive to the assumptions used and are strongly dependent upon the accident sequence selected. For instance, the calculations assume a 100-percent core debris at 1-percent-rated thermal power, no heat from exothermic reactions, all heat rejection to the water, and failure of one passive floodler valve to operate. The staff concludes that this is acceptable as it provides a relative time frame to judge the adequacy of the LDF design, but the actual flooding rate and time to complete flooding are accident-sequence specific.

Based on the above discussions, the staff concludes that the LDF meets the criteria of SECY-93-087 for providing a means to flood the reactor cavity to assist in the cooling process of core debris.

19.2.3.3.2.1.3 AC-Independent Water Addition System

In addition to the three electrically and mechanically independent divisions of the RHR system of the ABWR that provide reactor vessel injection and containment spray, an ACIWA system has been incorporated. The ACIWA system consists of piping and manual valves connecting the fire protection system to the loop C RHR pump discharge line downstream of the pump's discharge check valve. The C loop is capable of providing low-pressure injection to the reactor vessel or containment spray to the upper drywell. Within the fire protection system, an independent diesel-driven pump exists that would provide the pumping capability. Additionally, an external hookup outside the reactor building for connection of a fire truck pump to an alternate water source is provided. An ac-driven fire pump is included within the fire protection system. However, its contribution to severe accident prevention and mitigation has been excluded because of its dependence on electrical power. The ACIWA system is discussed in Sections 5.4.7 and 19K.11.5 of the SSAR.

Injection to the reactor vessel using the ACIWA system is intended to prevent core damage. In the event that it is not

initiated in time to prevent core damage and reactor vessel melt-through, the ACIWA, when operated in the reactor vessel injection mode, would provide water to the lower drywell through the breach in the reactor vessel to assist in cooling ex-vessel core debris. This flooding of the lower drywell could be in addition to or in-place of the flooding provided by the LDF. The actual circumstances are accident-sequence specific. For example, if the ACIWA provides flooding to the lower drywell immediately following vessel breach, then the temperature within the lower drywell may never reach the initiation temperature for the LDF or if initiation of the ACIWA is delayed, the LDF would open followed by the ACIWA.

Operation of the ACIWA in the containment spray mode controls atmospheric temperatures in the upper drywell and provides fission product scrubbing. This system is very beneficial in delaying the time to or preventing the opening of the COPS as is indicated in Appendix 19E to Section 19 of the SSAR.

In both the vessel injection mode or containment spray mode, the ACIWA supplies water to the containment thus increasing the thermal mass, which in turn slows the overall pressure rise. Operation of the ACIWA is manual. The diesel-driven fire pump can be operated from the control room; the injection valves, which must be opened or closed, are located in the same loop C ECCS valve room. The ACIWA system can supply, from either the diesel-driven fire pump or fire truck pump, between 0.04 m³/sec (630 gpm) and 0.06 m³/sec (950 gpm) for conditions between runout and back pressure equal to the COPS initiation setpoint. Inspection and testing of the ACIWA system are discussed in SSAR Section 19K.11.5.

Based on the above discussions, the staff concludes that the ACIWA provides another means of flooding the lower drywell to assist in the cooling process of core debris, as specified in SECY-93-087.

19.2.3.3.2.1.4 Sacrificial Basaltic Concrete

Basaltic concrete is a type of siliceous concrete used in the construction of nuclear power plants and is found throughout the United States. This concrete melts over a range of 1,077 °C - 1,376 °C (1970 - 2,510 °F) and typically liberates 1.5 weight-percent carbon dioxide gas and 5 weight-percent water vapor when heated to melting (NUREG/CR-5564).

In Section 6.2.1.1.10.3 of the SSAR, GE stated that the ABWR will use a 1.5 m layer of basaltic concrete above the containment liner with a low gas content. The basaltic concrete selected will have less than 4 weight-percent of calcium carbonate, which results in low gas generation

rates when subjected to CCI. This concrete is designed to protect the containment liner in the event that CCI occurs. Basaltic concrete was chosen as it reduces the production of noncondensable gases when compared with other forms of concrete, such as limestone and limestone-common sand concrete. Reduced noncondensable gas and water vapor production from basaltic concrete has been observed during testing (NUREG/CR-5564 and NUREG/CR-5423). However, basaltic concrete does have a lower melting point and quicker ablation rate when compared with limestone and limestone-common sand concrete.

Using the observations discussed above, based on engineering judgment, the staff concludes that the 1.5 m layer of basaltic concrete meets the criteria specified in SECY-93-087 relating to protecting the containment liner and provides sufficient protection for the containment liner.

19.2.3.3.2.1.5 Reactor Pressure Vessel Pedestal

The basaltic concrete discussed above protects the containment liner from core-concrete attack in the axial direction. Core-concrete attack in the radial direction could affect the RPV pedestal. The cylindrical RPV pedestal is formed from two concentric steel rings interspaced with internal stiffeners and filled with concrete. The RPV pedestal is rigidly connected to the diaphragm floor and separates the lower drywell from the wetwell while supporting the loads from the RPV and the diaphragm floor. Built into the RPV pedestal is the drywell-to-wetwell connecting vent system that directs steam from the lower drywell to the suppression pool and upper drywell.

The inner diameter of the RPV pedestal is the outer boundary of the lower drywell. As such, the pedestal is the radial barrier to the horizontal flow of corium. If corium contacts the RPV pedestal, the inner steel cylinder would be attacked and the concrete fill would be subject to ablation. Unabated ablation could lead to failure of the pedestal and subsequent collapse of the RPV and diaphragm floor leading to gross containment failure.

The width of the RPV pedestal is 1.7 m (5.6 ft). The steel rings and internal stiffeners provide the design strength for the RPV pedestal, while the concrete strength is not considered. In Section 19EC of the ABWR SSAR, GE presents the results of an analysis that indicate that only the steel outer shell and 15 cm (6 in.) of internal stiffeners are required to maintain RPV pedestal loads below 90 percent of yield strength.

The staff performed an estimate of the stresses in the RPV pedestal based on the methodology in "Formulas for Stress and Strain," by R. J. Roark and W. Young, McGraw Hill,

1982. Based on these approximate calculations, the staff concludes that adequate margin exists to the yield strength of the RPV pedestal following 1.5 m (5 ft) of radial ablation and that the RPV pedestal is thick enough to withstand the effects of radial ablation resulting from CCI. These attributes meet the criteria specified in SECY-93-087 for protecting structural members with concrete.

19.2.3.3.2.1.6 Containment Overpressure Protection System

The COPS passively relieves containment pressurization before containment pressure reaches ASME Service Level C limits. This system provides for a controlled release through a containment vent pathway with fission product scrubbing provided by the suppression pool. With respect to CCI, the COPS prevents catastrophic overpressurization failure of the containment for severe accident sequences involving prolonged periods of CCI. The COPS ensures that containment pressurization resulting from CCI does not exceed the ASME Service Level C limit of 666.9 kPa gage (97 psig), as the actuation setpoint is 617.8 kPa gage (90 psig).

19.2.3.3.2.2 Analyses

In SECY-93-087, the staff concluded that the evolutionary light water reactors should ensure that the best estimate environmental conditions (pressure and temperature) resulting from core-concrete interactions do not exceed service Level C for steel containments or factored load category for concrete containments, for approximately 24 hours. In addition, designers should ensure that the containment capability has a margin to accommodate uncertainties in the environmental conditions from core-concrete interactions.

The staff concluded that twenty-four hours was an appropriate time period based on sufficient time to allow for decay of fission products, operator intervention, utilization of accident management strategies, fission product deposition in the containment through natural mechanisms, and offsite protective measures. It was developed as a guideline and not a strict criterion in recognition of the uncertainties in severe accident progression and phenomenology.

19.2.3.3.2.2.1 GE Analyses

In Section 19E.2 of the ABWR SSAR, GE provided the results of its deterministic evaluation for several specific accident challenges to evaluate the containments performance. To perform this evaluation, GE used the MAAP3.0B code modified to model the configuration of

Severe Accidents

the ABWR. The new version of the code is referred to as MAAP-ABWR.

Using the ABWR probabilistic safety assessment, GE considered accident classes representing the largest frequencies in selecting the accident sequences to be studied. Eight accident sequences were selected for analysis using MAAP-ABWR. These accident sequences include loss of core cooling with the reactor vessel failing at low and high pressure, SBO, loss of containment heat removal, large break loss-of-coolant accident, and ATWS at low and high pressure and ATWS concurrent with an SBO. For each accident sequence, several mitigating systems could be used to prevent or reduce the release of fission products into the environment. These mitigating systems include in-vessel recovery, passive flooders system, ACIWA, containment heat removal, and containment sprays.

The results of the analyses for each accident sequence are presented in summary form in Table 19E.2-16 of the ABWR SSAR. These analyses generally indicate core debris coolability and little, if any, CCI. The time-to-release of fission products ranges from 8.6 to 50 hours from the start of the transient with the most likely fission product release location through the COPS. The COPS prevents the containment pressure from reaching the ASME Service Level C limit. However, for some sequences, the time to COPS actuation is less than 24 hours. For example, the accident sequence resulting in a release time of 8.6 hours is of extremely low probability involving an SBO with failure of the combustible gas turbine concurrent with an ATWS in which all reactivity control fails. However, if credit is given to operation of the ACIWA in the containment spray mode, the time-to-release of fission products increases to 26.4 hours.

A benchmark of the containment's passive pressure capability is its ability to accommodate the loss of containment heat removal sequence analyzed by GE in section 19E.2.2.4 of the ABWR SSAR. This analysis assumes that reactor vessel injection is maintained with all the decay heat being transferred to the suppression pool. Core damage does not occur. This analysis indicates that the COPS would actuate in approximately 21.7 hours. COPS actuation results from saturation of the suppression pool pressurizing the containment. This sequence indicates that the time to COPS actuation, even without the added pressurization and energy sources from severe accidents, cannot be extended much beyond 20 hours in the absence of active decay heat removal.

With the addition of noncondensable gases from CCI and heat from the exothermic metal-water reactions during a severe accident, the time to COPS actuation will be less.

This is an important point in that COPS actuation before 24 hours cannot be prevented unless additional heat capacity is added to the containment or a containment heat removal system is recovered. The ACIWA system, as discussed above in Section 19.2.3.3.2.1.3, can provide additional heat capacity to prolong the time to COPS actuation. Based on GE's analysis provided in Table 19E.2-16 of the ABWR SSAR, the time to COPS actuation is delayed by at least 10 hours for cases in which additional water is added to the containment by the ACIWA, when compared with the same sequence in which only the LDF system actuates to cool the core debris. The ACIWA is crucial to delaying the time to COPS actuation.

Section 19EC presents the results of an uncertainty analyses performed by GE using MAAP-ABWR to investigate the uncertainties associated with debris coolability. These analyses evaluated the impact of parameters such as the amount of core debris, debris-to-water heat transfer, amount of steel in the debris, delayed flooding of the lower drywell, and use of the ACIWA system on CCI, containment pressurization, COPS actuation, and fission product release.

As discussed in Section 19.2.3.3.2.1.4 above, the ABWR will have a 1.5 m (4.9 ft) layer of basaltic concrete above the containment liner. This concrete layer is designed to protect the containment liner from being breached in the event that significant CCI occurs. In Section 19EC using the MAAP-ABWR code, GE provided the results of an uncertainty analyses that calculated the extent of axial ablation. The results, provided in Table 19ED.5-2 of the ABWR SSAR, indicate that axial ablation will not exceed 1 m (3.3 ft) in a 24-hour period.

As discussed in Section 19.2.3.3.2.1.5 above, GE indicated that the distance the molten corium must ablate in the radial direction is 1.55 m (5.1 ft) before the minimum wall thickness of the pedestal is reached. In Section 19EC using the MAAP-ABWR code, GE provided the results of an uncertainty analyses that calculated the extent of radial ablation by multiplying the axial ablation depth by 1/5. GE selected the 1/5 value based on the results of previous CCI experiments. This multiplying factor was necessary, as MAAP assumes that radial and axial penetration are identical. The results, provided in Table 19ED.5-2 of the ABWR SSAR, indicate that radial ablation does not represent a significant threat to the containment.

19.2.3.3.2.2 Staff Analyses

The staff analyzed in-house the response of the ABWR using the MELCOR code. In addition, the staff's contractor Sandia National Laboratories (SNL) performed

additional analyses using the MELCOR code. The results of the SNL evaluation were sent to GE and placed on the docket. The MELCOR results generally reproduced the event sequences predicted by MAAP, albeit usually with timing shifts. These timing shifts did not affect the safety insights for the containment analyses.

19.2.3.3.2.3 Conclusions

The staff did not rely on any one specific sequence or scenario performed by GE using the MAAP-ABWR code nor by the staff's contractor (Sandia National Laboratories) in determining whether the ABWR met the criterion in SECY-93-087 for ensuring that containment conditions do not exceed Service Level C for approximately 24 hours from CCI. Rather, the staff evaluated the range of results provided by these codes, with due consideration of the uncertainties inherent within them, and the capability of the design to extend the time period to COPS actuation through intervention. The ACIWA is fundamental to prolonging the period to COPS actuation. Once COPS is actuated, containment pressurization is relieved through a controlled pathway that takes advantage of scrubbing by the suppression pool. The staff recognizes that there are sequences in which COPS actuation in under 24 hours is required to maintain containment stresses below ASME Service Level C limits.

The staff concludes that the ABWR design meets the criterion when use of the mitigation systems incorporated into the design is factored in, such as the LDF and ACIWA system.

19.2.3.3.2.3 Basis for Acceptability

The ABWR meets the criteria of SECY-93-087 and the staff's proposed applicable regulation for core debris coolability through (1) providing a lower drywell unobstructed floor area greater than 79 m² (850 ft²) to enhance debris spreading, (2) providing an LDF system and ACIWA system to flood the lower drywell, (3) providing a 1.5 m (4.92 ft) layer of basaltic concrete to protect the containment liner, (4) providing a thick reactor vessel pedestal, and (5) providing a COPS. Containment conditions resulting from CCI can be maintained below Service Level C for approximately 24 hours, through incorporation of the above-listed design features.

19.2.3.3.3 High-Pressure Core Melt Ejection

High-pressure core melt ejection (HPME) and subsequent DCH are severe accident phenomena that could lead to early containment failure resulting in large radioactive releases into the environment. HPME is the ejection of core debris from the reactor vessel at a high pressure.

DCH is the sudden heatup and pressurization of the containment resulting from the fragmentation and dispersal of core debris within the containment atmosphere.

19.2.3.3.3.1 Preventive and/or Mitigative Features

In SECY-90-016, Evolutionary Light-Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements, the staff concluded that evolutionary LWR designs should include a depressurization system and cavity design features to contain ejected core debris. In its June 26, 1990, SRM, the Commission approved the staff's position that evolutionary LWR designs include a depressurization system and cavity design to contain core debris. In addition, the Commission stated that the cavity design, as a mitigating feature, should not unduly interfere with operations including refueling, maintenance, or surveillance activities.

In SECY-93-087, Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs, the staff recommended that the Commission approve the general criteria that the evolutionary LWR designs provide a reliable depressurization system and cavity design features to decrease the amount of ejected core debris that reaches the upper containment. In its July 21, 1993, SRM, the Commission approved the staff's position.

Based on engineering judgment, the staff believes that examples of cavity design features that will decrease the amount of ejected core debris that reaches the upper containment include ledges or walls that would deflect core debris and an indirect path from the lower drywell to the upper containment. The staff position within SECY-93-087 evolved from the staff position in SECY-90-016 and forms the basis for the staff's review and evaluation.

Therefore, the staff's proposed applicable regulation for high-pressure core melt ejection is as follows:

The standard design must provide a reliable means to depressurize the reactor coolant system and cavity design features to reduce the amount of ejected core debris that may reach the upper containment so that the potential for and effects of interactions with molten core ejected under high pressure are reduced.

The ABWR has an ADS that is discussed in Sections 5.2.2, 6.3, 7.3, and 19D.6.2.5 of the SSAR. The staff's evaluation of the ADS is provided in Sections 6.3 and 7.3.1.2 of this report. The ADS is a safety grade

Severe Accidents

system that can be used to depressurize the reactor when it is shut down and isolated. The ADS consists of 8 SRVs, which are a subset of a total of 18 SRVs. The SRVs provide three main functions: overpressure relief operation using pneumatic actuators, safety operation using steam overpressure, and depressurization operation using the ADS valves. The eight ADS SRVs can function in either the ADS or SRV mode. All of the SRVs are located on the main steamlines and discharge to the suppression pool. The ADS is automatically initiated or can be manually initiated. ADS requires dc power for the solenoid valves and a nitrogen gas supply for the servo valves and pneumatic actuators. Nitrogen gas is supplied from either the high-pressure nitrogen gas supply system or two backup safety-grade nitrogen gas supplies through an accumulator. The SRVs of the ADS each have two accumulators: one for the ADS function and one for the relief function. The accumulator capacity is sufficient for one actuation at drywell design pressure or five actuations at normal drywell pressure.

The ADS SRVs must remain open during the in-vessel phase of a severe accident to ensure that any potential vessel failure occurs at low pressure. Once the reactor vessel has failed, the ADS system is no longer needed. GE indicated that the capability of the depressurization system will not be degraded as a result of the radiation exposure or thermal loads. The DBA radiation environment (TID-14484) is more limiting than that predicted through best-estimate analysis for a severe accident. The thermal loads on the valve actuators are expected to be similar to those used for equipment qualification, and therefore the ADS valves will not be subject to degradation. Nitrogen, which is used to hold the ADS valves open, is supplied from outside of containment and therefore will not be exposed to the harsh severe accident environment. GE indicated that the nitrogen supply will be adequate to assure SRV operability over a full range of hypothetical accidents.

The design of the lower drywell of the ABWR is expected to decrease the amount of ejected core debris that reaches the upper drywell. This decrease is anticipated through the following: (1) capture and trapping of some debris in the lower drywell, (2) impaction and removal of core debris as it is transported between the lower and upper drywell, and (3) division of exiting core debris and gas from the lower drywell into both the upper drywell and wetwell. The lower drywell is a cylindrical cavity with horizontal vent openings to the downcomers at two-thirds of the cavity height. The upper portion of the lower drywell contains the CRD mechanisms. Debris circulating within the lower drywell may be trapped on the CRD mechanisms and other stagnant areas.

For debris to travel from the lower cavity floor to the upper drywell, it must travel vertically from the lower drywell, horizontally to the downcomer vent, and then vertically through the downcomer to the upper containment. This tortuous path, which contains two 90-degree turns, provides an indirect path from the lower drywell to the upper drywell and is expected to enhance removal of core debris from the gas jet stream through impaction. As pressurization of the lower and upper drywell increases, the suppression pool level within the lower downcomers will be forced down to expose the horizontal vents to the suppression pool. Once the horizontal vents have been cleared, the gas and debris leaving the lower drywell will split into two paths: one to the upper drywell and the other to the suppression pool.

The pathway alongside of the reactor vessel is closed off by the reactor vessel skirt. This prevents core debris from the lower drywell from being ejected alongside of the vessel into the upper drywell.

The ABWR containment is inert. A postulated loading from an HPME/DCH event results from hydrogen generation and combustion generated from the oxidation of metallic debris ejected with the core melt. As the ABWR is inerted, any combustion of hydrogen and resulting pressurization loadings is limited to the amount of residual oxygen present within the containment atmosphere.

19.2.3.3.2 Basis for Acceptability

In SECY-93-087, Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs, the staff recommended that the Commission approve the general criteria that the evolutionary LWR designs provide a reliable depressurization system and cavity design features to decrease the amount of ejected core debris that reaches the upper containment. In its July 21, 1993 SRM, the Commission approved the staff's position.

The ADS of the ABWR is provided with a reliable nitrogen supply and dc power to ensure its operability. The containment design of the ABWR is expected to decrease the amount of ejected core debris that reaches the upper drywell. This decrease is anticipated through the following: (1) capture and trapping of debris in the lower drywell, (2) impaction and removal of core debris as it is transported between the lower and upper drywell, and (3) division of exiting core debris and gas from the lower drywell into both the upper drywell and wetwell. Based on the above, the staff concludes that the criteria of SECY-93-087 and the staff's proposed applicable regulation for high-pressure core melt ejection have been met.

19.2.3.3.4 ABWR Containment Vent Design

In SECY-90-016, the staff discussed the incorporation of a containment vent system in the ABWR. The containment vent system is identified as the COPS. The design basis of the COPS is discussed in Sections 6.2.5.2.6.2, 19.E.2.8.1, and 19K.11.6 of the ABWR SSAR.

The desirability of venting a BWR containment to mitigate multiple-failure accidents beyond the design basis has been accepted for some time. Since 1981, the BWR EPGs, developed by the BWR Owners' Group and approved by the NRC for existing BWRs, have called for venting the containment wetwell airspace. The ABWR has a COPS that is designed to avoid gross containment failure resulting from postulated slow-rising overpressure scenarios. The COPS vents the containment from the wetwell airspace, thereby taking advantage of the scrubbing capability of the suppression pool. In transient events, fission products will be directed to the suppression pool through the SRVs. For LOCAs and severe accident scenarios in which the reactor vessel fails, fission products will be directed through the DVCs to the suppression pool.

Without incorporation of the COPS, overpressurization of the containment could lead to failure of the drywell head and fission product releases that have not been scrubbed. The COPS provides a controlled scrubbed vent path from the wetwell with provisions that allow for reisolation of the containment.

19.2.3.3.4.1 System Description

The COPS consists of two containment isolation valves (CIVs) (F007 and F010) and two 200 mm (8 in.) diameter overpressure relief rupture disks (D001 and D002) mounted in succession in a 250 mm (10 in.) diameter line that connects the wetwell airspace to the plant stack. The CIVs are located in the reactor building as close as practical to the containment. Downstream of the CIVs is the first rupture disk (D001) with a pressure setpoint of 617.8 kPa (90 psig) at 93 °C (200 °F). This rupture disk is expected to have a mean opening tolerance of ± 5 percent pressure. Further downstream, before the entrance to the plant stack, is the second rupture disk (D002) with a pressure setpoint of approximately 0.03 MPa (4.35 psig).

The area between the rupture disks is made inert with nitrogen to eliminate the potential for combustion within the portion of the vent path within the reactor building. The rationale for setting the second rupture disk setpoint much lower than that of the first rupture disk is to eliminate the possibility of pressurization between the rupture disks adversely affecting the actuation pressure.

For example, if some type of in-leakage occurred between two high-pressure rupture disks, the containment pressure required to burst the first rupture disk could exceed the design bursting pressure of the disk. By utilizing a high-pressure rupture disk in succession with a low-pressure rupture disk, it is expected that significant in-leakage would pressurize the airspace and burst the low-pressure rupture disk and thereby not affect the required containment pressure for bursting the high-pressure rupture disk.

The CIVs in the COPS pathway are subject to the leak-testing requirements associated with 10 CFR Part 50, Appendix J as specified in the SSAR Section 6.2 and the Inservice Testing Requirements as specified in the SSAR Section 3.9. These CIVs are normally open so that the containment atmospheric pressure is exposed to the first rupture disk during all modes of operation. The CIVs are intentionally not provided with an automatic isolation signal to ensure that they remain open in the event that the COPS is needed to mitigate the consequences of a severe accident. With the CIVs open, the first rupture disk provides the barrier to releases during DBA scenarios. In addition, the CIVs are designed to fail-open upon loss of actuating power. This failure position ensures the availability of the COPS during severe accident scenarios involving multiple failures. If under design-basis conditions, leakage past the rupture disks occurs, the CIVs can be remotely isolated from the control room. In-line radiation monitoring of the vent pathway could be used to detect leakage. However, no leakage is expected within the containment design-basis spectrum 309.9 kPa gage (45 psig). This is based on the design pressure of the rupture disk being substantially above the containment design-basis pressure and pressures associated with a DBA.

The rupture disks will be tested and replaced every 5 years providing additional confidence of actuation pressure. When the rupture disks are procured, a number will be procured at the same time to provide uniformity in the relief pressure.

The purpose of the CIVs is to contribute to the control of the venting process. Following rupture disk actuation, plant operators may decide to reclose the vent path based upon accident management guidance or procedures to be developed by the COL applicant. The CIVs are designed to be capable of fully opening and closing against the pressures associated with venting. GE indicated that the sizing of the COPS vent path is sufficient to allow 35 kg/sec (77.2 lbm/sec) of steam flow at the opening pressure of 617.8 kPa gage (90 psig), which corresponds to an energy flow of about 2.4 percent rated power. At the time of rupture disk actuation, the decay power is

Severe Accidents

expected to be well below 1 percent rated power. The staff concludes that a sufficient margin exists to ensure pressure relief once the rupture disk has been actuated. The sizing of the COPS also prevents suppression-pool-level swell that could force water into the COPS pathway.

The rupture disk setpoint was selected to assure an adequate margin prior to drywell head failure while maximizing the time before fission product releases through COPS. The rupture pressure of the COPS is slightly below the containment's Service Level C capability of 666.9 kPa gage (97 psig), which provides confidence that the integrity of the containment will be maintained before rupture disk actuation.

19.2.3.3.4.2 Basis for Acceptability

In SECY-90-016, the staff recommended that the Commission approve the use of an overpressure protection system that used a dedicated containment vent for the ABWR. In its June 26, 1990, SRM, the Commission approved the staff's recommended use of the COPS on the ABWR, subject to the results of a comprehensive regulatory review, which should fully weigh the potential "downside" risks with the mitigation benefits of the system. In addition, the Commission directed the staff to ensure that the design should provide full capability to maintain control over the venting process.

19.2.3.3.4.2.1 Regulatory Review of the Downside Risks and Mitigation Benefits

The COPS is intended to protect the containment against sequences in which containment integrity is challenged by overpressurization. Without the COPS, the containment failure location is expected to be the upper drywell head with a mean ultimate pressure capability of 1,025 kPa absolute (134 psig). The staff concludes that COPS actuation is preferable to failure of the upper drywell head for the following reasons: (1) releases through the COPS have the advantage of suppression pool scrubbing of fission products, (2) following COPS actuation, the vent pathway can be reisolated, and (3) the actuation setpoint and relief capacity of COPS are selected for optimal containment performance.

The staff evaluated the potential downside risks of COPS on the containment failure frequency and source term.

The pressure setpoint for COPS actuation has an impact on the containment failure frequency. If set too high, the potential exists for failure of the upper drywell head before COPS actuation. GE evaluated the variability in the pressure setpoint at which COPS is actuated, as well as the

uncertainty of the drywell head failure pressure. Based on this evaluation, GE concluded that there is between a 2 percent and 5 percent probability of drywell head failure before COPS actuation, dependent upon the specific accident scenario.

The temperature of the rupture disks has an impact on its bursting pressure. GE evaluated a range of temperatures from 38 to 149 °C (100 to 300 °F) for the time various temperatures would take to the rupture disk's opening. Higher temperatures cause a decrease in the bursting pressure, whereas lower temperatures cause an increase. GE concluded that for the range of the temperatures evaluated, the time to rupture disk's opening was within 0.8 hours of the base case and the probability of drywell head failure before rupture disk's opening varied slightly. These results indicate that temperature variations have only a minor effect on the COPS.

A potential adverse impact of the COPS involves its actuation when it might have been possible to recover a containment heat removal system in the time period after COPS initiation and before failure of the upper drywell head. The recovery of the containment heat removal system would therefore prevent the fission product release through the COPS. GE estimated the probability of this to be about 4 to 11 percent depending upon the accident scenarios.

In Section 19E.2.8.1.4 of the ABWR SSAR, GE provided a comparison of ABWR performance with and without the use of the COPS. This included comparisons of the time and magnitude of fission product release for the frequency-dominant sequence for the ABWR, as well as an assessment of the impact of COPS on the frequency of core damage and drywell head failure for various accident classes. These results indicate that for the dominant sequence, COPS reduces the time of fission product release from 27 to 20 hours if drywell sprays are not available and from 35 to 31 hours if drywell sprays are available. The cesium iodide release fractions are reduced from about 4 percent without COPS to less than 1E-7 with COPS as a result of suppression pool scrubbing.

Table 19E.2-27 of the ABWR SSAR provides the results of the probability of release modes with and without COPS. The probability of drywell head failure increases by about a factor of 40 for accident classes involving transients and loss-of-coolant accidents without COPS. For accident classes that include loss of containment heat removal but successful core cooling before containment failure, the frequency of both drywell head failure and resulting core damage would increase by 2 orders of magnitude without COPS.

In summary, the major benefit of COPS is that it provides assurance that releases from the containment are scrubbed. This fission product scrubbing is provided by the suppression pool. Operators can control releases via COPS through manual isolation valves. While actuation of COPS results in an earlier release of fission products when compared with failure of the upper drywell head, the magnitude of the release is significantly less. Additionally, COPS is expected to reduce the frequency of core damage resulting from accident sequences involving loss of core cooling induced by containment failure.

A major concern related to COPS is that it could lead to earlier, and perhaps unnecessary, releases. However, this does not seem to be a significant factor for the ABWR. In particular, the selection of the system setpoint appears to provide a reasonable balance between the competing goals of minimizing the probability of drywell head failure and maximizing the time before fission product release into the environment. The setpoint is sufficiently high that the time of release will generally still be on the order of 15 to 20 hours, yet low enough that the probability of containment failure before COPS actuation is very small (about 5 percent). Also, the probability of unnecessary system actuation (when it might have been possible to prevent a release by recovering containment heat removal systems before containment failure) is reasonably low (about 10 percent) and COPS is capable of being manually isolated from the control room.

The staff concludes that the COPS has a significant net benefit that outweighs the potential negative aspects of the system.

19.2.3.3.4.2 Control Over the Venting Process

The staff reviewed the provisions provided by GE for ensuring full capability to maintain control over the venting process. These provisions include:

- (a) selection and justification of the containment pressure when the COPS will actuate
- (b) provisions for reclosure of the vent pathway using CIVs that are designed to be capable of fully opening and closing at pressures up to the COPS actuation pressure
- (c) providing radiation monitoring within the vent pathway to monitor the potential for an early release or provide guidance for accident management strategies for reclosure of the vent pathway
- (d) sizing of the vent pathway to prevent further containment pressurization, once the venting

process is actuated, and prevent suppression-pool-level swell into the system piping.

The use of a passive rupture disk prevents the opportunity for venting at pressures below the actuation pressure of the rupture disks. This was a deliberate decision made by GE to prevent early venting and maintain the integrity of the containment as long as possible. To allow for early venting, GE could have installed motor-operated valves instead of a rupture disk.

The staff believes that good engineering rationale can be developed to support either position — to allow for early venting or to maintain containment integrity as long as possible. The decision to provide rupture disks and therefore prevent early venting is philosophical in nature, with the underlying theme of maintaining an intact containment and venting only as a last resort. The containment is the final barrier preventing release of radioactivity into the environment. As such, it should not be unnecessarily or prematurely breached by plant staff.

GE evaluated severe accidents in Chapter 19 of the ABWR SSAR. This evaluation included use of the COPS, where needed, with the rupture disks installed. The evaluation did not include early venting. In this Section (19.2) and Section 19.1 of this report, the staff concludes that GE's analysis in Chapter 19 is acceptable. This acceptability includes the COPS as an integral part of the containment.

19.2.3.3.4.3 Conclusions

The ABWR dedicated containment vent is the COPS, which consists of a vent path from the wetwell airspace to the plant stack. The COPS actuates at a pressure less than ASME Service Level C and has provisions for isolation following actuation. Control over the venting process is assured through selection of the actuation pressure and the capability for vent path reclosure. The results of the regulatory review indicate that the net benefits of the system outweigh the potential negative aspects. Based on the above, the staff concludes that the GE ABWR design meets the Commission-approved staff's position in SECY-90-016 for inclusion of a dedicated containment vent path.

19.2.3.3.5 Suppression Pool Bypass

Although suppression pool bypass is not a severe accident phenomenon, it can become a significant contributor to plant risk. Suppression pool bypass is associated with the failure of the containment system to channel steam and fission product releases through the suppression pool.

Severe Accidents

The fundamental characteristic of a BWR pressure-suppression containment is that steam released from the RCS will be condensed and scrubbed of radionuclides in the suppression pool and that the pressure rise in the containment will therefore be limited. This is accomplished by directing the steam from the RCS to the suppression pool through a drywell to wetwell connecting vent system. However, leakage paths could exist in the pathway between the drywell and wetwell that could allow steam to bypass the suppression pool, pressurize the containment, and lead to a release. Potential sources of steam bypass include leakage through the vacuum relief valves, cracking of the drywell structure, and penetrations through the drywell structure.

19.2.3.3.5.1 Preventive and/or Mitigative Features

In SECY-90-016, the staff concluded that a special effort should be made to eliminate or further reduce the likelihood of a sequence that could bypass the containment. In SECY-93-087, the staff stated that vendors should make reasonable efforts to minimize the possibility of bypass leakage and should account, in their containment designs, for a certain amount of bypass leakage.

The bypass scenario with the greatest threat to containment integrity in the ABWR is suppression pool bypass through the wetwell-to-drywell vacuum breakers. The vacuum breakers prevent the passage of steam from the drywell to the wetwell by use of a single check valve. However, pressure transients in the wetwell and drywell or suppression pool swell could force the vacuum breaker check valve to open. If the vacuum breaker disk fails to reseal, a bypass path would be created and allow drywell steam passage directly to the wetwell airspace. Such a scenario allows steam to bypass the condensation function of the suppression pool, to pressurize the wetwell, and challenge containment integrity. If sufficient steam bypass occurs, containment pressurization rates would increase, thereby decreasing the time to opening of the COPS.

GE analyzed the maximum allowable leakage path area of the ABWR design for DBA-type scenarios. The DBA results are in Section 6.2.1.1.5 of the SSAR. The staff's evaluation of the DBA results are provided in Section 6.2.1.8 of this report.

To mitigate the consequences of suppression pool bypass for both DBAs and severe accidents, the ABWR design includes a containment spray system in the wetwell and upper drywell. The B or C train of the RHR system supplies the water flow to the safety-grade sprays. The spray flow is split with about 800 m³/hr (211,000 gal/hr) going to the drywell spray header and 114 m³/hr (30,100 gal/hr) going to the wetwell spray header.

Although RHR pumps are automatically aligned for their ECCS function, the flow can be manually diverted to the containment sprays to mitigate the pressurization of the wetwell airspace during suppression pool bypass.

In the event of failure of the RHR system, the accident-independent water addition (ACIWA) system can be interconnected with the C loop of the RHR system to provide a flow of between 2385 liters/m (630 gpm) and 3596 liters/m (950 gpm) to the upper drywell containment spray header. The COPS prevents gross containment failure for bypass scenarios in which the containment pressure would exceed the rupture pressure of the rupture disk. COPS allows for a controlled release through reclosure of isolation valves in the vent pathway.

19.2.3.3.5.2 GE Analyses

In Section 19EE of the SSAR, GE performed an analysis to determine the impact on the containment from varying amounts of suppression pool bypass. The study examined the effect of varying a vacuum breaker's bypass leakage area (from zero leakage to leakage from one full open vacuum breaker) on the time to fission product release and the Cesium Iodine (CsI) release fraction at 72 hours for five different scenarios. These scenarios are as follows:

- (1) Bypass leakage begins after passive flooders activation; aerosol plugging is neglected.
- (2) Bypass leakage is present from the beginning of the accident; aerosol plugging is neglected.
- (3) Bypass leakage begins after passive flooders activation; aerosol plugging of the vacuum breaker opening is considered.
- (4) Bypass leakage is present from the beginning of the accident; aerosol plugging of the vacuum breaker opening is considered.
- (5) Bypass leakage is present from the beginning of the accident and the operator initiates the firewater spray system.

The effective vacuum breaker area was varied from 0 to 2030 cm², the latter figure corresponding to one fully open vacuum breaker. The time to fission product release and the CsI release fractions were determined from MAAP-ABWR runs. The dominant severe accident sequence, LCLP (loss of all core cooling with vessel failure occurring at low pressure), was chosen to evaluate plant performance.

A reference point for comparison of the results of GE's analysis is the case with zero bypass leakage and without the firewater spray system. For this case, the elapsed time before rupture disk opening is about 20 hours and the Cal release fraction is less than $10E-07$.

Cal release fractions are orders of magnitude larger for cases with bypass leakage than for the case without bypass leakage. For cases with effective suppression pool bypass areas greater than 400 cm^2 (62 in^2), the 72-hour Cal release fractions are approximately 17 percent.

Scenario 5 examined the effects of drywell spray from the ACTWA on cases with bypass leakage present from the beginning of the accident. Assuming the operator initiates the firewater spray within 2 hours of the start of the accident, the elapsed time to rupture disk opening can be delayed to nearly 30 hours for bypass pathways experiencing up to one fully open vacuum breaker.

19.2.3.3.5.3 Basis for Acceptability

The ABWR has the containment spray system and COPS to mitigate the effects of containment bypass and prevent a bypass scenario from progressing to containment failure. The containment spray system can be supplied from either the RHR or ACTWA system. In SECY-90-016, the staff stated that venting should be delayed for approximately 24 hours following the onset of core damage. For the cases in which one vacuum breaker is fully open, the ABWR meets the intent of this criterion when the initiating of containment sprays through the ACTWA system within 2 hours (as described in scenario 5 above) is factored in. The staff concludes that GE has performed a relatively complete analysis to allow an understanding of the capability of the ABWR containment to accommodate a range of bypass conditions through the vacuum breakers. This analysis highlights the importance of the containment spray system to mitigating the consequences of suppression pool bypass.

19.2.3.3.6 Fuel-Coolant Interaction

The containment function can be challenged by energetic or rapid energy releases. One such energetic or rapid energy release is an FCI that results in a steam explosion. The term "steam explosion" refers to a phenomenon in which molten fuel rapidly fragments and transfers its energy to the coolant resulting in rapid steam generation, shock waves, and possible mechanical damage. To be a significant safety concern, the interaction must be very rapid and must involve a large fraction of the core mass. Steam explosions can occur either in-vessel or ex-vessel.

19.2.3.3.6.1 In-Vessel Steam Explosion

NUREG-1116, A Review of the Current Understanding of the Potential for Containment Failure From In-Vessel Steam Explosions, summarized the deliberations of the Steam Explosion Review Group's (SERG's) understanding of the potential for containment failure arising from in-vessel steam explosions during core melt accidents. The consensus reached by the SERG was that the occurrence of an in-vessel steam explosion of sufficient energetics to lead to containment failure was sufficiently low in probability to allow elimination as a credible threat.

This conclusion was reached despite the expression of differing opinions on the modeling of basic steam explosion sequence phenomenology. An opinion supported by most members of the group is that the probability of containment failure is reduced because of the expectation of limited melt mass involvement in the explosion and/or low thermal-to-mechanical energy conversion.

This conclusion was reaffirmed at the meeting of the Committee on the Safety of Nuclear Installations (CSNI), "Specialist Meeting on Fuel-Coolant," in January 1993. The conclusion of the meeting was that alpha-mode failure was highly unlikely because of the structures in the lower reactor vessel head. These structures, such as the CRD guide tubes, would limit the melt mass involvement by causing incoherent relocation of the molten corium.

19.2.3.3.6.2 Ex-Vessel Steam Explosion

In SECY-93-087, the staff stated that any dynamic forces resulting from ex-vessel FCI on the integrity of the containment should be evaluated. One of the conditions necessary for an ex-vessel FCI in the ABWR is for the molten corium to be discharged from the reactor vessel into a body of water in the lower drywell. The design of the ABWR containment substantially reduces the probability of a preexisting body of water in the lower drywell at the time of reactor vessel failure.

The reactor vessel skirt is solid, preventing water transfer from the upper drywell to the lower drywell. In addition, there are no active injection systems in the lower drywell, and the passive flooders system does not actuate until the atmosphere within the lower drywell is approximately 260°C (500°F). The connection between the upper drywell and lower drywell is through the vertical connecting vent system, which contains a horizontal 90-degree bend preventing water from reaching the lower drywell.

Severe Accidents

In Section 19EB.1.1 of the ABWR SSAR, GE stated that only 0.3 percent ($5.1\text{E-}10$) of all core damage sequences would result in water in the lower drywell at the time of vessel failure. In Section 19.1.3.5.4 of this report, the staff evaluated GE's estimate and concluded that the probability of a flooded lower drywell at the time of reactor vessel failure is extremely small.

The staff believes that the low likelihood ($5.1\text{E-}10$) of a flooded lower drywell at the time of reactor vessel failure provides a sufficient basis to conclude that the probability of an ex-vessel steam explosion has been reduced to an acceptably low value and is therefore acceptable. Nevertheless, GE and the staff performed analyses to determine the capability of the ABWR containment to withstand ex-vessel steam explosions.

19.2.3.3.6.2.1 GE Analysis

GE provided an uncertainty and sensitivity analysis of the FCI phenomenon in Section 19EB of the ABWR SSAR. This analysis estimated the ability of the ABWR containment, specifically the lower drywell, to withstand a large energetic FCI (steam explosion). The analysis determined the peak pressure the ABWR pedestal is capable of withstanding and the amount of molten corium interacting with water that would be necessary to produce this peak pressure.

GE calculated the peak pressure the pedestal was capable of withstanding during a steam explosion by determining the average pressure of an impulse the amplitude of which can be estimated by the maximum pressure rise expected during an FCI. The ratio of resistance to deformation to the average pressure of an impulse is given by a series of curves. Using this approach, GE calculated the pedestal's resistance to deformation to be 1.7 MPa (246.5 psia).

When an impulse duration of 5 msec is used, which appears to be reasonable based on pulse widths observed during FCI experiments involving corium simulates, the ratio of resistance to deformation to the average pressure of an impulse is approximately 1.0. This implies that the pedestal can withstand a peak pressure of 1.7 MPa (246.5 psia). For additional conservatism, GE eliminated the need for a specific pulse duration by using the curve with the largest ratio of resistance to deformation to the average pressure of an impulse, which is 2.0. This resulted in a peak pressure capability of 0.85 MPa (123 psia).

To determine the amount of corium necessary to cause this peak pressure, GE calculated the steam formation rate assuming a mass of corium fragments into droplets of an identical radius 2.5 mm (.1 in.) and interacts with water.

The mass of corium necessary to produce a peak pressure of 0.85 MPa (123 psia) was 22,400 kg (49,383 lbm), which is approximately 9.5 percent of the entire corium inventory. The peak pressure calculated by GE is at the location of the FCI and does not account for decay of the shock wave as it propagates towards the pedestal wall. GE concluded that the reactor pedestal wall could withstand an FCI involving 9.5 percent of the corium inventory.

19.2.3.3.6.2.2 Staff Analysis

The staff performed an independent assessment of the ABWR containment to withstand a steam explosion using the TEXAS-II computer code. The results of the assessment are documented in report EPRI/NRC 93-203, An Assessment of Ex-Vessel Fuel-Coolant-Interaction Energetics for the General Electric Advanced Boiling Water Reactor (Letter dated March 12, 1993, Richard Borchardt, NRC, to Patrick Marriott, GE).

This assessment evaluated two different possible accident progression sequences, one based on the MAAP code and the other based on the BWR SAR code. The MAAP code scenario used a relatively large (540 kg/sec) (1,190 lbm/sec) corium release composed of a lot of oxides; whereas, the BWR SAR code scenario is more gradual (16.7 kg/sec) (36.8 lbm/sec) and composed of mostly metallics. The results from the MAAP code scenario were identified as being conservative because of the large release rate, whereas the BWR SAR code scenario was identified as being best estimate. The reactor pedestal pressure loads were determined to be approximately 1.1 MPa (160 psia) for the best estimate case and 1.6 MPa (232 psia) for the conservative case. These estimates correspond to local pressure impulse loads of 2.6 kPa-sec (0.38 psia/sec) and 3.7 kPa-sec (0.54 psia/sec).

In a separate analysis, the staff concluded that the pedestal wall could withstand an FCI-generated pressure impulse of 3.7 kPa-sec (0.54 psia/sec) using a ductility ratio of 1.6. It also concluded that the associated radial deflection at this ductility ratio would not compromise the integrity of the reactor vessel and other safety-related piping and equipment.

19.2.3.3.6.3 Basis for Acceptability

Based on the conclusions reached in NUREG-1116 and reaffirmed in the recent CSNI meeting, the staff concludes that in-vessel steam explosions are not a threat to the ABWR containment.

As discussed above in Section 19.2.3.3.6.2, the ABWR containment substantially reduces the probability of a preexisting body of water in the lower drywell at the time

of reactor vessel failure. The staff believes that the low likelihood of a flooded lower drywell at the time of reactor vessel failure provides a sufficient basis to conclude that the probability of an ex-vessel steam explosion has been reduced to an acceptably low value and is therefore acceptable.

19.2.3.3.7 Equipment Survivability

The purpose of this section is to discuss the survivability of equipment, both electrical and mechanical, that is needed to prevent and mitigate the consequences of severe accidents. GE addressed equipment survivability in Section 19E.2.1.2.3 of the ABWR SSAR.

Design bases events are defined as conditions of normal operation, including anticipated operational occurrences, design basis accidents, external events, and natural phenomena for which the plant must be designed. Safety-related equipment, both electrical and mechanical, must perform its safety function during design bases events. Section 3.11 of the ABWR SSAR defines the environmental conditions with respect to limiting design conditions for all safety-related mechanical and electrical equipment. The common terminology used for the level of assurance provided for equipment necessary for design bases events is "environmental qualification" or "equipment qualification."

Beyond design basis events can generally be categorized into in-vessel and ex-vessel severe accidents. The environmental conditions resulting from these events are generally more limiting than those from design bases events. The NRC established a criterion to provide a reasonable level of confidence that the necessary equipment will function in the severe accident environment for the time span for which it is needed. This criterion is commonly referred to as "equipment survivability" and is fundamentally different from equipment qualification.

In its SRM of June 16, 1990, relating to SECY-90-016, the Commission approved the staff position that features provided only (not required for design basis accidents) for severe-accident protection (prevention and mitigation) need not be subject to the 10 CFR 50.49 environmental qualification requirements; 10 CFR Part 50, Appendix B quality assurance requirements; and 10 CFR Part 50, Appendix A redundancy/diversity requirements. The reason for this judgement is that the staff does not believe that severe core damage accidents should be design basis accidents in the traditional sense that DBAs have been treated in the past. Therefore, the staff's proposed applicable regulation for equipment survivability is as follows:

The standard design must include analyses, based on best-available methods, to demonstrate that:

Equipment, both electrical and mechanical, needed to prevent and mitigate the consequences of severe accidents is capable of performing its function for the time period needed in the best-estimate environmental conditions of the severe accident (e.g., pressure, temperature, radiation) in which the equipment is relied upon to function.

Instrumentation needed to monitor plant conditions during a severe accident is capable of performing its function for the time period needed in the best-estimate environmental conditions of the severe accident (e.g., pressure, temperature, radiation) in which the instrumentation is relied upon to function.

19.2.3.3.7.1 In-Vessel Severe Accidents

The applicable criterion for equipment, both mechanical and electrical, required for recovery from in-vessel severe accidents is provided in 10 CFR 50.34(f).

Part 50.34(f)(2)(ix)(C) states that equipment necessary for achieving and maintaining safe shutdown of the plant and maintaining containment integrity will perform its safety function during and after being exposed to the environmental conditions attendant with the release of hydrogen generated by the equivalent of a 100 percent fuel-clad metal-water reaction including the environmental conditions created by activation of the hydrogen control system.

Part 50.34(f)(3)(v) states that systems necessary to ensure containment integrity shall be demonstrated to perform their function under conditions associated with an accident that releases hydrogen generated from 100 percent fuel-clad metal-water reaction.

Part 50.34(f)(2)(xvii) requires instrumentation to measure containment pressure, containment water level, containment hydrogen concentration, containment radiation intensity, and noble gas effluents at all potential accident release points.

Part 50.34(f)(2)(xix) requires instrumentation adequate for monitoring plant conditions following an accident that includes core damage.

These regulations collectively indicate the need to perform a systematic evaluation of all equipment, both electrical and mechanical, and instrumentation to ensure its survivability for intervention into an in-vessel severe

Severe Accidents

accident. GE did not perform this systematic evaluation as of SSAR Amendment 32.

The staff stated in the advance SER that it believed that an acceptable resolution of this issue would entail the following:

- (1) GE should perform an evaluation using best-estimate means of a degraded in-vessel core damage accident that results in a 100 percent metal-water reaction. The basis for the evaluation should be included. The evaluation should identify the most likely sequences resulting in substantial oxidation of the fuel cladding as a result of the probabilistic safety assessment (PSA). An example of an acceptable sequence would involve accident conditions in which ECCS performance is degraded for a sufficient time to cause cladding oxidation but is later recovered to ensure a safe shutdown. If the analysis assumes an intact primary loop, the basis for this should be supported by the results of the PSA (i.e., LOCA does not contribute significantly to core melt). The impact on the reactor system and containment system from the pressure, temperature, and radiation released should be evaluated. As an example, the safe shutdown and containment equipment identified below should be evaluated. Plots showing pressure and temperature as a function of time should be provided.

If the in-vessel severe accident environment has no effect on the equipment performance, this should be clearly indicated along with the supporting rationale. Examples of such instances include cases in which the equipment has already performed its function before the onset of the accident conditions or the equipment is located in an area not exposed to the environmental conditions, such as being located outside the primary containment. For equipment in which environmental conditions as a result of the in-vessel severe accident are in excess of the equipment qualification range, an engineering rationale must be developed as to why the equipment would survive the environment for the needed time span. This rationale could include such factors as limited time period in the environment; the use of similar equipment in commercial industry exposed to the same environment; the use of analytical extrapolations; or the results of tests performed in the nuclear industry or at national laboratories.

An acceptable example using this rationale is the work that GE performed for electrical penetration assemblies in Section 19F.3.2.2 of the SSAR. In

particular, GE referred to experimental tests performed at Sandia National Laboratories on actual electrical penetration assemblies (EPAs) used in operating plants. The tests were performed representative under severe accident conditions with temperatures up to 371 °C (700 °F) and pressures up to 965 kPa (140 psig). Using the results of this work, GE committed to providing EPAs that will maintain leak tightness up to containment pressure of 924 kPa (134 psig) and a temperature of 371 °C (700 °F). The end result of this is that the assumptions used for equipment performance in GE's severe accident evaluation are consistent with the as-built plant.

Safe shutdown equipment that should be addressed include scram equipment, HPCF motor and pump, HPCF isolation valves, HPCF controls, RCIC turbine and pump, RCIC steam valves and cables, RCIC controls, RHR, ADS, shutdown cooling, and others.

Equipment for containment integrity should include containment structure, CIVs - inboard and outboard, electrical penetrations, mechanical penetrations, hatches, sealing mechanisms (welds, bellows, O-ring), as well as others.

- (2) With respect to instrumentation requirements, the staff believes that sufficient instrumentation should exist to inform operators of the status of the reactor and the containment at all times as the in-vessel severe accident is intended to be recoverable from and lead to safe shutdown with containment integrity maintained. The emergency operating procedures (EOPs) direct specific manual operator actions based on instrumentation readings and as such all instrumentation should exist where manual operator actions are specified within the EOPs. As a minimum, the instrumentation identified below should be evaluated.

The instrumentation is designed to survive the environment as specified in RG 1.97. However, RG 1.97 only ensures that the instrumentation will survive in the worst environment resulting from a design bases event and not a severe accident. Therefore, engineering rationale must be developed as to why the instrumentation would survive the environment. This rationale could include such factors as limited time period in the environment; the use of similar equipment in commercial industry exposed to the same environment; the use of analytical extrapolations; or the results of tests

performed in the nuclear industry or at national laboratories.

Instrumentation should include neutron flux, RPV water level, RPV pressure, sup pool temperature, sup pool level, drywell/wetwell (CW/WW) H₂ concentration, DW/WW O₂ concentration, DW temperature, DW pressure, WW pressure, WW temperature, DW water level, among others.

In response to the open item, GE provided the environmental profiles, a table of the necessary equipment, and the accompanying rationale for in-vessel severe accidents in Section 19E.2.1.2.3 of SSAR Amendment 34. The staff finds this information acceptable. Therefore, Open Item F19.2.3.3.7.1-1 is resolved.

19.2.3.3.7.2 Ex-Vessel Severe Accidents

The applicable criteria for equipment, both electrical and mechanical, required to mitigate the consequences of ex-vessel severe accidents is discussed in the Equipment Survivability section of SECY-90-016. This section indicates that features provided only (not required for design basis accidents) for severe-accident protection (prevention and mitigation) need not be subject to the 10 CFR 50.49 environmental qualification requirements; 10 CFR Part 50, Appendix B quality assurance requirements; and 10 CFR Part 50, Appendix A redundancy/diversity requirements. The reason for this judgement is that the staff does not believe that severe core damage accidents should be design basis accidents in the traditional sense that DBAs have been treated in the past.

However, mitigation features must be designed to provide reasonable assurance that they will operate in the severe-accident environment for which they are intended and over the time span for which they are needed. In cases where safety-related equipment (equipment provided for DBAs) is relied upon to cope with severe accident situations, there should be reasonable assurance that this equipment will survive accident conditions for the period that is needed to perform its intended function.

According to SECY-90-016, GE was to review the various severe accident scenarios analyzed and identify the equipment needed to perform various functions during a severe accident and the environmental conditions under which the equipment must function. Equipment survivability expectations under severe accident conditions should include consideration of the circumstances of applicable initiating events (e.g., SBO and earthquakes) and the environment (e.g., pressure, temperature and radiation) in which the equipment is relied upon to function. The staff concluded that GE had not performed

the evaluation as outlined by SECY-90-016 as of Amendment 32 to the SSAR. This was identified as Open Item F19.2.3.3.7.2-1.

As stated in the Advance SER, the staff believed that an acceptable resolution of this issue would entail the following:

- (1) GE should provide an evaluation of the dominant accident sequences identified in Section 19E.2.2 of the SSAR. For each accident sequence, GE should identify the mitigation features. Mitigation features should include ADS, ACIWA, and RCIC as appropriate.

In addition, the specific environment profile (pressure, temperature, radiation fields) should be specified. For each mitigation feature, an assessment of survivability should be done using ground rules similar to those specified above for in-vessel accidents. At least the following mitigation features should be evaluated SRVs, containment structure, vacuum breakers, inboard and outboard CIVs, electrical penetrations, mechanical penetrations, hatches, sealing mechanisms (welds, bellows, O-rings), passive flooders, COPS, COPS CIVs, and others.

- (2) With respect to instrumentation requirements, the staff believes that sufficient instrumentation should exist to inform operators of the status of the containment at all times. This instrumentation should also inform the status of the reactor during the early stages of the accident to ensure reactor failure at low pressure or to allow for low-pressure injection from the ac-independent water addition system.

As a minimum, the list of instrumentation identified below should be evaluated. Where extended ranges of operation of the instrumentation is needed, it should be identified along with the environment to which the instrumentation will be exposed.

The instrumentation is designed to survive the environment as specified in RG 1.97. However, RG 1.97 only ensures that the instrumentation will survive in the worst environment resulting from a design bases event and not from a severe accident. Therefore, engineering rationale must be developed as to why the instrumentation would survive the environment. This rationale could include such factors as limited time period in the environment; the use of similar equipment in commercial industry exposed to the same environment; the use of

Severe Accidents

analytical extrapolations; or the results of tests performed in the nuclear industry or at national laboratories.

At least the following instrumentation should be evaluated: RPV water level, RPV pressure, sup pool temperature, sup pool level, DW/WW H₂ concentration, DW/WW O₂ concentration, DW temperature, WW pressure, WW temperature, and others.

In response to the open item, GE provided the environmental profiles, a table of the necessary equipment, and the accompanying rationale for ex-vessel severe accidents in Section 19E.2.1.2.3 of SSAR Amendment 34. The staff finds this information acceptable. Therefore, Open Item F19.2.3.3.7.2-1 is resolved.

19.2.3.3.7.3 Basis for Acceptability

GE developed a set of curves representing the bounding environmental conditions for both in-vessel and ex-vessel severe accidents. The environmental conditions were then compared to the equipment capabilities to provide a measure of confidence that the necessary equipment would survive the expected conditions. The staff concludes that the systematic process used by GE for assessing equipment survivability is acceptable and consistent with the assumptions used in GE's deterministic severe accident assessment. Further, the staff concludes that this meets the requirements of 10 CFR 50.34 discussed in Section 19.2.3.3.7.1, and the staff's proposed applicable regulation for equipment survivability.

19.2.3.3.8 Protection of Containment Sumps

The lower drywell contains two sumps: an equipment drain sump (EDS) and a floor drain sump (FDS). Figures 1.2-3b and 1.2-13e of the ABWR SSAR indicate that the sumps are embedded in the lower drywell floor with dimensions of approximately 1 m (3 ft) wide, 2 m (7 ft) long, and 1.25 m (4 ft) deep. Figure 1.2-3b of the ABWR SSAR indicates that the lower drywell has approximately 1.6 m (5.75 ft) of concrete protecting the liner, while SSAR Section 6.2.1.1.10.3 indicates that there are 1.5 m (5 ft) of concrete protecting the liner. Therefore, in the sump region, there is approximately 0.25 to 0.35 m (1 ft) of concrete protecting the containment liner. An accumulation of core debris within the sumps could lead to accelerated core-concrete interactions and, given the decreased thickness of concrete protecting the containment liner, the time to liner melt-through in the sump region from core-concrete interactions could be adversely affected.

To prevent liner melt-through in the sump region, the ABWR will have a protective layer of refractory bricks (corium shield) built around each sump to prevent corium ingress. The corium shield design is discussed in Sections 6.2.1.1.10.4 and 19ED of the ABWR SSAR.

19.2.3.3.8.1 Sump Design Criteria

The following general criteria included in Amendment 32, were developed by GE for designing the sumps:

- Corium shield height greater than maximum height of core debris bed
- Melting point of corium shield material above initial contact temperature
- Corium shield material having good chemical resistance to siliceous slags and reducing environments
- Seismic adequacy determined during the detailed design phase
- Shield roofs have provisions to allow water to flow into the sumps when the lower drywell is flooded.
- Shields extend to the floor of the sumps to prevent debris tunneling.

The EDS and FDS have different functions and therefore specific design criteria in addition to the above General Design Criteria (GDC) were developed. The specific design criteria are discussed below.

19.2.3.3.8.1.1 Equipment Drain Sump

The purpose of the EDS is to collect water leaking from valves and piping within the containment. The water enters and exits through piping from above the sump. As such, the following additional design criteria were specified in Amendment 32:

- Solid corium shield, except for the inlet and outlet piping through the roof
- Corium shield walls thick enough to withstand ablation

19.2.3.3.8.1.2 Floor Drain Sump

The purpose of the FDS is to collect water that falls onto the lower drywell floor. The water flows across the drywell floor and runs into the FDS at a height equal to the lower drywell elevation. As such, the following additional design criteria were specified in Amendment 32:

- The corium shield will have channels, which do not face the center of the lower drywell, at the lower drywell elevation to allow for water collection during normal operation.
- The channel length must ensure that debris will freeze before reaching the sump.
- The width and number of channels must ensure the required water flow rate during normal operation.
- Corium shield walls must be thick enough to allow residence time for debris solidification within the channels.
- The corium shield will extend beneath the lower drywell floor.
- The corium shield must be high enough to ensure long-term debris solidification.

19.2.3.3.8.2 Corium Shield Design

In Section 6.2.1.1.10.4.2, GE indicated in Amendment 32 that the corium shields are made of alumina with a height of 0.4 m (1 ft). The shield walls extend to the floor of the sumps.

For the FDS corium shield, GE analyzed in Amendment 32 the ability of the corium shield to initially freeze molten debris as it enters the channels and to transfer sufficient heat so that the debris remains solid in the long term. This analysis was in Sections 19ED.4 and 19ED.5 of the ABWR SSAR. It was used along with the GDC and the specific design criteria specified above to determine the actual corium shield design. The channels in the FDS corium shield are 1 cm (.4 in.) high and at least 0.5 m (1.64 ft) long; however, the width has not been specified.

19.2.3.3.8.3 Discussion

The staff stated in the advance SER that it believes that the sump shield designs proposed by GE have considerable merit and that some conservatism exists in the specified design criteria. For example, the design criteria are intended to ensure that no core debris enters the sumps. However, in actuality, the sumps could withstand limited amounts of core debris. In addition, GE in Amendment 32 did not take into account or factor in flooding the lower drywell with the LDF system or ac-independent water addition system.

Based on engineering judgment, the staff believes that the sump shields would prevent a substantial accumulation of

core debris and that the channels within the FDS would lead to freezing of debris within them. However, the analysis provided to support the proposed shield designs in Amendment 32 was not sufficient to reach this conclusion. In particular, GE did not make use of existing experimental data and analytical tools in justifying its design in SSAR Amendment 33. This was Open Item F19.2.3.3.8.2-1.

In the advance SER, the staff stated that an acceptable resolution to this issue would entail the following:

- (1) GE should evaluate related experimental and analytical work performed in this area to lend additional credibility to its design. In particular, GE should address how the results of its previous work supports its design. This would include a discussion of the prototypicality of the core debris, important parameters, and results. The staff has performed a quick review of related work and tools in this area and believes that they are relevant and readily available.
 - (a) Experiments performed at (1) Kernforschungszentrum Karlsruhe (KfK) on ingress of molten debris into small cracks and openings, (2) Winfrith in the United Kingdom and (3) Grenoble in France.
 - (b) Analytical tools such as PLUGM computer code (NUREG/CR-3190) and BUCOGEL computer code developed in France.
 - (c) Work performed in the forging and casting industries.
- (2) The analysis performed by GE in Amendment 32 for sizing the FDS evaluated an oxidic melt of around 2,227 °C (4,040 °F) and a eutectic melt of around 1,427 °C (2,600 °F). However, GE used the same correlations and key parameters for both, such as thermal conductivity and latent heat of fusion. To account for uncertainty in the progression of a severe accident and a range of material properties (such as density, melting point, and thermal conductivity), GE should perform separate analyses for oxidic, metallic, and eutectic melts clearly identifying the material properties and providing suitable references. In addition, GE should identify the parameters that the shields are most sensitive to (e.g., freezing point, heat of fusion, velocity of debris in channel, atmosphere temperature, and melt superheat). GE can use the results of its MAAP runs to identify the core debris composition at the time it enters the lower drywell.

Severe Accidents

In addition, GE could use the results of other code predictions (BWRSAR, MELCOR) as documented in NUREGs for similar BWRs.

- (3) GE should address why the velocity of the debris in the FDS channels is not affected by the initial velocity of debris falling from the reactor pressure vessel (RPV)
- (4) GE should modify the design criteria to do the following:
 - (a) Specify that the EDS extend below the lower drywell floor and that both shields prevent tunneling of core debris under them.
 - (b) Specify sloping of the shield roof to prevent accumulation of core debris or show that the long-term debris solidification in the channels is not affected by minor amounts of debris on the roof.
- (5) GE should provide the thickness of the EDS corium shield necessary to withstand ablation.

GE has addressed these 5 issues in an SSAR markup of Sections 6.2.1.1.10.4 and Appendix 19ED dated February 7, 1994. Specifically, GE provided a markup of SSAR Section 19ED.6 providing an overview of related experimental and analytical work concerning the freezing of molten fuel in narrow channels to address Item 1. For Item 2, GE modified the analysis of channel length in Subsection 19ED.4 to account for three debris scenarios covering the expected range of melt phenomena. For Item 3, GE clarified the FDS shield design to clarify the location of channels. For Item 4, GE modified Subsections 19ED.3 and 19ED.5.2 and added 19ED.5.3 to establish that the EDS and FDS shields extend to the sump floor to prevent debris tunneling. Further, GE modified its long-term analysis in Subsection 19ED.5.1 to credit flooding of the lower drywell. Lastly, for Item 5, GE added Subsection 19ED.5.3 to address the thickness of the EDS shield walls and the shield wall of the FDS without channels. In addition, GE modified its general and specific design criteria for the EDS and FDS. This resulted in a revision to the design dimensions. Further, GE takes credit for the lower drywell flooders in determining the sump shield design. The staff concludes that the sump shield design proposed by GE is acceptable. This is based on GE's development of design criteria, proposed analytical solution, evaluation of the shield design to variations in key parameters, and review of existing related experimental and analytical work. GE has included the above information in SSAR Amendment 34 and the

staff finds it to be acceptable. Therefore, Open Item F19.2.3.3.8.3-1 is resolved.

19.2.4 Containment Performance

The NRC approach for ensuring containment survivability from severe accident challenges consists of requiring inclusion of accident prevention and consequence mitigation features and the containment performance goal (CPG). The CPG ensures that the containment would perform its function in the face of most severe accident challenges and that the design (including its mitigation features) would be adequate if called upon to mitigate a severe accident.

Two alternative CPGs were identified in SECY-90-016: a conditional containment failure probability (CCFP) of 0.1 or a deterministic containment performance goal that offers comparable protection. In its June 26, 1990, SRM, the Commission approved the use of a 0.1 CCFP as a basis for establishing regulatory guidance for evolutionary LWRs. Two definitions of containment failure were discussed in SECY-91-309, "Draft Safety Evaluation Report on the General Electric Boiling Water Reactor Design Covering Chapter 19 of the Standard Safety Analysis Report, Response to Severe Accident Policy Statement." These include a CCFP based on a structural integrity definition and on a dose definition. For internal events, Section 19.1.3.5.1 of this report provides the results of the CCFP analyses and concludes that the ABWR design limits the CCFP to approximately 0.1. The treatment of external events for the ABWR is discussed in Section 19.1.2 of this report.

The Commission directed that the use of a 0.1 CCFP should not be imposed as a requirement, and that the use of the CCFP should not discourage accident prevention. Therefore, the staff's proposed applicable regulation for containment performance is as follows:

The standard design must include design features to limit the conditional containment failure probability for the more likely severe accident challenges.

Section 19.1.3.5 of this report provides the staff's analysis of the design features that contribute to limiting the CCFP. The severe accident phenomena that are mitigated by these design features are evaluated in Sections 19.2.3.3 and 19.2.6 of this report. Based on the evaluations in these sections, the staff concludes that the acceptance criteria in SECY-90-016, SECY-93-087, and the staff's proposed applicable regulation for containment performance have been met.

19.2.5 Accident Management

The staff concluded, based on PRAs and severe accident analyses, that the risk associated with severe accidents could be further reduced through effective accident management (AM). AM encompasses those actions taken during the course of an accident by the plant operating and technical staff to (1) prevent core damage, (2) terminate the progress of core damage if it begins and retain the core within the reactor vessel, (3) maintain containment integrity as long as possible, and (4) minimize offsite releases. AM, in effect, extends the defense-in-depth principle to plant operating staff by extending the operating procedures well beyond the plant design basis into severe fuel damage regimes, using existing plant equipment and operator skills and creativity to terminate severe accidents and limit offsite releases.

In SECY-88-147 and Generic Letter 88-20, the staff identified the development of an "accident management plan" by each operating reactor licensee for severe accident "closure." A comprehensive description of the major goals, framework, and elements of an AM plan was subsequently provided in SECY-89-012, Staff Plans for Accident Management Regulatory and Research Programs. The AM plan provides a framework for evaluating information on severe accidents, for preparing and implementing severe accident operating procedures, and for training operators and managers in these procedures. An effective AM plan could reduce the risk associated with severe accidents by incorporating improvements in five general areas:

- Accident Management Strategies and Implementing Procedures
- Training in Severe Accidents
- Guidance and Computational Aids for Technical Support
- Instrumentation
- Delineation of Decision Making Responsibilities

In response, the nuclear industry has initiated an AM program, as described in SECY-90-313, Status of Accident Management Program and Plans for Implementation. Key issues to be resolved in establishing the AM program include industry completion and NRC review of (1) the industry-proposed process for evaluating AM capabilities, and (2) vendor-specific AM guidance. The industry-proposed AM program was scheduled for completion in 1993.

The overall responsibility for AM, including development, implementation, and maintenance of the AM plan, lies with the COL applicant, since the applicant is ultimately responsible for the safety of the plant and for establishing

and maintaining an emergency response organization capable of effectively responding to potential accident situations.

The COL applicant should submit the AM plan as part of the COL application. The plan should provide the applicant's commitment to perform a systematic evaluation of the plant's ability to deal with potential severe accidents and to implement the necessary enhancements within the detailed plant design and organization. The staff will review the AM plan at the COL stage to assure that the evaluation process and commitments proposed by the COL applicant provide an acceptable means of systematically assessing, enhancing, and maintaining AM capabilities, consistent with staff expectations. The COL applicant would later implement the plan and submit the results for staff review before plant operation.

19.2.6 Capacity of the ABWR Primary Containmentment Vessel

19.2.6.1 Introduction

In Appendix 19F to Chapter 19 of the ABWR SSAR, the applicant (GE) discussed the ultimate capacity of its primary containment. The pressure boundary of the containment consists of the reinforced concrete containment vessel (RCCV) and steel torispherical upper drywell head (STUDH). The staff's evaluation of the adequacy of the containment to withstand the postulated design basis loads is provided in Sections 3.8.1 and 3.8.2 of this report. The purpose of this evaluation was to assess the containment's capability beyond the design basis.

19.2.6.2 Evaluation

The ABWR's containment consists of an RCCV, a right cylindrical structure built of steel-lined reinforced concrete, and a STUDH. In order to establish the ultimate capacity of the containment, GE discussed in Appendix 19F of its SSAR (Reference (Ref.) 1) the analyses performed for the RCCV. It also provided the calculated capacity of the STUDH and related the potential leak path under pressures and temperatures that could represent the environment inside the containment during severe accident conditions. This evaluation was based on a review of the information provided in Reference 1.

GE arrived at the following conclusions regarding the structural capability and functionality of the containment structure:

- (1) The STUDH of the containment will have an internal pressure capacity of 770.1 kPa (97 psig) at

Severe Accidents

260.0 °C (500 °F) when the allowable stresses in the steel drywell head are held to the Level C Service Limit of the ASME Code Section III, Subarticle NE-3220 (Ref. 1, Section 19F.3.1.2).

- (2) The pressure capacity of the STUDH at the median fragility (as defined by the conditional failure probability at 50 percent confidence level) is determined to be 1025.3 kPa (134 psig) at 260.0 °C (500 °F), and 928.7 kPa (120 psig) at 371.1 °C (700 °F) (Ref. 1, Section 19F.3.1.2).
- (3) The equivalent ASME Level C Service Limit of the RCCV is 1232.1 kPa (164 psig); the ultimate capacity of the RCCV, being higher, is not estimated by the applicant (Ref. 1, Sections 19F.1, 19F.2, and 19F.3.1.1).
- (4) Liner plate and its anchorages will maintain their integrity when subjected to a severe accident pressure of 1025.3 kPa (134 psig) and a temperature of 260.0 °C (500 °F) (Ref. 1, Section 19F.3.2.1).
- (5) The total leak area through various penetrations at 928.7 kPa (120 psig) is estimated to be 67.55 cm² (10.47 in²) (Ref. 1, Section 19F.3.2.2).

The acceptability of GE's five conclusions is discussed and evaluated by the staff in the following sections.

19.2.6.2.1 Level C Service Limit

19.2.6.2.1.1 Steel Torispherical Upper Drywell Head

Membrane Stress Intensity

On the basis of NASTRAN analysis and the stress intensity criterion provided in the ASME Code, Section III, Paragraph NE-3221, GE stated that the general primary membrane stress controls the design at the Level C Service Limit for an internal pressure of 832.2 kPa (106 psig) at 171.1 °C (340 °F). At 260.0 °C (500 °F), the allowable internal pressure at the Level C Service Limit for membrane stress intensity was found to be 770.1 kPa (97 psig). To verify GE's results concerning the stresses using NASTRAN computer code, the staff independently calculated the membrane stresses using ALGOR computer code. The staff's comparison (Table 19.2-1) confirmed the applicant's conclusion.

Buckling

Galletly (Ref. 2) developed, on the basis of the actual test data, a simple parametric equation to calculate limiting internal pressure that would prevent buckling of steel torispherical heads. Based on the equation and the analysis of the data in Reference 2, GE calculated a best-estimate internal pressure value of 1838.8 kPa (252 psig) and a lower bound value of 1245.9 kPa (166 psig) as values corresponding to buckling failure values of torispherical heads tested.

When test data are used to establish the ASME Code Section III, NE-3222, Allowable Buckling Stress, it is appropriate to use the best estimate test data. Accordingly, when the best-estimate test data values are used in NE-3222, the allowable internal pressure at the Level C Service Limit for buckling stress is 797.7 kPa (101 psig). An alternate method of computing the allowable buckling stress is provided in ASME Code, Section III, Code Case N-284. The factor of safety at the Level C Service Limit in Code Case N-284 (1.67) is less than that in NE-3222 (2.5). However, the factor of safety has to be applied to the lower-bound test data. When the lower-bound test data are used in Code Case N-284, the allowable internal pressure at the Level C Service Limit is 783.9 kPa (99 psig). This evaluation is based on the staff's position on the shell buckling as a result of internal pressure (See Appendix A to Section 3.8.1 of this report).

From the above three limiting pressure values of 770.1 kPa (97 psig) based on membrane stress intensity criterion, 797.7 kPa (101 psig) from buckling criterion of NE-3222, and 783.9 kPa (99 psig) from buckling criterion of N-284, the staff finds that because the 770.1 kPa (97 psig) is the least of the pressure values, it controls the internal pressure capacity of STUDH corresponding to the ASME Level C Service Limit stress criterion. Thus, the staff finds the proposed internal pressure of 770.1 kPa (97 psig) as an acceptable value for the STUDH.

19.2.6.2.1.2 Concrete

For the concrete portion of the RCCV, the staff considers the factored load acceptance standards of ASME Code Section III, Division II, Article CC-3000 as an appropriate acceptable criterion. It should be noted that in applying the criterion, a factor of 1.0 (instead of 1.5) should be Table CC-3230-1 for the internal pressure generated by severe accidents. The pressure capability estimate for the RCCV using this criterion is discussed in the following paragraphs.

Table 19.2-1 GE ABWR drywell head comparison of membrane stresses

75 psig internal pressure, 0.03175 m (1.25 in.) thickness, 171 °C (340 °F)

	Hoop MPa (psi)	Longitudinal MPa (psi)	Intensity MPa (psi)	Remarks
NASTRAN	101.5 (-14,725)	59.57 (8,640)	161.1 (23,365)	5° wedge
ALGOR ¹	101.0 (-14,650)	58.05 (8,420)	159.1 (23,070)	5° wedge 1 st quadrant
ALGOR	92.91 (-13,475)	59.50 (8,630)	152.4 (22,105)	10° wedge Full model

From NASTRAN,

Stress intensity, P_m : 161.1 MPa (23,365 psi) @ 0.618 MPa (75 psig) and 171 °C (340 °F)Yield strength (S_y) (SA-516, Gr. 70): 229.7 MPa (33,300 psi) @ 171 °C (340 °F)

211.7 MPa (30,700 psi) @ 260 °C (500 °F)

Allowable stress intensity (S_{mc}) (SA-516, Gr. 70): 133.1 MPa (19,300 psi)Level C Service Limit Criteria (NE-3221): $P_m \leq \max. (1.0 S_y, 1.2 S_{mc}),$ $P_m \leq 229.7 \text{ MPa (33,300 psi) @ 171 °C (340 °F)}$ $P_m \leq 211.7 \text{ MPa (30,700 psi) @ 260 °C (500 °F)}$

Allowable pressure = $(75) \cdot (33,300) / (23,365)$
 = 0.832 MPa (106 psig) @ 171 °C (340 °F)
 = $(106) \cdot (30,700) / (33,300)$
 = 0.770 MPa (97 psig) @ 260 °C (500 °F)

¹ ALGOR is a three-dimensional finite element program for structural analysis and design and has been used by the staff for its independent evaluation.

Severe Accidents

Initially, in Amendment 21 of Ref. 1, the applicant provided the information related to the RCCV ultimate capacity based on the scaled model test of the RCCV. Later, in Amendment 30 of Ref. 1, to avoid inclusion of proprietary information in the SSAR and recognizing the differences in the model and the proposed RCCV (having a significantly higher percentage of reinforcement) GE decided to perform analyses of the RCCV for severe accident conditions. The FINEL computer program developed by Bechtel Corporation is used for the nonlinear finite element analysis of the RCCV. In the analysis, the RCCV and the internal structures are considered as axisymmetric, and the reinforced concrete girders integral with the top slab of the RCCV are represented by solid elements with appropriate stiffness. The pressure capacity of the top slab is calculated based on the extrapolation of the elastic 3-D STARDYNE analysis results.

Within the elastic range of response, both codes have been well recognized and benchmarked against the structural integrity tests of containments. Though the FINEL computer program permits the specification of bilinear, brittle, and ductile material properties, it is not benchmarked against results of any containment test with nonlinear responses. However, within the pressure range of interest, the RCCV responses are in the linear range and hence the use of these codes to compute pressure capacities of the RCCV is reasonable.

The results of the combined analyses indicate that the weakest link in the RCCV is the top slab where the equivalent ASME Level C Service Limit acceptance criterion is reached at a pressure of 1232.1 kPa (164 psig).

Thus, the staff finds the containment performance under an internal pressure of 770.1 kPa (97 psig) to be acceptable and is aware that it is limited by the capacity of the STUDH. The RCCV has higher capacity and does not limit the containment performance.

19.2.6.2.2 Median Fragility Level for the Containment

19.2.6.2.2.1 Steel Torispherical Upper Drywell Head

GE stated that the limit pressure for plastic deformation was found to be 940.4 kPa (121.7 psig) at 260.0 °C (500 °F) by Shield and Drucker's proposed equation (Ref. 3). The minimum yield strength of material SA-516, Gr. 70, as specified in Appendix I of ASME Section III was increased by 10 percent for the realistic estimate of the structural strength. Thus, the limiting internal pressure is determined to be 1025.3 kPa (134 psig) at 260.0 °C (500 °F) and 928.7 kPa (120 psig) at 371.1 °C (700 °F) to account for the lower yield strength of the material.

From the STUDH buckling capability, the applicant predicted a best-estimate internal pressure value of 1838.8 kPa (252 psig) and a lower bound value of 1245.9 kPa (166 psig) as values corresponding to buckling failure of the STUDH. These values are based on the results of the tests (Ref. 2) performed on stainless steel and carbon steel torispherical heads fabricated using the pressed and spun (PS) technique as well as the crown and segment technique (CS). Out of 43 tests, six data points were from the actual failure resulting from internal pressure on the carbon steel CS heads. These are the most relevant data for the drywell head. The lower-bound value is estimated to be 1245.9 kPa (166 psig). As these values are based on a limited data base and as they correspond to actual buckling failure of the heads, use of them to arrive at the actual median fragility internal pressure for the STUDH should be made with some margin on the lower bounds. Providing an arbitrary knockdown factor of 1.2, the internal buckling pressure of the median fragility level can be derived as 1052.8 kPa (138 psig). However, GE adopts 1025.3 kPa (134 psig) as the capability pressure and uses it as the median fragility value at 260.0 °C (500 °F). Based on this median fragility value of 1025.3 kPa (134 psig), a containment pressure capacity fragility curve was provided with the following uncertainty parameters in Appendix 19FA to Reference 1.

In Appendix 19FA to Reference 1, the applicant estimated the uncertainties associated with the median fragility value using engineering judgment and the results from prior analysis. The uncertainties in the prediction of the failure pressure generally result from uncertainties in modeling and material strength. The lognormal distribution is selected to characterize the fragility curve and defined in terms of the median pressure capacity and the combined logarithmic standard deviation. The logarithmic standard deviations from uncertainties for modeling and material properties of steel structures (β_m and β_s) are estimated as 0.14 and 0.08, respectively, and the combined logarithmic standard deviation (β_c) is estimated to be 0.16.

The use of 0.14 for β_m is acceptable because it is consistent with Reference 4 in which the coefficient of variation (COV) [defined as $(\exp(\beta^2)-1)^{1/2}$] associated with the modeling error by the use of approximate methods including torispherical heads is 0.12.

The use of 0.08 for β_s is acceptable because the variability associated with material strength is expected to be the same regardless of temperature (Ref. 5). The statistical data for SA-516, Gr. 70, show that the average yield strength of the material is 48.62 ksi and the standard deviation is 3.525 ksi (Ref. 6). The COV is 0.073, which is less than 0.08 used for β_s .

Therefore, the combined logarithmic standard deviation (β_c) of $0.16 ((\beta_m^2 + \beta_s^2)^{1/2})$ is acceptable.

Based on the staff's evaluation and by taking the above factors into account, the staff considers the median fragility value for the STUDH of 1025.3 kPa (134 psig) at 260.0 °C (500 °F) with the combined logarithmic standard deviation of 0.16 to be reasonable. If the temperature reaches 371.1 °C (700 °F), the median fragility internal pressure value is reduced to 928.7 kPa (120 psig).

19.2.6.2.2.2 Concrete

As discussed in A.2 above, the equivalent Level C Service Limit internal pressure capacity of the RCCV is estimated as 1130 kPa (164 psig). As this capacity is higher than the median fragility internal pressure capacity of the STUDH, GE decided not to spend resources in estimating the ultimate (median fragility) pressure capacity value for the RCCV. The staff considered this approach acceptable provided GE demonstrated the integrity of the RCCV liner plate and its anchorages under 1025.3 kPa (134 psig) and 260.0 °C (500 °F) temperature.

In Section 19F.3.2.1 (Ref. 1), GE provided an evaluation of the liner plate and its anchorages under the specified conditions. The applicant determined the maximum hoop strain in the liner plate at 1025.3 kPa (134 psig) to be 0.13 percent. With a strain concentration factor of 33 to account for a failure mode similar to that in the Sandia test (Ref. 7), GE uses the argument that the resulting strain in the ABWR RCCV is 4.3 percent, which is significantly lower than the ultimate tensile strain (i.e., 21 percent) of the liner plate material. The staff recognizes that the geometry and type of anchors to be used for the ABWR RCCV are quite different and the strain concentration factors may be different between the Sandia test and the ABWR. Nevertheless, a factor of 33 is conservative and the staff finds it acceptable.

Additionally, GE evaluated the effects of thermal loading on the liner and the anchorages using Bechtel's topical report, BC-TOP-1, Rev. 1, Containment Building Liner Plate Design Report. For this evaluation, the temperature inside the RCCV is considered as 260.0 °C (500 °F), and that outside the RCCV is assumed as 37.8 °C (100 °F). The liner is shown to be buckled (because of temperature-induced compressive stresses) between the continuous anchorages, but the force on the anchorages is shown to be less than its yield capacity.

Although the staff finds this approach reasonable, it should be recognized that there is a difference in the duration and

magnitude of the temperature loading that would be applied during a design-basis accident (DBA) (which is of a smaller and magnitude shorter duration) compared with that postulated to occur during a controlling severe accident (SA) scenario. Also, the condition of the RCCV structure would be different in the two cases. The pressure loading during an SA would reduce the stiffness of the structure considerably. Thus, the assumptions used in the BC-TOP-1 report are not necessarily applicable. However, they provided a basis for calculating an upper-bound value for compressive buckling of the plate and for the forces on the liner anchorages. Alternatively, under an SA, the liner together with the extensively cracked concrete can be considered to grow because of high (260.0 °C (500 °F)) temperature. Considering the differential temperature of 204.4 °C (400 °F), the staff calculated the tensile strain in the liner to be 0.24 percent. The combined tensile strain in the general shell liner could be as high as 0.37 percent. When the same strain concentration factor of 33 is used again, the maximum tensile strain in the liner at a discontinuity could be as high as 7.9 percent. This strain is 40 percent of the ultimate tensile strain of the liner material. Under this condition, the continuous anchorages could detach themselves from the cracked concrete, but, the liner would still provide the required leaktightness.

Thus, the staff finds the median fragility internal value of 1025.3 kPa (134 psig) at 260.0 °C (500 °F) and 928.7 kPa (120 psig) at 371.1 °C (700 °F) acceptable as limiting values.

19.2.6.2.3 Interface Between STUDH and RCCV

GE performed the pressure capacity calculations for the interface between the STUDH flanges and the top slab of the RCCV and arrived at the following pressure capacities: (a) 1176.9 kPa (156 psig) for the maximum allowable concrete peripheral shear stress of the top slab using the acceptance criteria of ASME Section III, Division II, CC-3421.6, (b) 1687.1 kPa (230 psig) for anchorage steel ring plate using the Level C acceptance criteria of ASME Section III, Division I, NE-3221, and (c) 1528.5 kPa (207 psig) for the anchorage steel gusset plates using the Level C acceptance criteria of ASME Section III, Division I, NE-3227.2. The staff finds the pressures to be above the median fragility value of 1025.3 kPa (134 psig), with the stresses in the interface area well within the ASME Section III Code allowables, and thus acceptable. Therefore, the staff agrees with GE's conclusion that the probable leakage path would be through the STUDH flanges. The leakage through the STUDH flanges is discussed in Section 2D of this report.

Severe Accidents

19.2.6.2.4 Leakage Through Penetrations

In Section 19F.3.2.2 (Ref. 1), GE discusses the leakages through various RCCV penetrations and from the seal area between the flanges in the STUDH under severe accident temperatures and pressures. The assumptions and rationale used in the discussions by the applicant are based upon the experimental work performed at SNL and Argonne National Laboratory (ANL). GE determined that under the postulated SA scenarios, the leakage through the fixed electrical and mechanical penetrations was negligible.

In a facsimile dated September 7, 1993, GE revised Subsection 19F.3.2.2 of the SSAR to discuss the leakage performance of the containment EPAs under conditions. GE stated that the EPAs to be used in the ABWR containment would be capable of maintaining leaktightness up to the containment pressure of 1025.3 kPa (134 psig) at 371.1 °C (700 °F). The staff finds GE's assessment regarding the fixed electrical and mechanical penetrations acceptable. GE has included this information in Subsection 19F.3.2.2 of the SSAR.

The containment has five operable penetrations: two pressure-seating airlocks, two pressure unseating equipment hatches, and the pressure-unseating STUDH. In Section 19F.3.2.2 of Ref. 1, GE estimated the leakage areas from the three pressure-unseating penetrations with conservative assumptions; such as, (1) the seal is assumed lost at 260.0 °C (500 °F), and (2) the springback capability of degraded seals was not factored in. The total leakage area is estimated as 110 percent of the leakage areas of the three pressure-unseating penetrations to account for the potential for small leakages through the airlocks. The leakage areas are provided for the pressures above the structural integrity test pressure at an interval of 170.3 kPa (10 psig). The total leakage area at 928.7 kPa (120 psig) is estimated as 67.5 cm² (10.47 in²).

The staff considers the estimate of these leakage areas to be reasonable.

19.2.6.3 Summary and Conclusion

Based on the review of the information GE provided and the staff's evaluation as discussed above, the staff concludes the following:

- (1) The staff evaluation indicates that the ABWR containment structure has an internal pressure capacity of 770.1 kPa (97 psig) and 260.0 °C (500 °F) when the allowable stresses in the steel drywell head are held to the Level C Service Limit.

- (2) As discussed in 2B of the evaluation, the staff considers the median fragility of the containment structure of 1025.3 kPa (134 psig) at 260.0 °C (500 °F) to be a conservative value.

References:

1. Appendix 19F to Chapter 19 of the General Electric Nuclear Energy, Advance Boiling Water Reactor Standard Safety Analysis Report (up to and including Amendment 34).
2. Galletly, G. D. "A Simple Design Equation for Preventing Buckling in Fabricated Torispherical Shells Under Internal Pressure," ASME Journal of Pressure Vessel Technology, Vol. 108, November 1986.
3. Shield, R. D., Drucker, D. C. "Design of Thin-Walled Torispherical and Toriconical Pressure-Vessel Heads," Transactions of ASME, June 1961.
4. NUREG/CR-2442, "Reliability Analysis of Steel-Containment Strength," June 1982.
5. NUREG/CR-5405, "Analysis of Shell-Rupture Failure Due to Hypothetical Elevated-Temperature Pressurization of the Sequoyah Unit 1 Steel Containment Building," February 1990.
6. NUREG/CR-2137, "Realistic Design Margins of Pumps, Valves, and Piping," June, 1981.
7. NUREG/CR-5341, "Round-Robin Analysis of the Behavior of a 1:6-Scale Reinforced Concrete Containment Model Pressurized to Failure: Posttest Evaluations," Sandia National Laboratory, October 1989.

19.3 Shutdown Risk

19.3.1 Introduction

Various incidents occurring at nuclear power plants during low power and shutdown operation modes over the past several years have raised NRC staff concerns regarding plant vulnerability during these operating modes. The shutdown events have caused plants to lose their ability to maintain core cooling, provide make-up coolant to the reactor, and to maintain electrical power to essential equipment. The April 10, 1987, event at the Diablo Canyon Nuclear Power Plant emphasized the sensitivity of operating a pressurized-water reactor (PWR) with a reduced inventory in the RCS. Following an evaluation, the staff issued Generic Letter 88-17, "Loss of Heat Decay Removal," on October 14, 1988, requesting PWR licensees

to address numerous generic deficiencies to improve safety during operation at reduced inventory.

On March 20, 1990, Vogtle Unit 1 experienced a complete loss of decay heat removal (DHR) capability during shutdown and refueling operations from a loss of the required offsite ac source and failure of the available onsite diesel generator to provide and maintain power to safety-related buses. In June 1990, the staff issued NUREG-1410, Loss of Vital AC Power and the Residual Heat Removal System During Mid-Loop Operations at Vogtle Unit 1 on March 20, 1990. In the NUREG, the staff discussed the loss of vital ac power and DHR capability during midloop operations and the need to manage risk for shutdown operations.

These events, as well as others, prompted the staff to begin a comprehensive review of low-power and shutdown operations including hot shutdown, cold shutdown, and refueling at all nuclear plants and other shutdown-related issues identified by foreign regulatory organizations and the NRC. The objective of the review was to assess risk during shutdown, refueling and low-power operation. In February 1992, the staff issued Draft NUREG-1449, Shutdown and Low Power Operation at Commercial Nuclear Power Plants in the United States, for public comment to document technical findings associated with shutdown conditions. In September 1993, the staff issued NUREG-1449 final report, which incorporated and responded to comments received on draft NUREG-1449. The staff is using these technical findings to prepare appropriate regulatory actions to address shutdown risk.

The fundamental conclusion of NUREG-1449 is that public health and safety have been adequately protected while plants were in shutdown conditions. However, numerous and significant events have indicated that substantial safety improvements should be made in the areas of:

- outage planning and control
- fire protection
- TS
- instrumentation.

The staff utilized the technical findings and insights from NUREG-1449 in its safety evaluation of the ABWR.

19.3.2 Evaluation Scope

On January 12, 1990, the Office of the Secretary of the Commission issued SECY-90-016, Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements. In this Commission paper, the staff identified reduced inventory operation and a number of other issues significant to

reactor safety that were considered fundamental to the agency's decisions on the acceptability of evolutionary ALWR designs. On July 12, 1993, the staff submitted SECY-93-190, Regulatory Approach To Shutdown and Low-Power Operations, in which it discussed the advantages and disadvantages of a proposed rulemaking to establish new regulatory requirements for shutdown and low-power operations in the following areas: outage planning and control, TS, fire protection, and instrumentation. The proposed new regulatory requirements, which will be developed further as part of the shutdown risk rulemaking process, were a result of technical findings and insights from NUREG-1449.

Based on the above, the NRC requested GE, as part of the design review process for the ABWR, to perform a systematic examination of shutdown risk, including evaluation of specific ABWR design features that minimize shutdown risk, quantification of the reliability of the DHR systems, identification of any vulnerabilities introduced by new design features and consideration of fires and floods with the plant in modes other than full power.

GE submitted its Shutdown Risk Evaluation on July 2, 1992, for staff review. In this report, GE stated that a total of 70 loss-of-power and over 100 loss-of-DHR events that occurred at operating BWRs were reviewed and evaluated from NUREG-1410, NSAC-88, Residual Heat Removal Experience Review and Safety Analysis — Boiling Water Reactor, Institute of Nuclear Power Operation (INPO) reports, and NRC information notices. GE stated that the ABWR design features would prevent or mitigate the most safety significant of these events. GE also addressed what it believed to be all shutdown risk issues, concerns, and design features to minimize risk associated with low-power and shutdown operations.

GE evaluated the ABWR design for risks associated with plant conditions in Modes 3 (hot shutdown), 4 (cold shutdown), and 5 (refueling), as well as areas that are considered operational improvements for shutdown risk operation which will be further addressed by COL holders implementing guidance provided by the reactor vendor (e.g., outage planning and control and operator training and procedures). Other shutdown risk concerns, such as DHR capability during low-power and shutdown operation, are addressed by specific design features and TS.

The staff evaluated this submittal based on technical findings and insights from NUREG-1449, a number of studies from the international community, and PRA of shutdown and low-power operating modes for a BWR to screen for important accident sequences. The staff also considered reports on operating plant events associated

Severe Accidents

with shutdown operation and staff-sponsored research issues associated with low-power and shutdown operation.

The purpose of the staff review is to ensure that the ABWR design has appropriately addressed the shutdown risk concerns based on experience with operating plants, including appropriate vendor guidance for COL applicants in areas of outage planning and control, fire protection, and instrumentation. Design improvements and/or design modifications GE identified were reviewed to ensure insights from shutdown operation experiences were addressed and that the design improvements reduce the likelihood of core damage and enhance public health and safety. Also, the staff evaluated vulnerabilities that may result from new design features; DHR capability; treatment of fires and floods with plant in modes other than full power; and related technical findings discussed in NUREG-1449.

Therefore, the staff's proposed applicable regulation for shutdown risk is as follows:

The application for design certification must include a systematic examination of shutdown risk including an assessment of:

- (1) specific design features that minimize shutdown risk;
- (2) the reliability of decay heat removal systems;
- (3) vulnerabilities introduced by new design features; and
- (4) fires and floods with the plant in modes other than full power.

These items are discussed in the sections below.

19.3.3 ABWR Design Features Minimizing Shutdown Risk

GE stated that the risks associated with shutdown operations have been reduced in the ABWR design by such features as the enhanced DHR system, improved electrical systems, instruments that give important safety parameters during shutdown, and alternate features that maintain core cooling in case of a loss of DHR. The applicant concluded that ABWR TS and utility operating and maintenance procedures will ensure that, during shutdown conditions, the ABWR is adequately protected against accidents.

19.3.3.1 Decay Heat-Removal Capability

In NUREG-1449, the staff stated that DHR can be lost at shutdown because of a loss of flow in the RHR system or a loss of an intermediate or ultimate heat sink caused by the loss of electric power or by valve failures. Events

during shutdown can lead to fuel's being uncovered and damaged. Past events have led to interrupted shutdown cooling (SDC) at BWR operating plants because of a loss of power to the RPS logic that caused the RHR system isolation valves to fail closed.

The applicant described the ABWR DHR capability in SSAR Section 19.Q.4.1 of the ABWR PRA Shutdown Risk Final Report. The DHR capability consists of several features that minimize the loss of DHR. One of these features consists of the three independent divisions of RHR systems and is the first line of defense in maintaining DHR. Three suction lines for shutdown cooling will be connected directly to the reactor pressure vessel (RPV). This is an improvement over the current BWR designs in which one common shutdown cooling line draws suction from the external reactor recirculation loops. This single suction line in current BWRs is more vulnerable to a loss of shutdown cooling by single failure of valves in the suction line. The RHR shutdown cooling return for RHR subsystem A will be routed to the RPV vessel through the feedwater system. RHR shutdown cooling return for RHR subsystems B and C will be routed directly to the RPV. If a single failure occurred in the operating RHR loop while a second loop was in maintenance, the third loop could be placed in service. Under certain conditions, all three RHR loops or any two RHR loops would be run in parallel. In these cases, failure of one loop would not result in a loss of RHR. The ABWR DHR systems include a mode selector control for choosing among five modes of operation: low pressure flooding, suppression pool cooling, shutdown cooling, wetwell spray, and drywell spray. If any one of these modes is selected, the DHR systems will automatically align the valves as required for the mode selected. This feature will reduce the chance of operator error from incorrectly aligning the required valves, thus increasing the availability of the DHR capability during shutdown operations.

The applicant indicated that a loss of power to the RPS will not result in isolation of the shutdown cooling (SDC) system. A loss of power to the multiplexed safety system logic would cause each SDC isolation valve to fail in its current position (fail as-is), thus preventing closure of the CIVs between the RHR system and the RCS. Shutdown cooling is thereby maintained.

Upon receiving a low RPV water-level signal, RHR shutdown cooling isolation valves will close to stop all flow out of the RPV. The RPV low-level set point is 3.18 m (10.43 ft) above the top of the active fuel. If the RPV low-level isolation feature fails, inventory loss resulting from flow out of the RPV will stop when the RHR suction cooling nozzle is uncovered. At this point, 1.7 m (5.58 ft) of water will remain above the top of the

active fuel. The applicant will design two-out-of-four logic to control the initiation of ABWR RPV water-level signals. This will reduce the likelihood of a false loss-of-coolant accident (LOCA) signal.

The staff finds these provisions acceptable and concludes that GE has appropriately addressed concerns in NUREG-1449 related to inadvertent loss of RHR.

19.3.3.1.1 ABWR Alternate Decay Heat Removal Features

In NUREG-1449, the staff stated that in case of a loss of RHR, BWR operators can significantly extend the time available for recovering the system by adding water to the core from several sources, including the condensate system and the CRD system, to raise water to a level that can support natural circulation and continue to remove decay heat, thus reducing the probability of damage to the core.

The applicant discussed the alternate DHR capability for the ABWR in SSAR Table 19.Q-1 of the ABWR PRA Shutdown Risk Final Report. In the event that RHR is not available, operators can use several non-safety-related systems as alternate methods to remove decay heat:

- Main steam SRVs can be used to vent steam to the suppression pool thus depressurizing the RPV and allowing the use of other low-pressure systems.
- During shutdown, the reactor water cleanup (CUW) system can be used under certain conditions to remove decay heat. Water is moved through a line attached to the RPV bottom head, through a series of heat exchangers and filter demineralizers and then returned to the RPV through an attachment to the upper head or the feedwater lines.
- The fuel pool cooling (FPC) system can be used for DHR during Mode 5 (refueling).
- RPV water boiling with the vessel head off is an effective way to remove heat but is not a preferred method because of the potential for offsite releases. However, it can be used as long as the RPV level can be maintained by available make-up sources.

The staff asked GE to discuss the impact of direct RPV water boiling to the containment as an acceptable alternate DHR method. In a letter dated January 13, 1993, GE stated that analysis results indicate that offsite doses from direct boiling in the RPV during Mode 5 will be much lower than required by regulatory limits of 10 CFR Part 100. Component operability is ensured because the RHR components will be qualified for a harsh

environment. Reliable components such as the CRD pumps are expected to survive for a significant period of time in a low-pressure steam environment to support alternate make-up capability. Minimum operator actions required to initiate direct boiling of RPV water with the head removed include opening three manual valves inside secondary containment and actuating the ACIWA system. Since the time to reach steam boiling is several hours after DHR cooling is lost, the operator actions to manually open the valves are reasonably assured. The staff concludes that direct boiling in the containment can be used as an alternate DHR method, but should be used only after attempts to restore other DHR methods, such as use of the FPC system, have been unsuccessful.

The ABWR design will enable operators to cool the core using alternate DHR methods as described above. The loss of RHR during shutdown can be responded to as long as non-safety-related equipment used for the alternate methods is made available and clear procedures have been prepared for applying the methods. Maintenance of the decay heat capability and procedures are discussed in Section 19.3.7.2 of this report. The staff concludes that the applicant has sufficiently addressed concerns in NUREG-1449 regarding the capability to provide alternate core cooling in case of a loss of RHR.

19.3.3.2 Inventory Control

In NUREG-1449, the staff stated that loss of inventory is more likely to occur during shutdown or refueling than during normal operating conditions because of system repairs, maintenance, and component replacement. Activities such as test and maintenance during shutdown that require seldom-used valve line-ups and plant configurations increase the probability of operator error associated with inventory control. Loss of inventory can lead to fuel damage by overheating if no make-up water is added to the core.

GE discussed the ABWR inventory control in SSAR Section 19.Q.4.2 of the ABWR PRA Shutdown Risk Final Report. In the event that the RPV level decreases during shutdown conditions, Modes 4 and 5, the ABWR design will automatically initiate, on low reactor water level, the high-pressure core flooders (HPCF), and the low-pressure flooders (LPFL) systems to inject water from the suppression pool. These features are part of the ECCS, and the TS require automatic actuation of the ECCS to be operable in these modes. Other manually initiated systems could also be used to inject water from the main condenser hotwell, CRD system, and ACIWA system. These features will be part of the ABWR alternate DHR capability and the COL applicant will develop administrative control procedures, with appropriate

Severe Accidents

guidance from the reactor vendor, to ensure that alternate DHR capability is available. All isolation valves used in the shutdown cooling mode of RHR and CUW, except those in the injection lines, will automatically close on a low RPV water level signal to isolate the inventory losses. If the HPCF and LPFL systems are in the test mode and an RPV low-level signal is received, the systems will automatically switch to the vessel injection mode. Additionally, the ABWR design uses reactor internal pumps that eliminate recirculation piping external to the RPV, hence the probability of LOCAs during normal and shutdown operations will likely be reduced.

To further reduce the likelihood of a loss of inventory, RHR system valves will be interlocked with the reactor system pressure to ensure that low-pressure RHR piping will not be exposed to full system pressure. The low-pressure portions of RHR piping are designed to withstand the full reactor pressure without rupture in the event that the interlocks fail or are bypassed.

In NUREG-1449, it was noted that draindown of the RPV water to the suppression pool could occur if the motor-operated valves in the RHR system inadvertently opened or by operator errors. The ABWR design includes provisions for preventing inadvertent draining of the RPV to the suppression pool. Interlocks require that the RPV shutdown cooling suction valve be fully closed before the suppression pool return or suction valves can be opened. In the reverse situation, interlocks require that suppression pool return or suction valves must be fully closed before SDC suction valve can be opened. The permanently installed RPV level indication system will give level indications and alarms to operators in the control room during shutdown operations.

19.3.3.2.1 Temporary Reactor Coolant System Boundaries

19.3.3.2.1.1 Use of Freeze Seals in ABWR

The RCS in the ABWR design includes significantly less piping than does the RCS in currently operating BWR designs. For example, the design includes no external recirculation loops, and the RHR piping connected to the RPV will enter at a higher level than the top of the active fuel. Therefore, inadvertent draining from these lines will stop without exposing the fuel. The ABWR will contain no pipes larger than 5.08 cm (2 in.) in diameter below the core, thus allowing operators more time to recover coolant inventory if coolant is lost as a result of maintenance, valve failures, or pipe breaks. However, freeze seals are sometimes used and are discussed in SSAR Section 19.Q.8 of the ABWR PRA Shutdown Risk Final Report. Freeze seals are often used to isolate system piping for the repair

and replacement of components such as valves, pipe connections, pipe fittings, and pipe stops, where it is impossible to perform the task without isolating the piping system in such a way. The use of freeze seals to perform such activity in the RCS boundary is of concern because a failed seal would compromise the structural integrity of the boundary, thus creating a loss of inventory control. In NUREG-1449, the staff discussed events in which a freeze seal used in secondary system equipment failed, resulting in a loss of inventory, causing flooding, and rendering equipment inoperable. These incidents occurred because plant personnel failed to follow the procedures and properly maintain the freezing media.

GE stated that the COL administrative procedures will ensure the integrity of the temporary boundary when freeze seals are used. Mitigative measures will be identified in advance, and appropriate back-up systems will be made available to minimize the effects of a loss of coolant inventory.

The staff considers this a COL action item and will ensure that COL applicants provide guidance on controlling and maintaining the integrity of freeze seals. The guidance should address the use of an engineering safety analysis on a case-by-case basis to ensure that the use of freeze seals, where a failure could result in loss of inventory, will not result in any unresolved safety review questions. GE has included this COL action item in SSAR Amendment 34 and the staff finds it to be acceptable.

19.3.3.2.1.2 Reactor Internal Pump Motor and Impeller Replacement

The ABWR reactor internal pumps (RIPs) are used to supply coolant circulation and to replace the external coolant recirculation system used in the BWR designs. This is a design improvement over the BWR designs in which an unisolable pipe break or component repair in the external recirculation system could result in a major loss of inventory control. The RIP concept was adopted from European BWRs that have been operating for more than 15 years and have had no indications of difficulty in maintenance or in operation that resulted in a loss of inventory. GE discussed the procedures to maintain and to replace the RIPs in SSAR Section 19.Q.4.2 of the ABWR PRA Shutdown Risk Final Report.

Removal of the RIP motors for maintenance is accomplished by using integral inflatable seals that act as backup sealing devices to assure that no RPV water leakage occur. Following each motor removal, a temporary cover plate is bolted to the bottom of the motor's housing, which forms part of the reactor vessel. The impeller is then removed from the top. Upon the

removal of the impeller, the bolted cover plate acts as the RPV boundary and prevents leakage of reactor water. A plug is then installed on the RPV bottom head at the impeller nozzle to provide additional protection against draining the RPV.

The staff asked GE to discuss the effect the increased temperature from a loss of DHR cooling during RIP replacement or maintenance could have on inflatable seals. In a letter dated January 13, 1993, GE stated that the inflatable seals on the RIP shaft will be permanently installed and will be designed to handle normal operating temperatures. Increased temperatures from a loss of DHR cooling will not affect the performance of the seals since the coolant temperature during shutdown conditions will be less than the design temperature of the seal.

The staff reviewed the RIP maintenance and replacement sequences and found that if leakage occurred during the removal of the RIP motor, the temporary cover plate could be installed to eliminate RPV water leakage. The pump internal primary and inflatable secondary seals would minimize the potential RPV water drainage, thus making the RIP motor removal process acceptable. However, the staff noted that during the RIP impeller and shaft removal, possible unisolable LOCA with an opening of about 20.32 cm (8 in.) exists in the event that operators failure to follow the maintenance procedure or possibly as a result of miscommunication. In addition, during pump impeller and shaft replacement, the containment would be opened, thus allowing a direct release path to the environment. The staff, therefore, requires that RIP impeller and shaft replacement be conducted only after fuel has been removed from vessel. This was Open Item F19.3.3.2.1-1.

In a letter dated January 14, 1994, and subsequent responses to staff questions, GE provided additional information with regard to RIP impeller-shaft removal and CRD replacement. During the RIP shaft and impeller removal, the following replacement sequence, maintenance requirements, and pump design features together with the refueling platform auxiliary hoist design are intended to minimize the likelihood of an unisolable LOCA.

- Upon completion of the RIP motor removal, a maintenance cover is bolted to the bottom of the motor housing, which forms a temporary RPV boundary. The motor housing is then pressurized to verify that the maintenance cover is providing a seal. The secondary inflatable seal is then depressurized. At this point, two seals (internal primary metal-to-metal, and maintenance cover) are still provided. Upon removal of the pump impeller-shaft, only the maintenance cover seal remains. To protect against removal of the impeller-

shaft in the event that maintenance seal is not in place, an auxiliary hoist interlock is provided. The refueling platform auxiliary hoist interlocks will interrupt the hoisting power if the load exceeds a specified setpoint. The hoist load setpoint is less than the sum of the impeller-shaft weight and the hydrostatic head on the impeller. To overcome the static head, the motor housing must be pressurized, which requires the maintenance cover plate to be secured in place, and thus sealing is assuring.

- When the pump impeller-shaft has been removed, a maintenance diffuser plug is then installed over the shaft opening. The diffuser plug provides sealing and is the only means to prevent possible unisolable LOCA when the motor housing is drained and the maintenance cover plate is removed for secondary inflatable seal and stretch tube inspection or replacement. To prevent this potential unisolable LOCA, the diffuser plug is designed with a break-away lifting lug. If the maintenance cover is not secured in place and pressurized, the lifting lug will break during the attempted removal due to the static head pressure exceeding the lug's design force, thus ensuring that the diffuser plug seal is maintained. In the event that the operator inadvertently removed the plug, abnormal or excessive drainage will be discovered when the motor housing is partially drained through the drain line. At this point, RIP sealing is still provided by the maintenance cover plate. Discontinued drainage of the motor housing will eliminate the loss of reactor coolant and allow corrective actions.

To further ensure that there is no leakage with the motor bottom cover installed, GE specifically states that COL applicant develops procedure to visually monitor for potential leakage from the motor housing during pump shaft lifting and maintenance plug removal. The staff finds these provisions acceptable and that the potential unisolated LOCA is minimized during RIP maintenance and replacement. Therefore, Open Item F19.3.3.2.1-1 is resolved. However, the staff considers this a COL action item and will review the COL procedures to ensure appropriate installation and verification of motor bottom cover, as well as visual monitoring of the potential leakage during impeller-shaft and maintenance plug removal have been considered. Also, the staff will ensure that COL applicant develops a contingency plan (e.g., close personnel access hatch, safety injection), which assures that core and spent fuel cooling can be provided in the event that a loss-of-coolant occurs during RIP maintenance. GE has included this COL action item in SSAR Amendment 34 and the staff finds it to be acceptable.

Severe Accidents

19.3.3.2.1.3 Control Rod Drive Replacement

CRD replacement for the ABWR is similar to that of current BWRs, and will use the same maintenance procedures. The CRD is withdrawn to the point where its blade fits onto the CRD guide tube. This provides a metal-to-metal seal that minimizes the RPV water drainage when the CRD is removed. The staff reviewed the replacement process and found that unisolated LOCA with an opening of about 5.08 cm (2 in.) exists at the bottom of the vessel head if the CRD blade and drive are simultaneously removed because of operator failure to follow the procedures. In the DFSE, the staff stated that its position was that TS should be included to prohibit the removal of the blade and drive of the same assembly. This was Open Item F19.3.3.2.1-2.

The procedure for removal of the line motion control rod drive (FMCRD) for maintenance or replacement is similar to current BWRs. The control rod is first withdrawn until the CRD blade is backseated onto the control rod guide tube. This provides a metal-to-metal seal that minimizes the RPV drainage when the FMCRD is subsequently lowered and removed. The CRD blade normally remains in this backseated condition at all times with the FMCRD out. In the event that the CRD blade is required to be removed for replacement, a temporary blind flange will be first installed on the end of the CRD housing to prevent draining of the reactor water.

During the FMCRD removal, personnel are required to monitor under the RPV for water leakage out of the CRD housing. If abnormal or excessive leakage occurs after only a partial lowering of the FMCRD, which is the indicative of a metal-to-metal seal that has not yet been established, the FMCRD can then be raised back into its installed position to eliminate the leak and allow corrective action. In the event that the CRD blade and drive of the same assembly were inadvertently removed due to operator failures to follow procedures during refueling operations with water level greater than 23 ft above the vessel flange, the analysis results indicated that it would take approximately 36 minutes for the water to fill the lower drywell sump, reach the tunnel entrances, and begin flowing into the access tunnels. With the expected flow rate of 174 cubic meters per hour (6,145 ft³/hr) from the CRD opening, the water in the spent fuel would drop approximately .3 m/hr (.98 ft/hr). The high drywell sump level and the low spent fuel level would alarm in the main control room approximately 2 minutes and 28 minutes, respectively, into the transient. The normally operating non-safety-related makeup water condensate system (MUWC) will automatically start upon receiving a low level alarm in the spent fuel pool and transfer water to the spent fuel pool cooling and cleanup (FPCCU) system. The

RHR spent fuel cooling mode can be manually initiated to provide makeup injection and the suppression pool cleanup system also can provide backup if the MUWC is not available. In the event of loss-of-offsite power, backup water also can be provided by RHR as independent water addition system.

Upon identified leakages from the bottom of the RPV, it is expected that personnel door and equipment hatch in the lower drywell areas will be closed within 30 minutes before the water level would reach the tunnel entrances and begin flowing into the access tunnel. Appropriate actions will then be taken to reinsert the CRD blade and to mitigate the event using various water sources and injection systems and mentioned.

Additionally, the FMCRD design also allows partial removal of certain mechanical assemblies without the need to withdraw the associated CRD. These mechanical components include the stepping motor, position indication probe (PIP), and spool piece. While these components are removed for maintenance, the associated CRD will be maintained in the fully inserted position by one of two mechanical anti-rotation locking devices, which are part of the FMCRD design. Details of the anti-rotation locking devices and verification process are discussed in SSAR Section 4.6.2.3.4

The staff reviewed the FMCRD and its associated anti-rotation locking devices design and concluded that adequate locking mechanisms are provided to assure that control rods remain fully inserted with the FMCRD subassemblies removed. In addition to the locking devices, TS prohibits removal of any two adjacent CRD subassemblies to prevent a potential inadvertent criticality event during refueling operation.

The staff also notes that only two or three complete FMCRDs are required to be removed for inspection each refueling outage. This is an improvement relative to the CRD system design at current BWRs which have piston seal replacement needs such that 20 to 30 drives are typically removed each refueling outage.

The staff finds that the FMCRD design improvements, provisions to control potential loss of reactor water, and ample time available for operators to initiate corrective actions in the event of coincident removal of the CRD blade and drive of the same assembly during refueling outage acceptable, and therefore, Open Item F19.3.3.2.1-2 is resolved. However, the staff considers this issue a COL action item and will ensure that maintenance procedures have provisions prohibit coincident removal of the CRD blade and drive of the same assembly. The staff also will ensure that COL applicant develops contingency

procedures to provide core and spent fuel cooling capability and mitigative actions during CRD replacement with fuel in the vessel. GE has included this COL action item in SSAR Amendment 34 and the staff finds it to be acceptable.

19.3.3.2 Alternate Reactor Inventory Control Feature

The ABWR design includes a non-safety-related feedwater and condensate system, consisting of three electric pumps and associated piping, that can be used as an alternate means for make-up during shutdown operation. The CRD pump can also be used to provide inventory control during shutdown by injecting water from the condensate storage tank to the RPV through the FMCRD system. An ACIWA system is also available to supply make-up water to the RPV if no ECCS make-up water is available.

The staff finds these provisions acceptable and concludes that GE has sufficiently addressed the concerns in NUREG-1449 related to alternate make-up capability to provide core decay heat removal. The alternate inventory control features using the feedwater, the condensate system, and the CRD pump will provide alternate core cooling upon loss of normal RHR capability. The staff also finds that an ACIWA system will further enhance the capability of the ABWR to maintain core cooling in the event that no ECCS make-up is available.

19.3.3.3 Containment Integrity

During refueling of the ABWR, the primary containment head is removed and cannot be readily repositioned to restore containment integrity. This is also the case for operating BWR plants with Mark I and II containments. NUREG-1449 stated that BWR secondary containments were judged unlikely to prevent an early release following initiation of boiling with an open RCS or during potential severe-core-damage scenarios. This is also the case for the ABWR.

In NUREG-1449, the staff evaluated the need to reestablish containment integrity for all operating plants under shutdown conditions. Based on operating experience, thermal-hydraulic analyses, and PRA assessments, it was concluded that containment integrity under some shutdown conditions may be necessary for pressurized water reactor (PWR) plants. However, this conclusion was not reached for BWR plants. This is a result in part, of the decreased frequency and significance of precursor events involving reduction in reactor vessel level or loss of RHR (or both) in BWRs as compared to PWRs. In addition, BWRs do not enter a midloop operating condition as do PWRs.

In NUREG-1449, the staff stated that operating BWR alternate DHR methods provide significant depth and diversity. For these reasons, the staff concluded that loss of RHR in BWRs during shutdown is not a significant safety issue as long as the equipment (pumps, valves, and instrumentation) needed for these methods is operable and clear procedures exist for applying the methods. As discussed in Sections 19.3.3.1, 19.3.3.1.1, and 19.3.3.2 of this report, GE provided design features to minimize the risk from shutdown events and ensure the availability of DHR and reactor inventory. GE stated that ABWR TS required that secondary containment automatically be isolated on high radiation from a radiological boundary breach or fuel handling accident. Also, procedures should be developed by the COL applicant to ensure that (1) the primary containment is available during Modes 3 and 4 (if appropriate), and (2) the secondary containment can be maintained functional as required, especially during higher-risk evolutions.

Based on the conclusions reached in NUREG-1449 and the improvements, beyond that of operating BWRs, provided in the ABWR design, the staff concludes that additional requirements are not necessary for the primary containment. The requirements to isolate secondary containment on high radiation and fuel-handling accidents, and procedures to ensure the availability of secondary containment during high-risk evolutions could contribute to the mitigation of a low-power and shutdown event.

19.3.3.4 Electrical Power

In NUREG-1449, the staff concluded that the availability of electrical power is vital to maintaining shutdown cooling. A loss of power could range from the complete loss of ac power to the loss of a dc bus or an instrument bus. Loss of electrical power generally leads to other events, such as a loss of SDC.

The staff reviewed the design for the electrical system described in SSAR Section 19.Q.4.4 of the ABWR PRA Shutdown Risk Evaluation Final Report. The ABWR electrical system design includes three diesel generators — one diesel generator for each safety division. A non-safety-related combustion gas turbine is an alternate means of supplying power. The combustion gas turbine can start a feedwater or other pump for DHR or inventory make-up, if required upon a loss of offsite power and diesel generator failure.

Two independent offsite power sources, three unit auxiliary transformers powering three Class 1E and non-1E buses, and four safety divisions of dc power would increase the availability of power for equipment.

Severe Accidents

The staff finds that the ABWR electrical power system design contains redundant electrical power sources. The availability of electrical power during shutdown is discussed in Section 19.3.7.3 of this report.

19.3.3.5 Reactivity Control

In NUREG-1449, the staff indicated that inadvertent criticality events at BWR plants have resulted in reactor trips. Inadvertent reactivity is most often caused by human error (the operator selecting the wrong control rod) and feedwater transients. GE addressed reactivity excursion events in SSAR Section 19.Q.4.5 of the ABWR PRA Shutdown Risk Final Report, and in Chapter 15 of the SSAR. GE stated that reactivity events during shutdown could result in critical events by moving control rods or making errors in handling fuel, which could jeopardize DHR or fuel integrity. These events could result from any of the following:

- Control rod drop
- Control rod ejection
- Refueling error
- Rod withdrawal error
- Fuel loading error.

19.3.3.5.1 Control Rod Drop

A control rod drop event could occur during control rod testing at shutdown. The applicant discussed the control rod drop accidents in details in Chapter 15 of the GE ABWR SSAR. To limit the reactivity increases that would result from a free-falling control rod, the ABWR will include a latch mechanism to restrict the distance of rod free-fall to an acceptable limit. The FMCRD design will detect the separation of the CRD mechanism. A rod block signal will prevent a second control rod from being withdrawn, if any one control rod is fully withdrawn. The latch mechanism would limit to 20.32 cm (8 in.) the distance a rod could drop if (1) the rod block signal failed, (2) the operator incorrectly selected an adjacent control rod for withdrawal, and (3) the incorrectly selected rod became stuck and decoupled from its drive. Two redundant and separate Class 1E switches will detect the separation of either the control rod from the hollow piston or the hollow piston from the ball nut. If either the FMCRD or the Class 1E separation detection system were actuated, an alarm would annunciate in the control room, thus reducing the probability of rod drop accidents going undetected. These features provide acceptable plant response as discussed in Section 15 of the FSER.

19.3.3.5.2 Control Rod Ejection

During shutdown operation, a major break on the CRD housing or associated CRD pipelines from RPV hydrostatic testing of a control rod could cause a control rod to be ejected. The FMCRD system will include redundant brake mechanisms to prevent this accident from causing severe consequences. The rod drop detection system also will provide the same protection against a rod ejection accident and will include alarms in the control room to alert operators to this accident. Details of the control rod ejection accidents and plant response are discussed in Section 15 of the FSER.

19.3.3.5.3 Rod Withdrawal Error

The staff stated that certain inadvertent criticality events resulted in reactor trips. Inadvertent criticality is most often caused by human error (such as the operator's selecting the wrong control rod). GE stated that the reactor could become critical if two adjacent control rods were withdrawn at the same time during refueling operations. To prevent a rod from being inadvertently withdrawn, the ABWR design will include interlocks to ensure that all control rods are inserted while fuel is being handled over the core. The design also will include an interlock to prevent more than one control rod from being withdrawn at a time. If the interlock fails and the control rod is withdrawn, the reactor would trip on a high-flux signal received in the control room. Details of the rod withdrawal errors are discussed in Chapter 15 of the ABWR SSAR and are evaluated in Section 15 of the FSER.

The ABWR refueling interlocks that prevent more than one control rod from being withdrawn at a time and the high-flux signal scram features will provide diverse protection against rod withdrawal errors during shutdown operations. The staff judged that there is only a small probability of the coincident failures of the refueling interlock and RPS occurring together with operator errors.

19.3.3.5.4 Refueling Error

A reactivity excursion event in an ABWR could occur during refueling operations if a fuel bundle is inserted at the maximum fuel grapple speed into a fueled region of the core. Details of the refueling error events are discussed in Chapter 15 of the ABWR SSAR. The ABWR will include the following design features to prevent or mitigate refueling errors:

- (1) An interlock with a mode switch in the REFUEL position prevents another fuel assembly from being hoisted over the vessel if a control blade has been removed.
- (2) While the mode switch is in the REFUEL position, only one rod can be withdrawn at a time. Any attempt to withdraw a second control rod would result in a rod block signal from the refueling interlocks.
- (3) A refueling error will cause the start-up neutron-monitoring system to send an alarm and alert operators to the condition.

Details of the refueling errors are discussed in Section 15 of this report.

19.3.3.5.5 Fuel Loading Error

A fuel loading error is similar to a refueling error. GE stated that if the refueling procedure were not followed, the core reactivity would increase to a value higher than the design value, and if the operator, in performing the core verification process, failed to identify a misplaced fuel bundle, subsequent control rod testing could result in an inadvertent critical event and a power excursion. The likelihood and acceptability of plant response to this event is discussed in Section 15 of this report.

19.3.4 ABWR Reactor Instrumentation

In NUREG-1449, the staff stressed the importance of dedicated shutdown annunciators and instruments used during shutdown operation to provide RCS coolant temperature indication, reactor water level indication, and RHR system status.

The applicant discussed ABWR instruments in SSAR Section 19.Q.5 of the GE ABWR PRA Shutdown Risk Final Report. The applicant stated that, to minimize risk during shutdown, the instrument system must monitor the RPV level, the water temperature, the make-up sources, and heat sinks, and must display these parameters in the control room to the operators in a reliable manner that is easily understood. These instruments can also supply signals to actuate ECCS functions upon receiving a signal for low reactor water level, to automatically insert control rods on high flux, and to close appropriate isolation valves.

The staff asked GE to discuss the reliability of power available for the CUW system to measure coolant temperatures and the adequacy of the DHR system parameters monitored by the ABWR instrument system.

The applicant stated in letters dated January 13 and 28, 1993, that resistance temperature detectors (RTDs) in the CUW system suction lines and the RHR pump discharge will be used to measure reactor water temperatures. The temperature indications and alarms will be located in the control room to enable operators to monitor coolant temperature during shutdown operations. Upon a loss of offsite power, the RHR systems will be powered from the EDGs and the CUW system can be powered by either the EDG or an alternate onsite ac power source combustion turbine generator (CTG). When an alarm is received from the CUW system, the operator would retrieve the associated system on a computer display to determine the specific cause for the alarm condition. GE also gave specific DHR system parameters that will be monitored in shutdown conditions.

Instrument features important to shutdown operations include the following:

- Automatic initiation of ECCS to ensure adequate RPV make-up during Modes 4 and 5
- Four channels of instruments to allow for bypass during maintenance and testing while retaining the redundancy of the system (the two-out-of-four logic reverts to two-out-of-three during maintenance bypass)
- Continuous monitoring to detect fires or flooding in safety-related and other areas
- Operability of the RPS during shutdown conditions to mitigate any reactivity excursions
- Interlocked refueling bridge operation to prevent reactivity excursions
- Automatic isolation of SDC valves (F-010 and F-011) on low level in the RPV to prevent fuel from being uncovered
- Interlocked RHR valves (SDC and suppression pool) to reduce the possibility of diverting coolant from the RPV to the suppression pool
- Ability to shut down the plant from the remote control panel if the control room becomes uninhabitable
- Ability to monitor radiation levels throughout the plant to detect breaches in radiological barriers.

Parameters that are monitored by the instrument system include the following:

- RPV level, water temperature, and pressure

Severe Accidents

- Neutron flux
- Drywell and wetwell pressure and temperature
- Suppression pool temperature and level
- Turbine building condenser pit and reactor component cooling water (RCW) rooms in the control building flooding level
- RHR flow rate, pump motor trip, loop logic power failure
- Fire detection in various buildings
- Electric power distribution system parameters (e.g., power, voltage, current, and frequency)
- Operation of the fire water system
- CUW outlet temperature high.

The staff finds these provisions acceptable and concludes that GE has appropriately addressed concerns in NUREG-1449 related to the importance of dedicated shutdown annunciators and instruments used during shutdown operations to provide RCS coolant temperature indication, reactor water level indication, and RHR system status.

19.3.5 Flooding and Fire Protection

In NUREG-1449, the staff stated that the safety significance of flooding or spills during shutdown depends on the equipment affected by the spills and that such spills are most often caused by human error. Plant activities during shutdown and refueling operations may increase fire hazards in safety-related systems essential to the plant's capability to maintain core cooling. Further, Appendix R to 10 CFR Part 50, Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979, and current NRC fire protection philosophy do not address shutdown and refueling conditions and the effect that a fire may have on the plant's ability to remove decay heat and to maintain shutdown cooling. SSAR Section 19.Q.6 of the ABWR PRA Shutdown Risk Evaluation Final Report addresses the capability of the plant design to protect safe-shutdown equipment from fire and floods.

The physical layout of the ABWR design is the primary means of ensuring that this requirement will be met should these hazardous conditions occur. The ABWR design incorporates such methods as proper plant layout, proper system layout and operation, physical and electrical separation, and the use of administrative controls to be

established by the referencing applicant to provide DHR function and to maintain safe shutdown operation.

The layout of the plant will minimize the propagation of fires and floods by locating the control building (CB) between the reactor building (RB) and the turbine building (TB). Most safe-shutdown equipment will be located in the RB and most non-safety-related equipment will be located in the TB; therefore, failures of systems in the TB because of fires or floods will not adversely affect safety-related equipment.

GE stated that the ABWR includes three independent safety-related RHR divisions. These divisions will be separated by 3-hour fire barriers, and divisional equipment will be electrically isolated from redundant divisions of equipment except for equipment in the main control room (MCR), remote shutdown panels (RSPs), primary containment, and several special cases in which divisional equipment is in close proximity to equipment in another division as described in SSAR Section 9A.5, Special Cases. Also, there are some cases where cables of more than one division are in relatively close proximity and require special justification. In each of these areas, the applicant has provided recommendations concerning fire protection during shutdown conditions to ensure that DHR capability remains intact.

The ABWR design and TS in Modes 3, 4, and 5 ensure physical and electrical separation that no more than one of the three redundant divisions of safe-shutdown equipment can be rendered inoperable because of a fire or flood. Physical and electrical separation of redundant safety-related divisions ensures that, if a division is rendered inoperable because of fire or flood, at least one division will be operable and available to provide DHR function. The ABWR will not include fire walls inside primary containment to provide divisional separation because of the need to rapidly equalize pressure among these divisions after a high-energy line break. In the ABWR SSAR, GE stated that the divisions of safe-shutdown equipment will be widely separated (at intervals of 120 degrees around the containment) so that a single fire will not be able to render inoperable any combination of active components that are used to ensure maintenance of a safe-shutdown condition. During normal plant operation, the primary containment will be made inert, therefore, the potential for fire in the containment is reduced substantially. During shutdown or refueling activities, the primary containment will not be made inert and, therefore, would be susceptible to fires because of maintenance activities that involve increased amounts of combustible loading and a number of ignition sources (cutting and welding equipment). The COL applicant referencing the ABWR design will prepare

administrative procedures to limit storage of combustibles and control ignition sources inside primary containment.

Although the divisions of safe-shutdown equipment located in the primary containment will not be located in the separated fire areas, other methods will be available to deal with fires in the primary containment such as minimizing the safe-shutdown components, maximizing distance between divisional equipment, and providing sprinkler coverage from the containment spray mode of RHR.

Within the primary containment, check valves and automatic depressurization system (ADS) SRVs are the only components that need to operate to achieve safe shutdown. A fire cannot disable the check valves or the spring-actuated SRVs, and the high-pressure injection pumps will provide enough pressure to lift the SRVs. Therefore, the safe shutdown capability will not be compromised as a result of a fire. GE stated that containment spray can be used to mitigate the consequences of a fire inside primary containment. The staff does not consider this to be a primary fire mitigation feature. It believes that the design of the SRVs and check valves, along with the wide spaces between divisional equipment and administrative procedures, provides sufficient assurance that safe-shutdown equipment will remain functional, should a fire or flood occur inside the primary containment during shutdown conditions.

Although the MCR includes controls and indications for safe-shutdown divisions, redundancy is achieved by providing the RSP in a separate area of the RB. The MCR and the RSP will be separated by 3-hour fire barriers and the RSP will be hard-wired to field devices and power supplies. Fiber optic cables from the MCR will transmit two identical digital control signals to operate equipment. In the event of a major fire in the MCR, the instrumentation and control power for safety-related cooling systems will be transferred from the MCR to the two RSPs, each of which is capable of serving one division of safety-related cooling systems. It is not likely that a fire in the MCR would cause spurious operation of equipment (barring inadvertent control switch operation) because two simultaneous identical signals are needed at the demultiplexer for control action to be taken at the field device (element).

The RSP rooms will be separated from each other by a fire barrier and fire door assembly. The transfer switches will be located in each room. GE stated that if a fire involved one of the safe-shutdown rooms and caused inadvertent operation of safety-related equipment, only one of the two remaining divisions would be needed to bring the reactor to a hot shutdown and then cold shutdown condition.

The heating, ventilating, and air-conditioning (HVAC) systems will be used during shutdown conditions to prevent smoke, hot gases, heat, or fire suppressant in one fire area from migrating into other fire areas to the extent that they could adversely affect the safe shutdown capability of redundant equipment. When the HVAC systems are operated in the smoke-removal mode, the system in the area experiencing fire will be maintained at a lower pressure than the other fire areas. This will ensure that any leakage across a barrier will be drawn into the area experiencing the fire and will provide a clean air space for personnel access to the fire. The fire suppression and detection system used in conjunction with administrative controls will enable the COL applicant to effectively fight the fire.

While the plant is in shutdown modes, the licensee will commonly bring additional personnel and materials to the site for various refueling and maintenance activities. During this time, some safe-shutdown systems may be inoperable for maintenance at a time when fires are more likely to occur. Administrative controls proposed by GE will ensure that at least one safety-related division is operable (the operable division) and that another safety-related division is operable with fire and flood barriers fully in place (the standby division). Thus while one division may be in maintenance, another division will be operable (although its fire or flood barriers may be breached), and the third division will be in standby. If the operable division is rendered inoperable by a fire or flood, the standby division will be available to perform the DHR function. Likewise, if a fire or flood renders the standby division inoperable, the operable division will continue to perform the DHR function. The COL applicant refer to the ABWR design will establish administrative procedures to ensure that an unanticipated breach of a fire barrier will be discovered and compensated for in a timely manner. The staff considers fire protection administrative procedures a plant-specific issue. In addition, the staff will ensure that the COL applicant has appropriately addressed the availability of the safe-shutdown equipment using the physical and electrical separation concept to provide DHR function and to maintain safe-shutdown operation. The staff also will ensure that the COL administrative procedures provide appropriate controls of combustibles and ignition sources during shutdown operations. GE has included this COL action item in SSAR Amendment 34 and the staff finds it to be acceptable.

The physical and electrical separation of redundant safe-shutdown equipment provides for flood protection. The ABWR will include barriers to protect against both flood and fires. Doors that protect against flood are watertight with seals that will seat with increased water pressure from outside the room. These watertight doors also will be

Severe Accidents

alarmed so that if the door is opened (i.e., the flood barrier is breached), security will be alerted and will take appropriate actions. The ABWR will also reduce flood hazards by minimizing sources of flood water near safety-related equipment and by alarming and isolating water sources on indication of high water levels. Safety-related equipment will be installed at a minimum level of 200 mm (8 in.) off the floor with adequate drainage in the floor spaces. GE will give COL applicants guidance on administrative controls to effectively deal with a flood situation.

The staff reviewed the protection philosophy for the ABWR, the design features used to protect safe-shutdown equipment from fires and floods and their effects, and concludes that the design provides for adequate fire and flood protection for systems and components required to achieve and maintain safe-shutdown and is acceptable.

19.3.6 Shutdown Risk Insights

In supporting the design certification of the ABWR design, the NRC required GE to prepare a suitable, systematic risk analysis for those operating modes other than full power. On July 2, 1992, GE submitted its shutdown risk evaluation for the NRC staff to review. The staff reviewed GE's shutdown risk PRA for the ABWR design. This PRA included discussion of dominant accident sequences; calculation of core damage frequency from internally initiated events in Modes 3, 4 and 5; and vulnerabilities while operating the plant in modes other than full power. The staff also considered human reliability insights, important human actions, insights from uncertainty, importance, and sensitivity analyses.

In SSAR Table 19.Q-6 of the shutdown risk evaluation final report, GE performed a failure modes and effects analysis (FMEA) of the ABWR new design features to determine if any vulnerabilities were introduced by the new technology. GE concluded that the new design features would not introduce new vulnerability. The staff discussed these assessments in Section 19.1.3.4 of this report.

The staff reviewed the shutdown risk PRA for the ABWR design. It concluded that the results of the shutdown risk evaluation provide sufficient information to determine that there would not be a disproportionate risk of operating the plant in modes other than full power. The staff also found no unreported shutdown risk vulnerabilities.

19.3.7 Technical Findings in NUREG-1449

In NUREG-1449, the staff identified several operational-related issues for regulatory action. These issues are the following:

- Outage planning and control
- operator training and procedures
- TS

19.3.7.1 Outage Planning and Control

In the absence of strict TS controls, licensees have considerable freedom in planning their outage activities. NUREG-1449 indicated that outage planning dictates what equipment will be available and what and when maintenance activities will be undertaken. It effectively establishes if and when a licensee will encounter circumstances that are likely to challenge safety functions and the level of mitigation equipment available to deal with such a challenge. The staff believes that a safety-oriented approach to planning and controlling an outage is needed and that such an approach will reduce risk during shutdown, thereby reducing the incidence of precursor events and improving the defense-in-depth concept.

In the GE ABWR PRA Shutdown Risk Final Report, SSAR Sections 19.Q.7 and 10, GE discussed the generic procedure guidelines for planning outages. In response to a staff RAI regarding the guidelines for conducting and planning outages, GE stated in a letter dated January 13, 1993, that the plant-specific operating procedures, and the GE ABWR procedure guidelines for outages and planning will endorse NUMARC 91-06, Guidelines for Industry Actions to Assess Shutdown Risk Management. The staff believes that in addition to NUMARC 91-06 guidelines, fire protection and appropriate use of instrumentation also should be considered. Fire protection provisions for plants during shutdown operation and the use of plant instruments important to shutdown to monitor and detect a loss of DHR cooling during reduced inventory are discussed in Sections 19.3.5 and 4 of this report, respectively. The procedure guidelines for outage and control planning will be based on an engineering evaluation including engineering safety analyses and will address the effective planning and control of outages and the maintenance of important shutdown functions: DHR capability, electrical power availability, reactivity control, and containment (primary/secondary) integrity.

The plant-specific guidelines for planning and controlling outages will include the following:

- An outage philosophy including a list of organizations responsible for scheduling outages. These guidelines should address both the initial outage plan and all safety-significant changes to schedule.
- Provisions to ensure that all activities receive adequate resources. The plan should also consider unanticipated changes and increases of the scope.

- Provisions to ensure defense in depth during shutdown and to ensure that safety margins are not reduced. An alternate or backup system must be made available if a safety system is removed from service.
- Provisions to ensure that all personnel involved in outage activities are adequately trained. This should include operator simulator training to the extent practicable. Other plant personnel, including temporary personnel, should receive training commensurate with the outage tasks they will be performing.
- Provisions for an independent safety review team that would be assigned to perform final review and grant approval.

The staff finds that improvement in safe operation of the ABWR plant in low-power and shutdown modes can be reasonably accomplished by implementing GE's guidelines for preparing and implementing an outage plan. It concludes that GE has adequately addressed important areas described in NUREG-1449 regarding outage planning and control. The staff also notes that specific shutdown TS requirements and guidelines for preparing and implementing an outage plan, will significantly improve safe shutdown operation. ABWR shutdown TS requirements are discussed in detail in Section 19.3.7.3 of this report.

The staff considers outage planning and control a plant-specific issue and will verify that each COL applicant has appropriately implemented GE's guidance and recommendations to improve low-power and shutdown operation. The staff will review the COL applicant's outage planning and control program to ensure that the safety principle is clearly defined and documented. The controlled procedure should clearly define an outage planning process and should incorporate preplanning for all outages. GE has included this COL action item in SSAR Amendment 34 and the staff finds it to be acceptable.

19.3.7.2 Operator Training and Procedures

In NUREG-1449, the staff stated that conditions and plant configurations during shutdown outages for refueling can place control room operators in an unfamiliar situation. Personnel that is properly trained and understand the problems that could arise during outages is essential in reducing risk associated with the outage activities.

GE stated in SSAR Section 19.Q.10 of the GE ABWR PRA Shutdown Risk Final Report that each utility must prepare plant-specific operating procedures based on

individual site characteristics and training program requirements. GE gave broad guidance to ensure important safety functions are maintained during shutdown operations as follows:

- Decay Heat Removal Capability. The RHR system will be the normal method of removing decay heat. The COL applicant will use a recovery strategy to address loss of normal RHR including determining an alternate DHR system and personnel responsible for executing the recovery. In planning for an outage, the COL applicant will emphasize the need for RHR systems during periods of high heat decay loads and the later maintenance of the RHR system when decay heat loads have been reduced or when the core has been unloaded to the spent-fuel pool. Procedures will be provided to maintain spent-fuel cooling during core unloading.
- Inventory Control. The COL applicant will implement appropriate procedures to ensure that adequate coolant inventory is maintained at all times during shutdown. Plant activities or configurations in which a single failure can result in loss of inventory will be identified, and compensatory measures will be provided. Specific activities that could result in a loss of inventory such as use of freeze seals, removal of control rods, CRDs, reactor internal pumps, and RHR valve actuation or other activities that could lead to diversion of RPV coolant to the suppression pool will be reviewed, compensatory measures will be provided.
- Electrical Power Availability. The COL applicant will implement procedures to ensure the defense in depth of electrical power sources. Maintenance of power sources will reflect the current plant configurations and conditions. GE recommended normal and alternate power sources be made available during high-risk conditions. The COL applicant will review all maintenance and switchyard activities to identify single failures or procedural errors that could result in loss of power to vital buses during shutdown. Procedures will be written to govern the use of alternate sources of power.
- Reactivity Control. Shutdown reactivity control for the ABWR will be maintained by core design analysis and interlocks to restrict the movement of fuel and CRDs. The COL applicant will provide procedures to ensure that the core is loaded according to design requirements and that fuel movements are not to be permitted while a CRD mechanism is in maintenance. If a refueling sequence must be altered, a new shutdown margin analysis must be performed. All fuel movements will be verified by knowledgeable trained personnel.

Severe Accidents

- **Containment Integrity.** The COL applicant will develop procedures to ensure the availability of primary containment during Modes 3 and 4 (if appropriate). In addition, procedures will be available to ensure that the secondary containment function can be maintained, as required, in all modes.

GE has recommended ways for COL applicants to maintain key safety functions during shutdown. However, the effectiveness of these recommendations would depend on the procedures, characteristics and training program requirements of each plant. The staff considers plant-specific procedures a COL action item and will require COL applicants to appropriately address and incorporate plant-specific safety-related issues and the vendor's operating guidance on safe operations during shutdown. GE has included this COL action item in SSAR Amendment 34 and the staff finds it to be acceptable.

19.3.7.3 Technical Specifications

The TS for current operating plants are the primary sources of operational requirements. The standard technical specifications (STS) include general requirements for reactivity control, inventory control, RHR, and containment capability during all plant conditions. The STS also include general requirements for fire protection.

The staff asked GE to provide shutdown TS requirements for the ABWR design. In letters of June 30 and July 9, 1993, GE submitted shutdown TS for the staff to review.

In the ABWR shutdown TS, GE established systematic requirements for operating the plant in modes other than full-power conditions. The staff reviewed the ABWR shutdown TS using insights from technical findings discussed in NUREG-1449 and the staff's proposed model TS improvements to enhance the safe shutdown operation of all nuclear plants. These proposed improvements were made available for public comment and industry review in July 1993. Table 19.3-1 of this report indicates additional limiting conditions, beyond those currently listed in the improved STS (NUREG-1434), for operation during reduced inventory that GE proposed. The staff compared TS requirement improvements for ABWR design with the BWR-6 TS. The ABWR shutdown TS reflects redundant onsite AC power sources (diesel generators), one offsite power source, and associated support systems to ensure the DHR capability can be maintained and to minimize the loss of DHR from a loss of electrical power. The ABWR shutdown TS closely follows the staff's guidance on the proposed model TS improvements. Therefore, the staff

concludes that the ABWR shutdown TS will include requirements needed for managing risk during shutdown operations.

19.3.8 Summary

The ABWR design provides flexible combinations of DHR systems, alternate features, and associated safety parameters monitored by the ABWR instrumentation and control system during shutdown operations. These combinations of safety-related systems and normally operating non-safety-related systems ensure that the ABWR is adequately protected against accidents during low-power and shutdown operations.

GE considered fire and flood hazards and plant damage that could occur during shutdown operation as evidenced by proper plant layout, proper system layout and operation, physical separation, and electrical separation. GE took appropriate actions for adequate fire and flood protection to ensure that at least one safety-related division is operable and the standby safety-related division also is operable with fire and flood barriers fully in place.

Outage planning and control will include specific operating procedures to address key safety features such as DHR capability, inventory control, electrical power availability, reactivity control, and containment control. Specific guidelines for planning and controlling outages will include organizations responsible for scheduling outages. Personnel involved in outage activities will be adequately trained and proper safety reviews will be conducted. Implementations of these operating procedures are COL action items.

The ABWR design includes specific TS requirements for operating the plant in modes other than full power. These TS requirements will provide appropriate redundancy in equipment during higher-risk evolutions during shutdown. These TS are consistent with the staff's proposed model TS developed from evaluation of shutdown and low-power operations.

19.3.9 Conclusion

Based on the above, the staff finds the ABWR PRA Shutdown Risk Evaluation Final Report acceptable, and meets the staff's proposed applicable regulation for shutdown risk. Further, the staff concludes that GE has adequately addressed the shutdown risk concerns in NUREG-1449 and has demonstrated that the ABWR design will not introduce significant risk during shutdown operations.

Table 19.3-1 Comparison of ABWR shutdown TS and BWR-6 TS

ABWR Shutdown TS	Modes	BWR-6 TS	Modes
*RCW/RSW and UHS — Shutdown Two RCW/RSW divisions and UHS operable	4 and 5 except when water level ≥ 23 ft	No requirements	
*RCW/RSW and UHS — Refueling One RCW/RSW division and UHS subsystems operable	5 with water level ≥ 23 ft	SSW and UHS — Refueling No requirements	
*AC Sources — Shutdown (low level) One offsite power source Two diesel generator (DG) (onsite) power sources	4 and 5 with water level ≤ 23 ft	AC source — shutdown one offsite power source One DG (onsite) power source	4 and 5
*AC Sources — Shutdown (high level) One offsite power source One DG (onsite) power source	4 and 5 with water level ≥ 23 ft		

* A new LCO has been added as a result of the shutdown risk program.

20 GENERIC ISSUES

This chapter covers the staff's evaluation of several topics: (1) unresolved safety issues (USIs) and generic safety issues (GSIs), (2) construction permit/manufacturing license (CP/ML) rule compliance, and (3) incorporation of operating experience in the advanced boiling water reactor (ABWR) design.

Since there is a large number of USIs, GSIs, and CP/ML rule items, the staff has grouped its evaluations according to issue type. Section 20.1 contains task action plan items, which include both USIs and GSIs. Sections 20.2 and 20.3 address new generic issues and human factors issues, respectively, all of which belong to the GSI category. Section 20.4 includes items listed by Three Mile Island (TMI) Action Plan (NUREG-0660) item number, but many are actually evaluated in Section 20.5 since they are also CP/ML items. TMI Action Plan items are in the GSI category. Section 20.5 deals with the CP/ML items, which are listed by their 10 CFR 50.34(f) paragraph number. CP/ML items can also be considered as GSIs. All issues are listed consecutively within each section.

Because of the considerable amount of overlap among CP/ML rule items and USIs and GSIs, Table 20-1 lists all the USIs and GSIs that are technically relevant to the ABWR design or that the staff needed to otherwise address. Table 20-2 lists all the CP/ML issues. These tables provide the issue designation, title, and a reference to the appropriate section(s) of this report containing the evaluation.

Section 20.6 covers incorporation of operating experience in the ABWR design.

Compliance with paragraph (1)(iv) of 10 CFR 52.47(a)

Paragraph (1)(iv) of 10 CFR 52.47(a) requires an application for design certification to include proposed technical resolutions of those USIs and medium- and high-priority GSIs identified in the version of NUREG-0933, "A Prioritization of Generic Safety Issues," current on the date 6 months prior to application and which are technically relevant to the design.

In the draft final safety evaluation report (DFSER), the staff required GE to modify the standard safety analysis report (SSAR) to explicitly discuss the resolution of each technically relevant USI and GSI per 10 CFR 52.47(a)(1)(iv) in the appropriate SSAR section for clarity and to enable the staff to evaluate each item. These were identified in the DFSER as Open Items 20.1-1 and 20.2-1, respectively.

As a result of the ABWR licensing review bases document (letter from T. Murley, NRC to R. Artigas, GE, dated August 7, 1987), GE agreed to address issues beyond the date 6 months prior to the ABWR application. During a conference call on January 13, 1993, the staff and GE mutually agreed that issues identified in NUREG-0933, with Supplements 1 through 15, would be addressed in the ABWR design certification review. It was also agreed that the list of issues contained in NUREG-0933, Appendix B, "Applicability of NUREG-0933 Issues to Operating Reactors and Future Plants," would be used as the baseline list of issues to be addressed for the ABWR design, excluding any issues that were shown in the list to be not applicable to BWR vendors or to future plants. In addition, GE agreed to address five other issues (A-17, A-29, B-5, 29, and 82) that were resolved without the issuance of new requirements, but for which the Office of Nuclear Reactor Research had recommended the development of specific guidance for future plants (although action to develop such guidance is suspended at this time). The staff also asked GE to address one other issue (C-8), the subject of which was an important ABWR review topic.

During the time frame of the review, several issues (113, 120, 121, and 151) were resolved by the NRC without the issuance of new requirements. Since the staff was already pursuing an ABWR response for those items, they were evaluated.

The advance safety evaluation report (SER) stated that based on the staff's evaluation of the issues listed in Table 20-1 of the SER and contingent on GE's incorporation of agreed-to issue mark-ups in the SSAR, the staff concluded that GE adequately demonstrated compliance with or proposed a method of compliance for the USIs and medium- and high-priority GSIs that are technically relevant to the ABWR design as required by 10 CFR 52.47(a)(1)(iv), with some exceptions that required further GE or staff action. These exceptions were that (1) Issue II.F.2 was still open pending the resolution of the differences of views between the staff and GE on the need for diverse instrumentation for reactor pressure vessel water level indication, and (2) Issues II.B.1 and II.K.3(15) required incorporation of COL action items in the ABWR SSAR. These exceptions have been addressed as discussed in Sections 20.5.30, 20.5.18, and 20.4.64, respectively, of this report, and GE has incorporated agreed-to issue mark-ups in the ABWR SSAR. Based on this information and the staff's review of the issues listed in Table 20-1 of this report, the staff concludes that GE has adequately demonstrated compliance with 10 CFR 52.47(a)(1)(iv) for the ABWR design. Therefore, DFSER Open Items 20.1-1 and 20.2-1 are resolved.

Generic Issues

Table 20-1 ABWR - Relevant USIs and GSIs

Issue Designation	Title	FSER Section(s)
A-1	Water Hammer (former USI)	20.1.1
A-7	Mark I Long-Term Program (former USI)	20.1.2
A-8	Mark II Containment Pool Dynamic Loads - Long-Term Program	20.1.3
A-9	ATWS (former USI)	20.1.4
A-10	BWR Feedwater Nozzle Cracking (former USI)	20.1.5
A-13	Snubber Operability Assurance	20.1.6
A-17	Systems Interaction (former USI)	20.1.7
A-24	Qualification of Class 1E Safety-Related Equipment (former USI)	20.1.8
A-25	Non-Safety Loads on Class 1E Power Sources	20.1.9
A-29	Nuclear Power Plant Design for the Reduction of Vulnerability to Industrial Sabotage	20.1.10
A-31	RHR Shutdown Requirements (former USI)	20.1.11
A-35	Adequacy of Offsite Power Systems	20.1.12
A-36	Control of Heavy Loads Near Spent Fuel (former USI)	20.1.13
A-39	Determination of Safety Relief Valve Pool Dynamic Loads and Temperature Limits (former USI)	20.1.14
A-40	Seismic Design Criteria - Short-Term Program (former USI)	20.1.15
A-42	Pipe Cracks in BWR (former USI)	20.1.16
A-44	Station Blackout (former USI)	20.1.17
A-47	Safety Implications of Control Systems (former USI)	20.1.18
A-48	Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment	20.1.19
B-5	Ductility of Two-Way Slabs and Shells and Buckling Behavior of Steel Containments	20.1.20
B-10	Behavior of BWR Mark III Containments	20.1.21
B-17	Criteria for Safety-Related Operator Actions	20.1.22
B-36	Develop Design, Testing, and Maintenance Criteria for Atmosphere Cleanup System Air Filtration and Adsorption Units for ESF Systems and Normal Ventilation Systems	20.1.25
B-55	Improved Reliability of Target Rock SRVs	20.1.26

Table 20-1 (Continued)

Issue Designation	Title	FSER Section(s)
B-56	Diesel Reliability	20.1.27
B-61	Allowable ECCS Equipment Outage Periods	20.1.28
B-63	Isolation of Low-Pressure Systems Connected to the Reactor Coolant Pressure Boundary	20.1.29
B-66	Control Room Infiltration Measurements	20.1.30
C-1	Assurance of Continuous Long-Term Capability of Hermetic Seals on Instrumentation and Electrical Equipment	20.1.31
C-8	Main Steam Line Leakage Control Systems	20.1.32
C-10	Effective Operation of Containment Sprays in a LOCA	20.1.33
C-17	Interim Acceptance Criteria for Solidification Agents for Radioactive Solid Wastes	20.1.34
15	Radiation Effects on Reactor Vessel Supports	20.2.1
23	Reactor Coolant Pump Seal Failures	20.2.2
25	Automatic Air Header Dump on BWR Scram System	20.2.3
29	Bolting Degradation or Failure in Nuclear Power Plants	20.2.4
40	Safety Concerns Associated With Pipe Breaks in the BWR Scram System	20.2.5
45	Inoperability of Instrumentation Due to Extreme Cold Weather	20.2.6
51	Proposed Requirements for Improving the Reliability of Open Cycle SWSs	20.2.7
57	Effects of Fire Protection System Actuation on Safety-Related Equipment	20.2.8
67.3.3	Steam Generator Staff Actions - Improved Accident Monitoring	20.2.9
75	Generic Implications of ATWS Events at the Salem Nuclear Plant	20.2.10
78	Monitoring of Fatigue Transient Limits for RCS	20.2.11
82	Beyond Design-Basis Accidents in Spent Fuel Pools	20.2.12
83	Control Room Habitability	20.2.13
86	Long-Range Plan for Dealing with Stress Corrosion Cracking in BWR Piping	20.2.14
87	Failure of HPCI Steam Line Without Isolation	20.2.15
89	Stiff Pipe Clamps	20.2.16

Generic Issues

Table 20-1 (Continued)

Issue Designation	Title	FSER Section(s)
103	Design for Probable Maximum Precipitation	20.2.17
105	Interfacing Systems LOCA at LWRs	20.1.29 20.2.19
106	Piping and Use of Highly Combustible	20.2.20
113	Dynamic Qualification Testing of LBHSs	20.2.21
118	Tendon Anchorage Failure	20.2.22
120	On-Line Testability of Protection Systems	20.2.23
121	Hydrogen Control for Large, Dry PWR Containments	20.2.24
124	Auxiliary Feedwater System Reliability	20.2.25
128	Electrical Power Reliability	20.2.26
142	Leakage Through Electrical Isolators in Instrumentation Circuits	20.2.28
143	Availability of Chilled Water Systems	20.2.29
145	Actions to Reduce Common Cause Failures	20.2.30
151	Reliability of RPT During an ATWS	20.2.31
153	Loss of Essential Service Water in LWRs	20.2.32
155.1	More Realistic Source Term Assumptions	20.2.33
HF 1.1	Staffing and Qualifications - Shift Staffing	20.3.1
HF 4.4	Procedures - Guidelines for Upgrading Other Procedures	20.3.2
HF 5.1	Man-Machine Interface - LCSs	20.3.3
HF 5.2	Man-Machine Interface - Review Criteria for Human Factors Aspects of Advanced Controls and Instrumentation	20.3.4
I.A.1.1	Operating Personnel -- Operating Personnel and Staffing - STA	20.4.1
I.A.1.2	Operating Personnel -- Operating Personnel and Staffing - Shift Supervisor Administrative Duties	20.4.2
I.A.1.3	Operating Personnel -- Operating Personnel and Staffing - Shift Manning	20.4.3
I.A.1.4	Operating Personnel -- Operating Personnel and Staffing - Long-Term Upgrading	20.4.4

Table 20-1 (Continued)

<u>Issue Designation</u>	<u>Title</u>	<u>FSEER Section(s)</u>
I.A.2.1	Operating Personnel -- Training and Qualification of Operating Personnel - Immediate Upgrading of Operator and Senior Operator Training and Qualifications	20.4.5
I.A.2.3	Operating Personnel -- Training and Qualifications of Operating Personnel - Administration of Training Programs	20.4.6
I.A.2.6(1)	Operating Personnel -- Training and Qualifications of Operating Personnel - Long-Term Upgrading of Training and Qualifications; Revise RG 1.8	20.4.7
I.A.3.1	Operating Personnel -- Licensing and Requalification of Operating Personnel - Revise Scope of Criteria for Licensing Examinations	20.4.8
I.A.4.1(2)	Operating Personnel -- Licensing and Requalification of Operating Personnel - Initial Simulator Improvement; Interim Changes in Training Simulators	20.4.9
I.A.4.2	Operating Personnel -- Simulator Use and Development - Long-Term Training Upgrade	18.7.2.2 20.4.10 20.5.13
I.C.1	Operating Procedures - Short-Term Accident Analysis and Procedures Revision	20.4.11
I.C.2	Operating Procedures - Shift Relief and Turnover Procedures	20.4.12
I.C.3	Operating Procedures - Shift Supervisor Responsibilities	20.4.13
I.C.4	Operating Procedures - Control Room Access	20.4.14
I.C.5	Operating Procedures - Procedures for Feedback of Operating Experience to Plant Staff	13.2 13.5 20.4.15 20.5.41
I.C.6	Operating Procedures - Procedures for Verification of Correct Performance of Operating Activities	20.4.16 20.5.41
I.C.7	Operating Procedures -- NSSS Vendor Review of Procedures	20.4.17
I.C.8	Operating Procedures -- Pilot Monitoring of Selected Emergency Procedures for NTOL Applicants	20.4.18
I.D.1	Control Room Design - Control Room Design Reviews	18.7.2.2 20.4.20 20.5.15
I.D.2	Control Room Design - Plant Safety Parameter Display Console	7.5.2 18.7.2.2 20.4.21 20.5.16

Generic Issues

Table 20-1 (Continued)

<u>Issue Designation</u>	<u>Title</u>	<u>FSER Section(s)</u>
I.D.3	Control Room Design - Safety System Status Monitoring	18.7.2.2 20.4.22 20.5.17
I.D.5(2)	Control Room Design - Plant Status and Post-Accident Monitoring	20.4.23
I.D.5(3)	Control Room Design - On-Line Reactor Surveillance System	20.4.24
I.F.2	Quality Assurance - Develop More Detailed Quality Assurance (QA) Criteria	20.4.26 20.5.43
I.G.1	Preoperational and Low-Power Testing - Training Requirements	13.2 20.4.27
I.G.2	Preoperational and Low-Power Testing - Scope of Test Program	20.4.28 14.2
II.B.1	Consideration of Degraded or Melted Cores in Safety Review - RCS Vents	5.2.2 20.4.29 20.5.18
II.B.2	Consideration of Degraded or Melted Cores in Safety Review - Plant Shielding to Provide Access to Vital Areas and Protect Safety Equipment for Post-Accident Operation	12.3.5.1 12.3.6 13.6.3.5 20.4.30 20.5.19
II.B.3	Consideration of Degraded or Melted Cores in Safety Review - Post-Accident Sampling	9.3.2.2 20.4.31 20.5.20
II.B.4	Consideration of Degraded or Melted Cores in Safety Review - Training for Mitigating Core Damage	20.4.32
II.B.8	Consideration of Degraded or Melted Cores in Safety Review - Rulemaking Proceeding on Degraded Core Accidents	6.2.5 9.3.1 20.4.33 20.5.1 20.5.21 20.5.44 20.5.45
II.D.1	RCS Relief and Safety Valves - Testing Requirements	20.4.34 20.5.22 5.2.2 3.9.3.2

Table 20-1 (Continued)

Issue Designation	Title	FSER Section(s)
II.D.3	RCS Relief and Safety Valves - Relief and Safety Valve Position Indication	20.4.35 20.5.23 6.3.3 5.2.2
II.E.1.3	System Design -- AFW System - Update the SRP and Develop Regulatory Guidance	20.4.38
II.E.4.1	System Design -- Containment Design - Dedicated Penetrations	20.4.40 20.5.46 6.3.5
II.E.4.2	System Design -- Containment Design - Isolation Dependability	20.4.41 20.5.26 6.2.4.1
II.E.6.1	System Design -- In-Situ Testing of Valves - Test Adequacy Study	20.4.44
II.F.1	Instrumentation and Controls (I&C) - Additional Accident Monitoring Instrumentation	20.4.46 20.5.29 12.3.6 12.3.4 11.5.2 11.5.1
II.F.2	I&C - Identification of and Recovery from Conditions Leading to Inadequate Core Cooling	20.4.47 20.5.30 6.3
II.F.3	I&C - Instruments for Monitoring Accident Conditions	20.4.48 20.5.31
II.J.4.1	General Implications of TMI for Design and Construction Activities - Revise Deficiency Reporting Requirements	20.4.51
II.K.1(5)	Measures to Mitigate SBLOCAs and Loss-of-Feedwater Accidents -- IE Bulletins - Safety-Related Valve Position Description	20.4.52 18.7.2.2
II.K.1(10)	Measures to Mitigate SBLOCAs and Loss-of-Feedwater Accidents -- IE Bulletins - Review and Modify Procedures for Removing Safety-Related Systems from Service	20.4.53
II.K.1(13)	Measures to Mitigate SBLOCAs and Loss-of-Feedwater Accidents -- IE Bulletins - Proposed TS Changes Reflecting Implementation of All Bulletin Items	20.4.54

Table 20-1 (Continued)

<u>Issue Designation</u>	<u>Title</u>	<u>FSER Section(s)</u>
II.K.1.(22)	Measures to Mitigate SBLOCAs and Loss-of-Feedwater Accidents -- IE Bulletins - Describe Automatic and Manual Actions for Proper Functioning of Auxiliary Heat Removal Systems When Feedwater System Not Operable	20.4.55 20.5.33 5.4.6
II.K.1.(23)	Measures to Mitigate SBLOCAs and Loss-of-Feedwater Accidents -- IE Bulletins - Describe Uses and Types of Reactor Vessel Level Indication for Automatic and Manual Initiation Safety Systems	20.4.56
II.K.3(3)	Measures to Mitigate SBLOCAs and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Report Safety and Relief Valve Failures Promptly and Challenges Annually	20.4.61
II.K.3(11)	Measures to Mitigate SBLOCAs and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Control Use of Power-Operated Relief Valves Supplied by Control Components, Inc., Until Further Review Complete	20.4.62
II.K.3(13)	Measures to Mitigate SBLOCAs and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Separation of HPCI and RCIC System Initiation Levels	20.4.63 20.5.5 5.4.6
II.K.3(15)	Measures to Mitigate SBLOCAs and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Modify Break Detection Logic to Prevent Spurious Isolation of HPCI and RCIC Systems	20.4.64
II.K.3(16)	Measures to Mitigate SBLOCAs and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Reduction of Challenges and Failures of Relief Valves; Feasibility Study and System Modification	20.4.65 20.5.6 5.2.2
II.K.3(17)	Measures to Mitigate SBLOCAs and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Report on Outage of ECCSs; Licensee Report and TS Changes	20.4.66
II.K.3(18)	Measures to Mitigate SBLOCAs and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Modification of ADS Logic; Feasibility Study and Modification for Increased Diversity for Some Event Sequences	20.4.67 20.5.7 6.3.3
II.K.3(21)	Measures to Mitigate SBLOCAs and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Restart of Core Spray and LPCI Systems on Low Level; Design and Modification	20.4.68 20.5.8 6.3
II.K.3(22)	Measures to Mitigate SBLOCAs and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Automatic Switchover of RCIC System Suction; Verify Procedures and Modify Design	20.4.69
II.K.3(24)	Measures to Mitigate SBLOCAs and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Confirm Adequacy of Space Cooling for HPCI and RCIC Systems	20.4.71 20.5.9 6.2.5 5.4.6

Table 20-1 (Continued)

Issue Designation	Title	FSER Section(s)
II.K.3(25)	Measures to Mitigate SBLOCAs and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Effect of Loss of ac Power on Pump Seals	20.4.72 20.5.3 5.4.1
II.K.3(27)	Measures to Mitigate SBLOCAs and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Provide Common Reference Level for Vessel Level Instrumentation	20.4.73
II.K.3(28)	Measures to Mitigate SBLOCAs and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Study and Verify Qualification of Accumulators on ADS Valves	20.4.74 20.5.10 7.3 6.3 5.2.2 3.11
II.K.3(30)	Measures to Mitigate SBLOCAs and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Revised SBLOCA Methods to Show Compliance with 10 CFR Part 50, Appendix A	20.4.75
II.K.3(31)	Measures to Mitigate SBLOCAs and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Plant-Specific Calculations to Show Compliance with 10 CFR 50.46	20.4.76 6.3
II.K.3(44)	Measures to Mitigate SBLOCAs and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Evaluation of Anticipated Transients with Single Failure to Verify no Significant Fuel Failure	20.4.77 15.1
II.K.3(45)	Measures to Mitigate SBLOCAs and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Evaluate Depressurization with Other Than Full ADS	20.4.78 20.5.11 9.6.3 6.3.3
II.K.3(46)	Measures to Mitigate SBLOCAs and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Response to List of Concerns from ACRS Consultant	20.4.79
III.A.1.1(1)	Emergency Preparedness and Radiation Effects -- Improve Licensee Emergency Preparedness - Short Term; Upgrade Emergency Preparedness, Implement Action Plan Requirements for Promptly Improving Licensee Emergency Preparedness	20.4.80
III.A.1.2	Emergency Preparedness and Radiation Effects -- Improve Licensee Emergency Preparedness - Short Term; Upgrade Emergency Preparedness	20.4.81 20.5.37 13.3
III.A.2.1	Emergency Preparedness and Radiation Effects -- Improve Licensee Emergency Preparedness - Long Term; Amend 10 CFR Part 50 and 10 CFR Part 50, Appendix E	20.4.82
III.A.2.2	Emergency Preparedness and Radiation Effects -- Improve Licensee Emergency Preparedness - Long Term; Development of Guidance and Criteria	20.4.83

Generic Issues

Table 20-1 (Continued)

Issue Designation	Title	FSER Section(s)
III.A.3.3	Emergency Preparedness and Radiation Effects -- Improving NRC Emergency Preparedness - Communications	20.4.84
III.D.1.1	Radiation Protection -- Radiation Source Control - Primary Coolant Sources Outside the Containment Structure	20.5.85 20.5.38
III.D.3.3	Radiation Protection --Worker Radiation Protection Improvement - Inplant Radiation Monitoring	20.4.86 20.5.39 12.5.1
III.D.3.4	Radiation Protection --Worker Radiation Protection Improvement - Control Room Habitability	20.4.87 20.5.40 9.4.1.1 6.4

Table 20-2 ABWR - CP/ML Rule Items

10 CFR 50.34(f) Paragraph	TMI Item	Title	FSER Section(s)
(1)(i)	II.B.8	Consideration of Degraded or Melted Cores in Safety Review - Rulemaking Proceeding on Degraded Core Accidents, "Design Alternatives from PRA"	20.5.1
(1)(ii)	II.E.1.1	System Design -- AFWS - AFWS Evaluation	20.5.2 20.4.36
(1)(iii)	II.K.2(16)	Measures to Mitigate SBLOCAs and Loss-of-Feedwater Accidents -- Commission Orders on B&W Plants - Impact of RCP Seal Damage Following SBLOCA With Loss of Offsite Power	20.5.3 20.4.59 5.41
(1)(iii)	II.K.3(25)	Measures to Mitigate SBLOCAs and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Effect of Loss of ac Power on Pump Seals	20.5.3 20.4.72 5.4.1
(1)(iv)	II.K.3(2)	Measures to Mitigate SBLOCAs and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Report on Overall Safety Effect of Power-Operated Relief Valve Isolation	20.5.4 20.4.60
(1)(v)	II.K.3(13)	Measures to Mitigate SBLOCAs and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Separation of HPCI and RCIC System Initiation Levels	20.5.5 20.4.63 5.4.6
(1)(vi)	II.K.3(16)	Measures to Mitigate SBLOCAs and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Reduction of Challenges and Failures of Relief Valves; Feasibility Study and System Modification	20.5.6 20.4.65 5.2.2

Table 20-2 (Continued)

<u>10 CFR 50.34(f) Paragraph</u>	<u>TMI Item</u>	<u>Title</u>	<u>FSER Section(s)</u>
(1)(vii)	II.K.3(18)	Measures to Mitigate SBLOCAs and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Modification of ADS Logic; Feasibility Study and Modification for Increased Diversity for Some Event Sequences	20.5.7 20.4.67 6.3.3
(1)(viii)	II.K.3(21)	Measures to Mitigate SBLOCAs and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Restart of Core Spray and LPCI Systems on Low Level; Design and Modification	20.5.8 20.4.68 6.3
(1)(ix)	II.K.3(24)	Measures to Mitigate SBLOCAs and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Confirm Adequacy of Space Cooling for HPCI and RCIC Systems	20.5.9 20.4.71 6.2.5 5.4.6
(1)(x)	II.K.3(28)	Measures to Mitigate SBLOCAs and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Study and Verify Qualification of Accumulators on ADS Valves	20.5.10 20.4.74 7.3 6.3 5.2.2 3.11
(1)(xi)	II.K.3(45)	Measures to Mitigate SBLOCAs and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Evaluate Depressurization with Other Than Full ADS	20.5.11 20.4.78 9.6.3 6.3.3
(1)(xii)	N/A	Evaluation of Alternative Hydrogen Control Systems	20.5.12
(2)(i)	I.A.4.2	Operating Personnel -- Simulator Use and Development - Long-Term Training Upgrade	20.5.13 20.4.10 18.7.2.2
(2)(ii)	I.C.9	Operating Procedures - Long-Term Program Plan Procedures for Upgrading of Procedures	20.5.14 20.4.19 18.7.2.2 13.5

Table 20-2 (Continued)

10 CFR 50.34(f) Paragraph	TMI Item	Title	FSER Section(s)
(2)(iii)	I.D.1	Control Room Design - Control Room Design Reviews	20.5.15 20.4.20 18.7.2.2
(2)(iv)	I.D.2	Control Room Design - Plant Safety Parameter Display Console	20.5.16 20.4.21 18.7.2.2 7.5.2
(2)(v)	I.D.3	Control Room Design - Safety System Status Monitoring	20.5.17 20.4.22 18.7.2.2
(2)(vi)	II.B.1	Consideration of Degraded or Melted Cores in Safety Review - RCS Vents	20.5.18 20.4.29 5.2.2
(2)(vii)	II.B.2	Consideration of Degraded or Melted Cores in Safety Review - Plant Shielding to Provide Access to Vital Areas and Protect Safety Equipment for Post-Accident Operation	20.5.19 20.4.30 13.6.3.5 12.3.6 12.3.5.1
(2)(viii)	II.B.3	Consideration of Degraded or Melted Cores in Safety Review - Post-Accident Sampling	20.5.20 20.4.31 9.3.2.2
(2)(ix)	II.B.8	Consideration of Degraded or Melted Cores in Safety Review - Rulemaking Proceeding on Degraded Core Accidents, "Hydrogen Control System"	20.5.21
(2)(x)	II.D.1	RCS Relief and Safety Valves - Testing Requirements	20.5.22 20.4.34 5.2.2 3.9.3.2
(2)(xi)	II.D.3	RCS Relief and Safety Valves - Relief and Safety Valve Position Indication	20.5.23 20.4.35 6.3.3 5.2.2
(2)(xii)	II.E.1.2	System Design -- AFWS - AFWS System Automatic Initiation and Flow Indication	20.5.247 20.4.37

Table 20-2 (Continued)

<u>10 CFR 50.34(f) Paragraph</u>	<u>TMI Item</u>	<u>Title</u>	<u>FSER Section(s)</u>
(2)(xiii)	II.E.3.1	System Design -- Decay Heat Removal - Reliability of Power Supplies for Natural Circulation	20.5.25 20.4.39
(2)(xiv)	II.E.4.2	System Design -- Containment Design - Isolation Dependability	20.5.26 20.4.41 6.2.4.1
(2)(xv)	II.E.4.4	System Design -- Containment Design - Purging	20.5.27 20.4.42 6.2.5
(2)(xvi)	II.E.5.1	System Design -- Design Sensitivity of B&W Reactors - Design Evaluation	20.5.28 20.4.43
(2)(xvii)	II.F.1	I&C - Additional Accident Monitoring Instrumentation	20.5.29 20.4.46 12.3.6 12.3.4 11.5.2 11.5.1
(2)(xviii)	II.F.2	I&C - Identification of and Recovery from Conditions Leading to Inadequate Core Cooling	20.5.30 20.4.47 6.3
(2)(xix)	II.F.3	I&C - Instruments for Monitoring Accident Conditions	20.5.31 20.4.48
(2)(xx)	II.G.1	Electrical Power - Power Supplies for Pressurizer Relief Valves, Block Valves, and Level Indicators	20.5.32 20.4.49
(2)(xxi)	II.K.1(22)	Measures to Mitigate SBLOCAs and Loss-of-Feedwater Accidents -- IE Bulletins - Describe Automatic and Manual Actions for Proper Functioning of Auxiliary Heat Removal Systems When Feedwater System Not Operable	20.5.33 20.4.55 5.4.6
(2)(xxii)	II.K.2(9)	Measures to Mitigate SBLOCAs and Loss-of-Feedwater Accidents -- Commission Orders on B&W Plants - Analysis and Upgrading of Integrated Control System	20.5.34 20.4.57

Table 20-2 (Continued)

<u>10 CFR 50.34(f) Paragraph</u>	<u>TMI Item</u>	<u>Title</u>	<u>FSER Section(s)</u>
(2)(xxiii)	II.K.2(10)	Measures to Mitigate SBLOCAs and Loss-of-Feedwater Accidents -- Commission Orders on B&W Plants - Hard-Wired Safety-Grade Anticipatory Reactor Trips	20.5.35 20.4.58
(2)(xxiv)	II.K.3(23)	Measures to Mitigate SBLOCAs and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Central Water Level Recording	20.5.36 20.4.70
(2)(xxv)	III.A.1.2	Emergency Preparedness and Radiation Effects -- Improve Licensee Emergency Preparedness - Short Term; Upgrade Emergency Preparedness	20.5.37 20.4.81 13.3
(2)(xxvi)	III.D.1.1	Radiation Protection -- Radiation Source Control - Primary Coolant Sources Outside the Containment Structure	20.5.38 20.4.85
(2)(xxvii)	III.D.3.3	Radiation Protection --Worker Radiation Protection Improvement - Inplant Radiation Monitoring	20.5.39 20.4.86 12.5.1
(2)(xxviii)	III.D.3.4	Radiation Protection --Worker Radiation Protection Improvement - Control Room Habitability	20.5.40 20.4.87 9.4.1.1 6.4
(3)(i)	I.C.5	Operating Procedures - Procedures for Feedback of Operating Experience to Plant Staff	20.5.41 20.4.16
(3)(ii)	I.F.1	Quality Assurance (QA) - Expand QA List	20.5.42 20.4.25
(3)(iii)	I.F.2	Quality Assurance - Develop More Detailed QA Criteria	20.5.43 20.4.26
(3)(iv)	II.B.8	Consideration of Degraded or Melted Cores in Safety Review - Rulemaking Proceeding on Degraded Core Accidents, ".91-Meter (3-Foot) Diameter Equivalent Dedicated Containment Penetration"	20.5.44

Table 20-2 (Continued)

<u>10 CFR 50.34(f) Paragraph</u>	<u>TMI Item</u>	<u>Title</u>	<u>FSER Section(s)</u>
(3)(v)	II.B.8	Consideration of Degraded or Melted Cores in Safety Review - Rulemaking Proceeding on Degraded Core Accidents, "Containment Integrity During an Accident Involving 100-Percent Fuel Clad Metal-Water Reaction"	20.5.45
(3)(vi)	II.E.4.1	System Design -- Containment Design - Dedicated Penetrations	20.5.46 20.4.40 6.3.5
(3)(vii)	II.J.3.1	General Implications of TMI for Design and Construction Activities -- Management for Design and Construction - Organization and Staffing to Oversee Design and Construction	20.5.47 20.4.50

Compliance with paragraph (1)(ii) of 10 CFR 52.47(a)

Paragraph (1)(ii) of 10 CFR 52.47(a) requires an application for a design certification to include a demonstration of compliance with any technically relevant portions of the TMI requirements identified in 10 CFR 50.34(f), often called the CP/ML rule.

GE addressed the TMI requirements of paragraph (1)(ii) of 10 CFR 52.47(a) in SSAR Appendix 19A and in other SSAR sections. In the DFSE, the staff provided evaluations for the majority of GE's submittal relating to compliance with this regulatory requirement and indicated that the additional items not included in the DFSE would be included in the final safety evaluation report (FSER). A number of the staff's evaluations contained open items, combined operating license (COL) action items, and/or technical specification (TS) items that needed to be addressed by GE. The closure of these items for the ABWR design certification review is discussed in the appropriate issue evaluations.

The advance SER stated that based on the staff's evaluation of the issues listed in Table 20-2 of the SER and contingent on GE's incorporation of agreed-to issue mark-ups in the SSAR, the staff concluded that GE adequately demonstrated compliance with or proposed a method of compliance for the technically relevant portions of 10 CFR 50.34(f) as required by 10 CFR 52.47(a)(1)(ii) for the ABWR design, with some exceptions that required further GE or staff action. These exceptions were that (1) 10 CFR 50.34(f)(2)(xviii) (TMI Item II.F.2) was still open pending the resolution of the differences of views between the staff and GE on the need for diverse instrumentation for reactor pressure vessel water level indication, and (2) 10 CFR 50.34(f)(2)(vi) (TMI Item II.B.1) required incorporation of a COL action item in the ABWR SSAR. These exceptions have been addressed as discussed in Sections 20.5.30 and 20.5.18, respectively, of this report, and GE has incorporated agreed-to issue mark-ups in the ABWR SSAR. Based on this information and the staff's review of the issues listed in Table 20-2 of this report, the staff concludes that GE has adequately demonstrated compliance with 10 CFR 52.47(a)(1)(ii) for the ABWR design.

Incorporation of operating experience in the ABWR design

In a staff requirements memorandum dated February 15, 1991, on SECY-90-377, "Requirements for Design Certification Under 10 CFR Part 52," the Commission directed the staff to ensure that the design certification process preserves operating experience insights in the certified design. As discussed in Section 20.6 of this

report, the staff concludes that GE has adequately considered operating experience identified by generic letters or bulletins issued since 1980 in the ABWR design.

20.1 Task Action Plan Items

This section addresses staff evaluation of USIs and GSIs that are categorized as "task action plan items" in NUREG-0933. All the following issues, with the exception of Issues B-29 and B-32, are relevant to the ABWR design. Issues B-29 and B-32 were evaluated in the DFSE and are discussed here for continuity only.

20.1.1 Issue A-1: Water Hammer (former USI)

Water hammer is defined as a rapid change in pressure caused by a change in velocity of a fluid in a closed volume. Water hammer occurs in various piping systems, such as emergency core cooling, residual heat removal (RHR), containment spray, service water, and feedwater and in steam lines. Water hammer may be caused by rapid condensation of steam pockets, steam-driven slugs of water, pump startup with partially empty lines, and rapid valve motion. Regardless of the cause, water hammer may result in a rapid acceleration of the fluid and may affect the piping system. Severity of the damage from water hammer may range from overstressing of pipe hangers to major damage of restraints, piping, and components.

The review criteria for this issue are stated in the following sections of NUREG-0800, "Standard Review Plan" (SRP): 5.4.7, 6.3, 9.2.1, 9.2.2, and 10.4.7 and Branch Technical Position (BTP) ASB 10-2. Specifically, the feedwater system, containment spray system, shutdown cooling, and other safety-related systems that may be adversely affected by water hammer should be designed to withstand the dynamic loads associated with water hammer. BTP ASB 10-2 requires that the feedwater system be subjected to preoperational testing to demonstrate the effectiveness of the design and operating procedures to mitigate the effects of water hammer.

SSAR Section 19B.2.2 indicates that all of the ABWR systems having potential for water hammer have been analyzed. Different forms of initiating events that could occur and which cause water hammer were considered, such as steam condensation, steam-driven slugs of water, pump startup with partially empty lines, and rapid valve cycling. Section 19B.2.2 states that GE has evaluated various systems for potential water hammer, including the condensate and feedwater system, main steam lines, and all components of the main steam supply system.

With regard to leak before break, GE states that feedwater lines were demonstrated to be immune to failure from

Generic Issues

water hammer effects. Reactor core isolation cooling (RCIC), high pressure core flooders (HPCF), and RHR systems are precluded from water hammer by keep-full features and the absence of fast-acting valves.

GE also states that the systems susceptible to water hammer effects will be kept full of water, thus preventing water hammer when pumps are started from a steady condition. These systems include the reactor service water (RSW); turbine service water; RHR; HPCF; RCIC; and heating, ventilation, and air conditioning (HVAC) emergency cooling water system. Based on the above, the staff concludes that the ABWR design meets the guidelines in the SRP sections listed above with respect to the dynamic effects associated with possible fluid flow instabilities, such as water hammer. The staff further concludes that GE adequately addressed this issue for the ABWR design.

20.1.2 Issue A-7: Mark I Long-Term Program (former USI)

During testing for an advanced BWR containment system design (Mark III), GE identified suppression pool hydrodynamic loads that had not been considered in the original design of the Mark I containment system. To address this issue, a Mark I Owners Group was formed and the assessment was divided into a short-term and a long-term program. The long-term program was conducted to provide a generic basis to define suppression pool hydrodynamic loads and the related structural acceptance criteria so that a comprehensive reassessment of each Mark I containment system would be performed. A series of experimental and analytical programs was conducted by the Mark I Owners Group. The program proposed to the NRC and reviewed and modified by the staff was to be used to perform plant-unique analyses and identify modifications, as necessary.

The review criteria for this issue are to establish design-basis, conservative loads that are appropriate for the anticipated life of each Mark I BWR containment and to restore the originally intended design margin of safety for the containment system. The principal thrust of the long-term program has been the development of generic methods for the definition of suppression pool hydrodynamic loadings and the associated structural assessment techniques for the Mark I configuration.

It is recognized that the Mark I torus pool and vent configuration is different from the ABWR annular pool and vent design. Therefore, the local loads evaluated within the issue are not applicable to the ABWR design. However, while the results of the Mark I Owners Group investigation cannot be directly applicable for definition of

the safety-relief valve (SRV) loads in the ABWR design, they are used as a data base for definition of the SRV loads that are specified in Issue A-39 as discussed in Section 20.1.14 of this report.

20.1.3 Issue A-8: Mark II Containment Pool Dynamic Loads - Long-Term Program

This issue deals with the new containment loads associated with the postulated loss-of-coolant accidents (LOCAs) that were identified as a result of tests by GE. These loads result from the dynamic effects of drywell air and steam being rapidly forced into the suppression pool during a postulated LOCA event. These loads, as well as the loads from actuation of SRVs in the Mark II containment, had not been previously accounted for.

The review criteria for this issue are contained in NUREG-0808, "Mark II Containment Program Evaluation and Acceptance Criteria," and SRP Section 6.2.1.1C and Appendices A and B. SRP Section 6.2.1.1C, Appendix A pertains to steam bypass from the drywell to the suppression pool air volume in the Mark I, II, and III containment designs and states that the system used to quench steam bypassing the suppression pool should be designed so that the steam bypass capability for small breaks satisfies the specified criteria. It also states that the bypass leakage should not substantially increase over the life of a plant. SRP Section 6.2.1.1C, Appendix B summarizes the generic loads acceptable to the NRC and provides information regarding load identification, a summary of the load specification, load specification clarifying criteria, and a reference to the NRC NUREG section that describes the NRC-specific load evaluation.

SSAR Section 3B.4.2.1 states that pool swell response calculations to quantify pool swell loads were based on a simplified, one-dimensional analytical model (described in NEDE-21544-P, "Mark II Pressure Suppression Containment Systems: Analytical Model of the Pool Swell Phenomenon"), which is the same as that reviewed and accepted by the staff in NUREG-0808 for application to Mark II plants. This analytical model was qualified against Mark II full-scale test data. It is recognized that although ABWR wetwell airspace is similar to that of the Mark II design, its vent system design is quite different. Therefore, recognizing the difference in vent system design, additional studies comparing model versus Mark III horizontal vent test data were performed to assure adequacy of the model for application to the ABWR. The staff concludes that this approach adequately addresses this issue for the ABWR since the differences between the ABWR design and the Mark II and Mark III containments have been taken into account in the analysis.

AIM

Association for Information and Image Management

1100 Wayne Avenue, Suite 1100
Silver Spring, Maryland 20910
301/587-8202

Centimeter

Inches

MANUFACTURED TO AIM STANDARDS
BY APPLIED IMAGE, INC.

9 of 10

20.1.4 Issue A-9: Anticipated Transient Without Scram (ATWS) (former USI)

This issue deals with the problem of occurrence of transients that require scram but for which scram does not occur. It involves devising measures, both design and operational, that can be taken to avoid or compensate for such occurrences.

The staff's technical findings on this issue were documented in NUREG-0460, "Anticipated Transients Without Scram for Light Water Reactors," and the requirements for resolution are contained in 10 CFR 50.62 (also known as the ATWS rule). For BWRs, 10 CFR 50.62 requires the reactor to have an alternate rod injection (ARI) system that is diverse (from the reactor trip system) from the sensor output to the final actuation device. This system is also required to have redundant scram air header exhaust valves and must be designed to perform its function reliably and be independent of the existing reactor trip system from sensor output to final actuation device. The ATWS rule also requires that each BWR have a standby liquid control system (SLCS) that has the capability of injecting a borated water solution into the reactor pressure vessel (RPV). The borated water must be of such flow rate, boron concentration, and boron-10 isotope enrichment that, when accounting for the RPV volume, the resulting reactivity control is equivalent to that resulting from the injection of 326 Lpm (86 gpm) of 13 weight percent sodium pentaborate decahydrate solution at the natural boron-10 isotope abundance into a 638 cm (251 in.) internal diameter reactor pressure vessel for a given core design. The SLCS must be automatically initiated and the system and its injection location must be designed to perform their functions in a reliable manner. Each BWR must also have equipment designed to reliably trip the reactor coolant recirculation pumps under conditions indicative of an ATWS.

SSAR Sections 15.8 and 19B.2.5 indicate that there are two ways to provide scram in the ABWR design: a motor-driven way and a hydraulic way. In response to a scram signal, the control rods are inserted hydraulically, by the stored energy in the scram accumulator, similar to the current operating BWRs. A scram signal is given simultaneously to insert the fine motion control rod drives (FMC RDs) electrically, via the FMC RD motor drive. This diversity (hydraulic and electric methods of scrambling) provides a high degree of assurance that control rods will be inserted when needed.

The ABWR has equipment to trip the reactor internal pumps (RIPs) automatically under ATWS conditions. The RIPs are automatically tripped on reactor high pressure 7860 kPa (1125 psig) (RIPs not connected to the motor-

generator set) and RPV Level 2 (RIPs connected to the motor-generator set).

The ABWR design provides recirculation runback for all scram signals, feedwater runback on reactor high pressure and startup range neutron monitoring system ATWS permissive for 2 minutes.

In SSAR Appendix 15E, GE provides ATWS performance evaluation for fuel integrity, containment integrity, primary system, and long-term shutdown cooling. GE states that all requirements of satisfactory performance in the case of an ATWS are met.

As discussed in Section 15.5.2 of this report, the staff performed audit calculations to verify that the ABWR design is satisfactory to mitigate the effects of an ATWS. The review focused on the consequences of manual SLCS actuation and no recirculation pump runback on scram signals other than reactor high pressure and reactor low level. Under some circumstances, a problem was identified regarding the power shift to the top of the core. The staff concluded that the new design of recirculation runback on any scram signal and any ARI FMC RD run-in signal ensure that there is no potential for any unacceptable power shift to the top of the core.

Based on the information provided by GE and the staff evaluation cited above, the staff concludes that GE has adequately addressed this issue for the ABWR. The ABWR complies with the ATWS rule (10 CFR 50.62), as discussed in Section 15.5 of this report.

20.1.5 Issue A-10: BWR Feedwater Nozzle Cracking (former USI)

Inspections of operating BWRs conducted up to April 1978 revealed cracks in the feedwater nozzles of 20 reactor vessels. These cracks ranged in depth from 1.3-1.90 cm (0.5-0.75 in.), including cladding. One crack penetrated the cladding to the base metal for a total depth of approximately 3.8 cm (1.5 in.). It was determined that cracking resulted from high-cycle thermal fatigue caused by fluctuations in water temperature within the vessel in the nozzle region. These fluctuations occurred during periods of low feedwater circulation when the flow was unsteady and intermittent. Once started, the cracks grew because of thermal cycling during startups and shutdowns.

The review criteria for this issue are stated in NUREG-0619, "BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking." This document states that the thermal fatigue and crack initiation of feedwater nozzles are caused by the incoming feedwater, which is considerably colder than the water in the reactor

Generic Issues

vessel. This is especially true during reactor startup (before feedwater heaters are in service) and during shutdown (after heaters are taken out of service). Turbulent mixing of the hot water returning from the steam separators and dryers and the incoming cold feedwater causes thermal stress cycling of the nozzle unless it is protected by a thermal sleeve.

The proposed design for the ABWR will require that an inner thermal sleeve leading the cooler feedwater to the feedwater sparger be welded to the nozzle safe end. The welded thermal sleeve will assure that there is no leakage of cold feedwater between the thermal sleeve and the safe end. A secondary thermal sleeve is to be placed concentrically in the annulus between the inner thermal sleeve and the nozzle bore to prevent cold water that may be shedding from the outside surface of the inner sleeve from impinging on the nozzle bore and the inside nozzle corner. This proposed double-sleeve design gives a low fatigue usage factor in the nozzle bore and at the inner nozzle corner.

The material of the nozzle forging is SA-508, Class 3, low-alloy steel and the material of the safe end is SA-508, Class 1, carbon steel. The carbon steel safe end is welded to the nozzle forging with a carbon steel weld.

The double thermal sleeve as applied to the ABWR has not been used in earlier plants, although the Monticello (U.S.) and Tsuruga (Japan) plants are using similar designs. A telephone conference with Northern States Power, owner of the Monticello Power Plant, disclosed that the double thermal sleeve performs satisfactorily.

GE proposed an inservice inspection (ISI) program consisting of the following:

- Ultrasonic examination from the external surface of the nozzle ends, nozzle bores, and nozzle blend radius every second outage. If indications are found in the safe ends, the indications will be evaluated per the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI.
- Visual inspection of flow holes and welds in sparger arms and sparger tees every fourth outage.
- Visual inspection of accessible areas of the nozzles from the inner diameter surface on the ASME Code, Section XI, schedule as core internal components.

The ISI program described above is acceptable to the staff because the reactor feedwater nozzles will be ultrasonically and visually examined in service according to ASME Code, Section XI, schedules and inspection criteria. This

should ensure that the reactor vessel nozzles will perform in service as designed.

On the basis of the above, the staff concludes that GE adequately addressed this issue for the ABWR design.

20.1.6 Issue A-13: Snubber Operability Assurance

In May 1978, the Advisory Committee on Reactor Safeguards (ACRS) staff observed that there are many licensee event reports (LERs) related to the malfunction of snubbers. The most common LERs involve (1) seal leakage in hydraulic snubbers and (2) high rejection rate during functional testing of snubbers. Snubbers are used as seismic and pipe whip restraints at operating plants. Their safety function is to provide supports to systems or components under dynamic load conditions such as earthquakes and severe hydraulic transients, e.g., pipe breaks. When snubbers are used as vibration arrestors, their fatigue strength must be considered.

The review criteria for this issue are contained in SRP Section 3.9.3. This section states that systems and components that utilize snubbers as shock and vibration arrestors must be analyzed to ascertain their interaction with the systems and components to which they are attached. Snubbers used as shock arrestors do not require fatigue evaluation if it can be demonstrated that certain conditions are satisfied. The criteria for inspection and testing of snubbers are also provided.

SSAR Section 19B.2.7 refers to Section 3.9.3.4.1(3), which in turn, provides the information pertinent to snubber operability assurance. The information consists of the design parameters regarding the required load capacity and snubber location, inspection, testing, replacement, design and testing, installation requirements, and preservice examination. As discussed in Section 3.12.6.6 of this report, the staff concluded that the information provided by GE is consistent with applicable portions of SRP Section 3.9.3 and, therefore, is acceptable. On the basis of the above, the staff concludes that GE adequately addressed this issue for the ABWR design.

20.1.7 Issue A-17: Systems Interaction (former USI)

Nuclear power plants contain many structures, systems, and components (SSCs), some of which are safety related. Some of these SSCs are designed to interact to perform their intended functions and are usually well recognized and accounted for in the evaluation of plant safety by designers and in plant safety assessments. Several significant, plant-specific events have involved unintended or unrecognized dependencies among various SSCs. Some of these events have involved subtle dependencies between

safety-related and non-safety-related SSCs and some have even involved subtle dependencies between redundant and supposedly independent SSCs. These dependencies have been termed "adverse systems interactions" (ASIs). This issue was instituted to investigate the potential that these ASIs have remained hidden and could lead to safety-significant events. Seismically induced systems interactions, originally covered in Issue A-46 (which applies only to operating plants), is covered for the ABWR in current licensing requirements.

The staff's technical findings are documented in NUREG-1174, "Evaluation of Systems Interactions in Nuclear Power Plants," and the regulatory analysis in NUREG-1229, "Regulatory Analysis for Resolution of USI A-17." The staff informed the Commission of the resolution in SECY-89-230, "Unresolved Safety Issue A-17, 'Systems Interactions in Nuclear Power Plants'." These documents provide adequate guidance on this issue, although it was resolved by the staff without the establishment of new requirements. Generic Letter (GL) 89-18 was issued to licensees and applicants on September 6, 1989, to inform them about the resolution of this issue.

SSAR Section 19B.2.59 describes the studies on the subject of ASIs that were carried out over the last 10 years and the ABWR design features that could prevent and/or mitigate them. These studies were performed by various organizations, such as NRC's Office of Analysis and Evaluation of Operational Data (AEOD), the Institute of Nuclear Power Operations (INPO), nuclear steam supply system (NSSS) vendors, and the NRC staff based on the operating experience that is available in various publications, such as generic letters, information notices, and bulletins. These studies allowed plant system designers to formulate certain attributes that could be incorporated into design of the ABWR. These attributes consist of the separation criteria, consideration of failure aspects (such as fail safe, diversity, redundancy), protective actions (such as auto versus manual), and so on.

GE states that consideration of ASIs has resulted in designing the ABWR to explicitly avoid unwanted, unacceptable, or unknown ASIs. This has been accomplished through such features as multiple fission barriers, inherent shutdown features and mechanisms, a redundant and diverse engineered safety features (ESFs) network, and a redundant and diverse instrumentation and controls (I&C) protection network.

The SSAR also compares the ABWR-unique features that are designed to prevent, mitigate, and accommodate ASIs with the features of other BWRs. These features include more redundant, diverse and independent power sources, RPV and containment makeup and cooling capabilities,

decay heat removal capabilities, and operator action capabilities. Also, the ABWR has more redundant fault-tolerant I&C protection and a more secure and protected ESF housing from fire and flood.

Based on review of the design aspects of the ABWR described above, the staff concludes that the ABWR reflects the proven technology and accepted design requirements and that GE adequately addressed the concerns of this issue.

20.1.8 Issue A-24: Qualification of Class 1E Safety-Related Equipment (former USI)

Construction permit (CP) applicants for which SERs were issued after July 1974 were required by the NRC to qualify all safety-related equipment in accordance with IEEE 323-1974, "Qualifying Class 1E Equipment for Nuclear Power Generating Stations." From the time that this standard was originated, methods to qualify equipment to IEEE 323 were developed by the industry, but some of them, such as testing margins, aging effects, and the simulation of the worst-case environments, have not been accepted by the NRC.

All major NSSS vendors and architect engineers submitted topical reports that describe their methods of qualification. These reports were reviewed by the NRC and the results documented in NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment." These requirements were later established in 10 CFR 50.49 and revised Regulatory Guide (RG) 1.89, "Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants," which describes acceptable methods for complying with 10 CFR 50.49. Dynamic and seismic qualification of Class 1E electrical equipment was not included in the scope of 10 CFR 50.49. Guidance on dynamic and seismic qualification is contained in RG 1.100 (Rev. 2), "Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants."

The criteria for this issue are contained in NUREG-0588 and in 10 CFR 50.49 for environmental qualification and RG 1.100 (Rev. 2), for dynamic and seismic qualification of Class 1E electrical equipment.

Dynamic and seismic qualification testing and analysis of the electrical equipment identified in SSAR Appendix 3I are addressed in SSAR Section 3.10, except for pump motors and valve motor operators which are addressed in Section 3.9. The tests and analyses are to be performed in accordance with IEEE-344, "IEEE Recommended Practices for Seismic Qualification of Class 1E Equipment

Generic Issues

for Nuclear Power Generating Stations," as modified and endorsed by RG 1.100.

Environmental qualification of safety-related mechanical and electrical equipment is described in SSAR Section 3.11. ABWR Class 1E electrical equipment, including pump and valve motors and electrical accessories, will be environmentally qualified by the methods documented in the NRC-approved report, NEDE-24326-1-P, "General Electric Environmental Qualification Program." These methods are in accordance with the guidance of IEEE 323-1974, NUREG-0588, RG 1.89 (Rev. 1), and the generic requirements of 10 CFR 50.49. Typical environmental conditions (temperature, pressure, humidity, integrated radiation dose, and exposure to chemicals) are provided in SSAR Appendix 3I and cover the design lifetime. Conditions are tabulated for normal operation in and outside of containment and for LOCAs and HELBs inside containment.

Environmental qualification tests and analyses are addressed in SSAR Section 3.11.2. The safety-related equipment in the areas of SSAR Appendix 3I is required to remain functional in the environmental conditions expected at the equipment location during and after the limiting DBA. Qualification tests and analyses of electrical equipment for the effects of aging, radiation, temperature, humidity, chemical spray, submergence, and power supply variation, as applicable, are to be performed and the results documented in accordance with NEDE-24326-1-P.

The proposed qualification program for ABWR electrical equipment is acceptable as discussed further in Sections 3.10 and 3.11 of this report. Therefore, the staff concludes that GE adequately addressed this issue for the ABWR design.

20.1.9 Issue A-25: Non-Safety Loads on Class 1E Power Sources

Class 1E power sources provide the electric power for the plant systems that are essential to reactor shutdown, containment isolation, reactor core cooling, and containment heat removal, and are otherwise essential in preventing significant release of radioactive material to the environment. In some cases, non-safety loads are supplied from the Class 1E power sources and if this is allowed, it is possible that the non-safety loads may cause degradation of the Class 1E power system by introducing loss of redundancy or by other failure mechanisms.

Resolution of this issue required that non-safety-related electrical equipment connected to the Class 1E power systems be limited and, if connected, conform to requirements (for example, independence, electrical

isolation, and physical separation) so that the Class 1E system to which the non-safety-related equipment is connected continues to meet the capacity, capability, independence, redundancy, and testability requirements of GDC 17.

The ABWR design incorporates three independent Class 1E diesel generators and a non-Class 1E combustion turbine generator. The combustion turbine generator is designed to automatically assume the majority of non-safety-related electrical equipment independently from the Class 1E diesel generators. Therefore, it is not necessary for non-safety-related electrical equipment to be connected to the Class 1E system.

The ABWR design excludes non-safety-related electrical equipment from the Class 1E system, with the exception of the fine-motion control rod drive subsystem and a portion of the lighting subsystem. The fine motion control rod drive subsystem meets Class 1E requirements from the Class 1E system buses to and including zone-select interlocks (isolation devices). In addition, the fine motion control rod drive subsystem is restricted to Division I in order to assure that the Class 1E subsystems do not violate their independence requirements. The lighting subsystem meets Class 1E requirements from the Class 1E system buses to and including the subsystem load. Because lighting fixtures and bulbs are not seismically qualified in accordance with Class 1E requirements, protective devices (breakers or fuses) and their coordination is provided to assure that Class 1E systems meet their independence and redundancy requirements. The Class 1E system is also sized with sufficient capacity to accommodate operation and failure of the connected non-safety-related subsystems.

The staff concludes that the connection of non-safety-related electrical equipment to the Class 1E system has been appropriately limited. It also concludes that the Class 1E system (with the limited number of connected non-safety-related electrical subsystems) meets the capacity, capability, independence, redundancy, and testability requirements of GDC 17. Therefore, the staff concludes that GE adequately addressed this issue for the ABWR design.

20.1.10 Issue A-29: Nuclear Power Plant Design for the Reduction of Vulnerability to Industrial Sabotage

Reduction of the vulnerability of reactors to radiological sabotage is currently treated as a plant physical security function and not as a plant design requirement. Although present reactor designs provide a great deal of inherent protection against industrial sabotage, extensive physical security measures are still required to provide an

acceptable level of protection. An alternative approach would be to consider more fully reactor vulnerabilities to sabotage during the preliminary design phase. Because emphasis is being placed on standardizing plants, it is especially important to consider measures that could reduce the vulnerability of reactors to sabotage. Any design features to enhance physical protection must be consistent with system safety requirements.

The staff resolved this issue without the establishment of new requirements. However, future plants may decrease vulnerability to sabotage by several means. Each division of safety system functions should be totally independent and separated, both mechanically and electrically. Each division should also include 3-hour fire barriers, physical protection of each division from flooding, and physical protection from pipe breaks, both inside and outside the containment. The site security system requirements should be compatible with the plant arrangement and safety system design, definition of vital systems, layout of vital components, security barriers, intrusion detection systems, isolation zone requirements, security alarms, access control, security communications, power supply, and data management. Since there exists a potential for sabotage by a "knowledgeable insider" with authorized access or for acts of sabotage that could occur during maintenance activities, advanced light water reactor (ALWR) plant designers should also analyze the vulnerability of their designs to insider sabotage before finalizing the designs.

SSAR Section 19B.2.4 contains a summary that describes this issue, the Electric Power Research Institute (EPRI) ALWR requirements document, and the proposed resolution of Issue A-29 for the ABWR. It states that the ABWR design will comply with the ALWR requirements document as defined in the SSAR. It also indicates that the ABWR design will mitigate the acts of sabotage through physical separations in the plant arrangement of engineering safety systems and the design and location of barriers to resist threats.

In SSAR Section 19B.2.4, GE states that a sabotage vulnerability analysis will be conducted before the design is finalized. The staff verified that GE established a COL action item in Table 1.9-1 to perform this analysis. This is an acceptable approach. Based on the above, the staff concludes that GE adequately addressed this issue for the ABWR design.

20.1.11 Issue A-31: RHR Shutdown Requirements (former USI)

This issue addresses the safe shutdown of the reactor following an accident or abnormal condition other than a LOCA from a hot standby (that is, the reactor is shut

down, but the primary system temperature and pressure are still at or near normal operating values) to a cold shutdown condition. Considerable emphasis has been placed on long-term cooling, which is achieved by the RHR system. The RHR starts to operate when the reactor coolant system (RCS) pressure and temperature are substantially lower than their hot-standby condition values.

Even though it may generally be considered safe to maintain a reactor in a hot-standby condition for a long time, experience has shown that there have been events that required eventual cooldown and long-term cooling until the reactor coolant is cool enough to perform inspection and repairs. For this reason, the ability to transfer heat from the reactor to the environment after a shutdown is an important safety function. It is essential that a power plant be able to go from hot-standby to a cold-shutdown condition after any abnormal occurrence.

The review criteria for this issue are contained in SRP Section 5.4.7, Revision 3. Specifically, the RHR system should meet the intent of the following:

- The design should be such that the reactor can be taken from normal operating to cold shutdown using only safety-grade systems that satisfy the criteria of GDC 1 through 5 of 10 CFR Part 50, Appendix A.
- The system(s) should have suitable redundancy in components and features, and suitable interconnections, leak connections, and isolation capabilities to assure that for onsite electrical power system operation (assuming that offsite power is not available), the system function can be accomplished assuming a single failure.
- The system is capable of being operated from the control room with either onsite or offsite power available.
- The system(s) should be capable of bringing the reactor to a cold shutdown condition, with either onsite or offsite power available, within a reasonable time following a shutdown, assuming the most limiting single failure.

In SSAR Section 19B.2.10, GE stated that the RHR system consists of three electrically and mechanically independent divisions, except for the outboard containment isolation valves, which are in different electrical divisions than the inboard valves. The system will be redundant so that its functional integrity will be assured for onsite electrical power system operation, when offsite power is not available, assuming a single failure. The RHR shutdown cooling subsystem will be activated manually by the

Generic Issues

operator from the control room following insertion of the control rods and normal blowdown to the main condenser. For emergency conditions, when one of the RHR loops has failed, the RHR system will be capable of bringing the reactor to the cold shutdown condition, i.e., 100 °C (212 °F), within 36 hours following reactor shutdown with two divisions. When all three RHR loops are functioning together, the RHR can remove residual heat (decay and sensible) from the reactor vessel water at a rate sufficient to cool it to 60 °C (140 °F) within 24 hours after the rods are inserted.

The ABWR RHR design does not meet SRP Section 5.4.7, BTP RSB 5-1, Sections B.1(b) and (c), which require the RHR suction side isolation valves to have independent diverse interlocks to prevent the valves from being opened unless RCS pressure is below the RHR system design pressure. Instead, the pressure signal that provides the interlock function is supplied from a 2-out-of-4 logic, which has four independent pressure sensor and transmitter inputs, each of which is in a separate instrument division. The staff concluded in Section 5.4.7 of this report that they satisfy the intent of BTP RSB 5-1. Therefore, the staff concludes that GE adequately addressed this issue for the ABWR design.

20.1.12 Issue A-35: Adequacy of Offsite Power Systems

GDC 17 of 10 CFR Part 50, Appendix A, requires provisions be included in the design to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated by the nuclear power unit, the loss of power from the transmission network, or the loss of power from the onsite electric power supplies.

To meet these requirements of GDC 17, the NRC, in the past, depended on (1) the results of transient and steady-state stability analyses documented in the safety analysis reports for license applications which indicated that the offsite power source remained stable and (2) on design provisions for the disconnection of the offsite power source on its loss of voltage. However, abnormal occurrences at several operating plants indicated that a sustained undervoltage condition of the offsite power can occur and not be detected by the existing loss of voltage design provisions. Failure to disconnect from the undervoltage condition could cause redundant safety-related equipment to be exposed to voltage levels below that for which they are designed. The sustained undervoltage condition could thus result in failure of redundant safety-related equipment.

To resolve this issue, the NRC evaluated the power systems of operating plants to determine the susceptibility

of safety-related electrical equipment to: (1) sustained undervoltage condition on the offsite power source, (2) interaction of the offsite and onsite power sources, and (3) adequacy of the existing testing requirements. From this evaluation, the NRC developed an additional requirement for a second level of loss of voltage protection. This second level of protection assures disconnection from offsite power when there is a sustained undervoltage condition on the offsite power source. This additional requirement for a second level of protection was imposed on all operating and future plant designs. In order to assure implementation on future plant designs, Appendix A to SRP 8.3.1 was revised to incorporate this requirement as BTP PSB-1, "Adequacy of Station Electric Distribution System Voltages." In the advance SER, the staff also stated that in resolving A-35 it evaluated the susceptibility of safety-related electrical equipment to the rapid rate of decay of the offsite power source. The advance SER stated that the staff determined this was not a significant safety concern. Details of this determination may be found in an NRC memorandum for K. Kniel from M. Srinivasan dated July 31, 1981.

The review guideline for this issue is that the design of the undervoltage protection schemes for the Class 1E buses of the onsite power system conform to the requirements of BTP PSB-1. Specifically, a second level of voltage protection should be provided for Class 1E equipment, in addition to the existing protection based on detecting the complete loss of offsite power to the Class 1E buses. The second level should have two separate time delays: one before alerting the control room operator and the other automatically separating the Class 1E buses from the offsite power source. The time delays should be long enough to ensure protection from sustained low voltage while avoiding disconnection from the offsite source because of short-term transients such as motor starting. The undervoltage protection scheme should have the capability to be tested and calibrated during power operation. Voltage levels at the safety-related buses should be optimized for the maximum and minimum load conditions that are expected throughout the anticipated range of offsite power source voltage variations. TS are to include limiting conditions of operation, surveillance requirements, and protection equipment set points.

The ABWR design provides two levels of protection for independence of offsite and onsite systems during loss of or degraded voltage conditions. During loss of voltage condition, that is, when the bus voltage decays to less than 70 percent of its normal rated value, a bus transfer to the diesel generator is initiated by the first level of protection. During degraded voltage conditions, that is, when the bus voltage decays to between 70 and 90 percent of its normal rated value for a sustained period of time, the bus is

tripped by the second level of protection. With the bus tripped, bus transfer to the diesel generator is initiated by the first level of protection. Equipment will be qualified for voltages below 90 percent for the period of time the equipment will be subjected to these voltage conditions.

The staff reviewed the ABWR design for a second level of protection in accordance with the guidelines of BTP PSB-1, and concluded in Section 8.2.3.2 of this report that the design meets the requirements of GDC 17 defined above. Therefore, the staff concludes that GE adequately addressed this issue for the ABWR design.

20.1.13 Issue A-36: Control of Heavy Loads Near Spent Fuel (former USI)

Overhead cranes are used at all nuclear plants to lift heavy objects in the vicinity of spent fuel. If a heavy object such as a spent fuel shipping cask or shielding block were to fall onto spent fuel in the storage pool or reactor core during refueling and damage the fuel, radioactivity could be released to the environment. Such an occurrence also has the potential of overexposing plant personnel to radiation. If the dropped object were large and the damaged fuel contained a considerable amount of undecayed fission products, radiation releases could exceed the guidelines of 10 CFR Part 100. With the advent of increased and longer-term storage of spent fuel, the NRC determined that there is a need for a systematic review of requirements, facility designs, and TS regarding the movement of heavy loads to assess safety margins and improve them where necessary.

The review criteria for this issue are stated in NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants." They provide for the safe path of the load, training of the operators, inspection and tests of the equipment involved, safety interlocks, and limit switches, etc. The review of the proposed resolution of this issue consists of determining incorporation of the guidelines provided in NUREG-0612 in the ABWR design.

SSAR Section 19B.2.12 states that a number of measures are required to preclude an accident involving heavy loads, in general, and in the vicinity of the storage pool, in particular. These measures include the following:

- The COL applicant will perform a study on all planned heavy-load-handling moves to evaluate and minimize safety risks. The study will establish the heavy-loads-handling paths and routing plans. The COL applicant will also be required to provide the NRC a confirmatory structural evaluation of the spent fuel racks.

- The major heavy-load-handling equipment components (such as cranes and hoists) will be provided with an operating instruction and maintenance manual, in conformance with the guidelines of NUREG-0612, for reference and utilization by operation and maintenance personnel.
- Crane inspections and testing will comply with the requirements of American National Standards Institute (ANSI) B30.2, "Overhead Gantry Cranes," and NUREG-0612, Section 5.1.1(6).
- The equipment-handling components used over the spent fuel pool are designed to meet the single-failure-proof criteria, according to the guidelines of NUREG-0554, "Single Failure Proof Cranes for Nuclear Power Plants." Safety interlocks and limit switches are provided to prevent transporting heavy loads, other than spent fuel by the refueling platform crane, over any spent fuel that is stored in the spent fuel storage pool.
- The reactor vessel head lifting strongback and the dryer/separator lifting strongback are designed in accordance with the guidelines of NUREG-0612 and ANSI N14.6, "Standard for Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 Kg) or More for Nuclear Materials."
- The design bases of the system will conform to the requirements of GDC 2, as it relates to the ability of structures, systems, and mechanisms to withstand the effects of earthquakes; GDC 4 as it relates to protection of safety-related equipment from the effects of internally generated missiles (i.e., dropped loads); and GDC 61 as it relates to the safe handling and storage of fuel.

The staff verified that GE established a COL action item in Table 1.9-1 to provide design details for the load handling equipment. This approach is acceptable to the staff. Based on the above information, the staff concludes that GE adequately addressed this issue for the ABWR design.

20.1.14 Issue A-39: Determination of Safety Relief Valve Pool Dynamic Loads and Temperature Limits (former USI)

Operation of BWR primary system pressure relief valves can result in hydrodynamic loads on the suppression pool retaining structures or those structures located within the pool. These loads result from initial vent clearing of relief valve piping and steam quenching because of high local pool temperatures. Overall, the definition methodology

Generic Issues

used for the ABWR containment is similar to that used for prior BWR containment designs. In spite of the unique features in the ABWR, such as pressurization of the wetwell gas space, the presence of a lower drywell, the smaller number of horizontal vents (30 in the ABWR versus 120 in the Mark III containment), and extension of horizontal vents into the pool, the hydrodynamic loads addressed in this issue are similar to those in other BWR designs.

The review criteria pertinent to this issue are contained in SRP Section 6.2.1.1.C, Appendix B, which lists the generic loads acceptable to the NRC, including load identification, a summary of load specification, load specification clarifying criteria, and reference to the NRC NUREG section that describes the NRC-specific load evaluation. The staff considers that this issue summarizes and incorporates the pertinent results of the studies described in Issues A-7 and B-10, Sections 20.1.1 and 20.1.21, respectively of this report.

SSAR Appendix 3B describes containment hydrodynamic loads, such as those resulting from the SRVs, quencher discharge loads, pressure and temperature transients, and submerged structure loads. The ABWR containment design has some unique features that differ from previously approved designs, such as the Mark III containment. These unique features include pressurization of the wetwell airspace, the presence of a lower drywell, the smaller number of horizontal vents into the pool, vent submergence, and suppression pool width. GE states in the SSAR that SRV discharge is completely condensed in the pool and steam condensation loads are low compared to those of other submerged structures loads because of SRV line air clearing and LOCAs. Consequently, dynamic loads on submerged structures during quencher steam condensation will not be defined and considered for containment evaluation. This is appropriate for the dynamic loads associated with the SRV.

Based on this information and the staff's evaluation of the ABWR containment analysis in Section 6.2.1.6 of this report, the staff concludes that GE adequately addressed this issue for the ABWR design.

20.1.15 Issue A-40: Seismic Design Criteria - Short-Term Program (former USI)

Seismic design of nuclear plants is reviewed in accordance with the provisions of SRP Sections 2.5.2, 3.7.1, 3.7.2, and 3.7.3. Over the years, there has been an evolution of seismic design requirements and technology. The objective of this issue was to investigate selected areas of the seismic design sequence to determine their conservatism for all types of sites, to investigate alternative approaches where

desirable, to quantify the overall conservatism of the design sequence, and to modify the NRC criteria in the SRP where justified. Studies were conducted and the results were documented in NUREG/CR-1161, "Recommended Revisions to Nuclear Regulatory Commission Seismic Design Criteria," with specific recommendations for changes in seismic design requirements.

SRP sections were then revised with the following principal areas of change: Section 2.5.2 was updated to reflect the current NRC staff review practice; Section 3.7.1, to reflect design time history criteria; Section 3.7.2, to reflect development of floor response criteria, damping values, soil-structure interaction uncertainties, and combination of modal responses; and Section 3.7.3, to reflect seismic analysis of above-ground tanks and Category 1 buried piping.

The review criterion for the resolution of this issue is conformance with the seismic design acceptance criteria of Revision 2 to SRP Sections 2.5.2, 3.7.1, 3.7.2, and 3.7.3. Specifically, these SRP sections cover review of the site characteristics and earthquake potential, the parameters to be used in seismic design, methods to be used in seismic analysis of the overall plant, and methods to be used in seismic analysis of individual systems and components.

SSAR Section 19B.2.14 states that the design ground motions, site parameters, and system and subsystem analyses criteria and methods described in SSAR Sections 2.3.2.22, 3.7.1, 3.7.2, and 3.7.3 meet the intent of the corresponding SRP sections, except that the operating-basis earthquake (OBE) requirement is not a requirement for the ABWR. Elimination of the OBE from the ABWR design is consistent with the Commission-approved staff position on the policy issue regarding elimination of the OBE addressed in SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light Water (ALWR) Designs," April 2, 1993.

Based on this information, the staff concludes that GE adequately addressed this issue for the ABWR design.

20.1.16 Issue A-42: Pipe Cracks in BWR (former USI)

Pipe cracking has occurred in the heat-affected zones of welds in primary system piping in BWRs since mid-1980. These cracks have occurred mainly in Type 304 unstabilized austenitic stainless steel, which is the pipe material used in most operating BWRs. The major problem is recognized to be intergranular stress corrosion cracking (IGSCC) of austenitic stainless steel components. These components have been made susceptible to this

failure by being exposed to a sensitizing temperature range 427-816 °C (800-1500 °F) during welding or post-weld heat treatment.

The review criteria for this issue are contained in NUREG-0313, Revision 2, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping."

The ABWR design complies with RG 1.44, "Control of the Use of Sensitized Stainless Steel," and with the guidelines of NUREG-0313, Revision 2. These documents specify that low-carbon austenitic stainless steels are used in the construction of BWR piping and that low carbon (with a minimum of 8 percent ferrite) weld metal as deposited, is utilized in the fabrication of BWR piping. This will ensure that sensitization of ABWR components will be avoided. Based on this information, the staff concludes that GE adequately addressed this issue for the ABWR design.

20.1.17 Issue A-44: Station Blackout (former USI)

The complete loss of ac electrical power to the essential and nonessential switchgear buses in a nuclear power plant is referred to as a "station blackout" (SBO). Because many safety systems required for reactor core decay heat removal are dependent on ac power, the consequences of an SBO could lead to a severe core damage accident. This issue involves the likelihood and duration of the loss of all ac power and the potential for severe core damage after a loss of all ac power.

The issue arose because of experience with the reliability of ac power supplies. Numerous reports of standby diesel generators failing to start and run had been received and a number of operating plants had experienced a total loss of offsite electrical power. In almost every one of these latter events, the onsite ac power supplies were available to supply power to the safety equipment. However, in some instances, one of the redundant onsite ac power supplies had not been available and, in a few cases, ac power was completely lost (although during these latter events, ac power was restored in a short time and no serious consequences resulted).

The results of WASH-1400/NUREG-75/014, "Reactor Safety Study, An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," showed that for one of the evaluated plants, an SBO could be an important contributor to the total risk from nuclear power plant accidents. Although the total risk was found to be small, the relative importance of an SBO was established.

To resolve this issue, the NRC designated SBO as an unresolved safety issue and implemented a task action plan to determine the need for additional safety requirements. The results, described in NUREG-1109, "Regulatory/Backfit Analysis for the Resolution of Unresolved Safety Issue A-44, Station Blackout," indicated that actions could be taken to reduce the risk from SBO events. The NRC amended its regulations in 10 CFR Part 50 to include the SBO rule (10 CFR 50.63) and issued an associated regulatory guide (RG 1.155, "Station Blackout") that provides guidance on an acceptable means to comply with the SBO rule.

Paragraph (a) of 10 CFR 50.63 requires that each light-water-cooled nuclear power plant be able to withstand and recover from an SBO of a specific duration. The specified SBO duration must be based on (1) the redundancy of the onsite standby ac power sources, (2) the reliability of the onsite standby ac power sources, (3) the expected frequency of loss of offsite power, and (4) the probable time needed to restore offsite power. During the specified SBO duration, the reactor core and associated coolant, control, and protection systems, including station batteries and any other necessary support systems, must provide sufficient capacity and capability to ensure that the core is cooled and appropriate containment integrity is maintained.

Paragraph (c)(2) of 10 CFR 50.63 allows an alternate ac (AAC) source to be used to meet the above defined requirements of 10 CFR 50.63(a) provided that:

- Either the AAC source can be demonstrated by test to be available within 10 minutes to supply power to switchgear buses that are capable of supplying power to required shutdown equipment, or the reactor core and associated coolant, control, and protection systems, including station batteries and any other necessary support systems, have sufficient capacity and capability to ensure that the core is cooled and appropriate containment integrity is maintained from the onset of SBO until the AAC source and required shutdown equipment are started and lined up to operate,
- The time required for startup and alignment of the AAC source and shutdown equipment is demonstrated by test, and
- The AAC source, as a minimum, has the capacity and capability to ensure the plant can be brought to and maintained in safe shutdown.

To meet the SBO rule, GE has provided a combustion turbine generator (CTG) as an AAC source. The CTG has the capability of being aligned with any one of the three

Generic Issues

Class 1E divisions within 10 minutes. The CTG has sufficient capacity and capability to supply the loads that can be connected to one Class 1E division. Each of the three Class 1E divisions has sufficient capacity and capability to ensure the plant can be brought to and maintained in safe shutdown. The capability of aligning the CTG with a Class 1E bus within 10 minutes can be demonstrated by test. In addition, the time required for startup and alignment of the CTG and shutdown equipment can be demonstrated by test.

The staff reviewed the ABWR design for its ability to withstand and recover from an SBO in accordance with the guidelines of RG 1.155, and concluded in Section 8.3.9 of this report that the design meets the above defined requirements of 10 CFR 50.63(c)(2). Therefore, the staff concludes that GE adequately addressed this issue for the ABWR design.

20.1.18 Issue A-47: Safety Implications of Control Systems (former USI)

Concerns have been raised regarding the potential for accidents or transients being made more severe as a result of control system failures, including control and instrumentation power supply faults. During the licensing review process, the staff performs an audit review of the non-safety-grade control systems to ensure that an adequate degree of separation and independence is provided between these non-safety-grade systems and the safety systems. On this basis, it is generally believed that control system failures are not likely to result in safety function losses that could lead to serious events or result in conditions that the safety systems are not able to mitigate. However, in-depth studies for all non-safety-grade systems have not been performed.

Generic Letter 89-19, "Request for Action Related to Resolution of Unresolved Safety Issue A-47, 'Safety Implication of Control Systems in LWR Nuclear Power Plants'," recommended that all GE BWR plant designs provide (1) automatic reactor vessel overfill protection to mitigate main feedwater (MFW) overfeed events and (2) plant procedures and TS to periodically verify the operability of overfill protection during power operation.

The ABWR reactor vessel overfill protection is described in SSAR Section 7.7.1.4(9). The level control system provides interlocks and control functions to other systems. When the reactor water level reaches the Level 8 trip set point, the feedwater control system (FWCS) simultaneously (1) annunciates a control room alarm, (2) sends a trip signal to the turbine control system to trip the turbine generator, and (3) sends trip signals to the condensate, feedwater, and condensate air extraction

(CF&CAE) system to trip all feed pumps and to close the MFW discharge valves. This interlock is enacted to protect the turbine from damage from high moisture content in the steam caused by excessive carry over while preventing the reactor water level from rising any higher.

In the event that the feedwater pump discharge valves fail to close following the Level 8 trip signal, the FWCS automatically issues another signal to the CF&CAE system to trip all condensate pumps in order to avoid over-pressurization of the vessel.

Based on the information above and in SSAR Section 19B.2.17, which states that the COL applicant will develop plant procedures including reactor vessel overfill considerations as shown in SSAR Figure 7.7-8, "Feedwater Control System IED," the staff finds that the proposed Issue A-47 resolutions are acceptable. Therefore, the staff concludes that GE adequately addressed this issue for the ABWR design.

20.1.19 Issue A-48: Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment

This issue concerns the control of large quantities of hydrogen in reactors with small volume containments. As a result of the accident at TMI-2, the Commission issued requirements on hydrogen control in 10 CFR 50.34(f) and 50.44. 10 CFR 50.34(f) requires a hydrogen control system. This system must be based on a 100-percent, fuel clad metal-water reaction and a hydrogen concentration limit of 10 percent on uniformly distributed hydrogen in the containment or on a post-accident atmosphere that will not support hydrogen combustion.

Provision of a noncombustible containment atmosphere (inerting) is an acceptable approach to addressing this issue and is mandated for those reactors with a Mark I or II type of containment. These plants must also have hydrogen recombiners, either internal or external. Reactors with a Mark III type of containment are required to provide a hydrogen control system.

During the TMI-2 accident, it became apparent that metal-water reactions generated hydrogen in excess of the amounts specified in 10 CFR 50.44. In June 1990, the Commission approved the staff's recommendations in SECY-90-016 as the requirements for evolutionary LWRs. These requirements are set forth in 10 CFR 50.34(f)(2)(ix).

The plant-specific design must also comply with 10 CFR 50.34(f) for combustible gas control. 10 CFR 50.34(f)(2)(ix) requires that a hydrogen control system be

based on a 100-percent, fuel clad metal-water reaction and a hydrogen concentration limit of 10 percent on uniformly distributed hydrogen in the containment or on a post-accident atmosphere that will not support hydrogen combustion. 10 CFR 50.34(f)(3)(v) requires that containment integrity be maintained below Service Level C limits for steel containments and the factored load category for concrete containments during an accident that releases hydrogen generated from a 100-percent, fuel clad metal-water reaction.

This issue was resolved by the requirements contained in 10 CFR 50.34(f) and 10 CFR 50.44. SSAR Section 19B.2.18 states that there are no design-basis events for the ABWR that result in core uncover or core heatup sufficient to cause significant metal-water reaction. It further states that this is equivalent to the reaction of the active clad to a depth of 0.00058 cm (0.00023 in.) or 0.72 percent of the active clad. SSAR Section 6.2.5.3 states that the atmospheric control system (ACS) is designed to maintain the containment in an inert condition, except for nitrogen make-up needed to maintain a positive containment pressure and prevent air leakage from the secondary into the primary containment.

GE analyzed consequences of hydrogen release and concluded that for 100-percent, fuel clad metal-water reaction, the resulting peak containment pressure would be about 517 kPa (75 psig). The ABWR has a concrete containment with a steel upper drywell head. This head, based on GE structural analysis, has been shown to be the most limiting structural component of the containment, with a Service Level C limit of 669 kPa (97 psig). Therefore, the containment pressurization resulting from a 100-percent, fuel clad metal-water reaction, coupled with a large-break LOCA, is below the Service Level C limit.

On the basis of this information and the staff's evaluation of ABWR compliance with 10 CFR 50.34(f)(2)(ix), as discussed in Sections 20.5.21 and 19.2.3.3.1 of this report, the staff concludes that GE adequately addressed this issue for the ABWR design.

20.1.20 Issue B-5: Ductility of Two-Way Slabs and Shells and Buckling Behavior of Steel Containments

This issue addresses two concerns relating to containment design: (1) that sufficient information is not available to predict the behavior of two-way reinforced concrete slabs, and (2) that the structural design of a steel containment vessel subjected to unsymmetrical dynamic loadings may be governed by the instability of the shell. The safety significance of the first concern is that in the event of the collapse of a floor that may be caused by an earthquake or

a LOCA, there would be a possibility that other portions of the RCS or safety-related systems could be damaged. The damage could lead to an accident sequence resulting in the release of radioactivity to the environment.

The other concern, identified in NUREG-0471, "Generic Task Problem Descriptions (Categories B, C, and D)," is over the lack of a uniform, well-defined approach to design evaluation of steel containments. The structural design of a steel containment vessel subjected to unsymmetrical dynamic loadings may be governed by the instability of the shell. For these types of loads, the current criteria and the current analytical techniques may not be as comprehensive as they should be.

The review criterion for the first concern is that the design code, American Concrete Institute code ACI 349, "Code Requirements for Nuclear Safety Related Structures," contains sufficient information pertaining to the design of two-way slabs subjected to dynamic loads and biaxial tension to enable a reasonably accurate analysis. ACI 349 should be used in conjunction with the pertinent regulatory documents such as SRP Section 3.5.3, Appendix A, and RG 1.142, "Safety-Related Concrete Structures for Nuclear Power Plants (Other than Reactor Vessels and Containments)."

The review criterion for the second concern is that all applied loads must be adequately addressed by the steel containment design. RG 1.57, "Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components," recommends a minimum factor of safety of 2 against buckling for the worst loading condition, provided that a detailed, rigorous analysis that considers inelastic behavior is performed. Also, the allowable stress values for buckling are contained in the ASME Code, Section III, Subsection NE-3222.

SSAR Sections 3.8.3, 3.8.4, and 19B.2.61 state that the design of the ABWR safety-related structures (other than the containment vessel), including consideration of the ductility requirements for the two-way slabs, is based on the ACI 349-80 Code. The approach used by GE for the design of two-way reinforced concrete slabs meets the guidelines of SRP Sections 3.8.3 and 3.8.4 and is, therefore, acceptable.

The applicant indicated that the ABWR containment is a concrete structure and the steel component not backed by concrete, that is, the ABWR reactor closure head, is designed in accordance with the ASME Code, Section III, Subsection NE. This approach meets the guidelines of SRP Section 3.8.2 for shell buckling and, therefore, is acceptable.

Generic Issues

Based on the above discussions, the staff concludes that GE adequately addressed this issue for the ABWR design.

20.1.21 Issue B-10: Behavior of BWR Mark III Containments

This issue deals with the dynamic loads present in the Mark III containment following a postulated LOCA when escaping steam forces the suppression pool water out of the drywell and into the wetwell. This action results in pool swell and loads from vent clearing, jets, chugging, impact of water, impact of froth impingement, pool fallback, condensation, and containment pressure. The concern is that these loadings may damage structures and components located within the wetwell. Although many of these structures (e.g., walkways) are by themselves not related to safety, the various emergency core cooling systems (ECCSs) take suction from the wetwell and, therefore, damage in the wetwell may affect the performance of the ECCSs.

The review criteria for this issue are provided in NUREG-0978, "Mark III LOCA-Related Hydrodynamic Load Definition," Appendix C. These criteria have been developed on the basis of large-scale tests conducted between 1973 and 1979 by GE in order to define the LOCA-related hydrodynamic loads for use in the design of the standard Mark III containment.

The ABWR horizontal vent confirmatory test program was performed to obtain data that could be used to determine condensation oscillation and chugging loads for design evaluation of containment structures. The test matrix included tests at conditions that produce bounding loads and additional tests to examine the sensitivity of the loads to system parameters. The test specifically documents work performed, including general evaluation of the test data and the specification of procedures that can be used to define containment loads.

The ABWR design utilizes a horizontal vent system, which is similar to that of the Mark III containment design, but includes some ABWR-unique design features. These unique features include pressurization of the wetwell airspace, the presence of a lower drywell, a smaller number of horizontal vents (30 in the ABWR containment versus 120 in the Mark III containment), extension of horizontal vents into the pool, vent submergence, and suppression pool width.

The ABWR horizontal vent test program has been based on resolution of several other issues (A-7, A-8, and B-10), which produced test data to confirm and define condensation oscillation and chugging loads for design application. SSAR Section 3B.2.2 describes a spectrum of

postulated LOCAs that was considered in assessing the design adequacy of the ABWR containment system. The results obtained from small-scale tests conducted within the scope of the ABWR design confirmed the applicability of Issue B-10 test data.

Based on this information and the staff's evaluation of the ABWR containment analysis in Section 6.2.1.6 of this report, the staff concludes that GE adequately addressed this issue for the ABWR design.

20.1.22 Issue B-17: Criteria for Safety-Related Operator Actions

Current plant designs are such that reliance on the operator to take action in response to certain transients is necessary. Consequently, it becomes necessary to develop appropriate criteria for safety-related operator actions (SROAs). The criteria would include a determination of actions that should be automated rather than manual and development of a time criterion for SROAs.

The review criteria for this issue are contained in ANSI/American Nuclear Society (ANS) 58.8-1984, "Time Response Design Criteria for Nuclear Safety Related Operator Actions," and ANSI/ANS 52.2-1983, "Nuclear Safety Criteria for the Design of Stationary Boiling Water Reactor Plants." Plants should perform task analysis, simulator studies, and analysis and evaluation of operational data to assess ESF and safety-related control system designs for conformance to the criteria. Where nonconformance is identified, modification of the design and hardware may be required.

SSAR Appendix 18E describes the program of human-factors-related activities conducted throughout the development of the ABWR plant system designs, including the development of the main control room (MCR) and remote shutdown system (RSS) designs. Appendix 18E describes the process through which the MCR and RSS human-system interface (HSI) design implementations will be conducted and evaluated through the application of human factors engineering (HFE) practices and principles.

The COL applicant is responsible for addressing B-17 as part of the detailed design implementation. The staff verified that GE has established a general COL action item in SSAR Section 18.8 (Item 18.8.1) to conduct the detailed HFE design according to design and implementation as defined by the ABWR certified design material (CDM) Table 3.1 Inspection, Tests, Analyses, and Acceptance Criteria (ITAAC) and SSAR Appendix 18E. The staff considers the SSAR to include commitments for the COL applicant to conduct the necessary analyses of the critical operator tasks. In addition, the staff verified that GE

established another COL action item in SSAR Section 18.8 (Item 18.8.15) to evaluate the adequacy of the HSI to provide the necessary controls, displays, and alarms for the timely performance of critical tasks. This approach is acceptable to the staff as discussed in Section 18.7.2.2 of this report.

Based on this information, the staff concludes that GE adequately addressed this issue for the ABWR design.

20.1.23 Issue B-29: Effectiveness of Ultimate Heat Sinks

This issue addressed the adequacy of existing NRC mathematical models for the prediction of the ultimate heat sink (UHS) performance by comparing model performance with field data. The issue also addressed the selection of site-specific meteorological data for use as UHS design basis meteorology.

The DFSER stated that the design of the UHS is not within the scope of the ABWR design. GE provided several interface requirements to be used as guidance for the design of systems not within the ABWR design scope. The DFSER contained a lengthy discussion of these interface requirements and concluded that GE provided sufficient information to allow the COL applicant to provide a plant-specific response to Issue B-29. The plant-specific response was identified in the DFSER as COL Action Item 20.1-1.

Issue B-29 is categorized in NUREG-0933 as a licensing issue and as such, is not required to be considered in meeting the requirements of 10 CFR 52.47(1)(a)(iv). The purpose of the issue was to confirm the validity of the NRC models used to make an independent assessment of UHS design safety. NRC's studies have confirmed that the capability of NRC's models and existing guidance are adequate. The determination of the adequacy of a specific UHS design is to be made by the staff during the review process using the models and guidance. The design of the UHS for the ABWR will be provided by the COL applicant as part of the COL application in accordance with the interface requirements discussed in Section 9.2.5 of this report. The UHS design will be reviewed at the COL application stage. This approach is acceptable to the staff.

Upon further consideration, the staff determined that because Issue B-29 is a licensing issue and not required to be considered in meeting the requirements of 10 CFR 52.47(1)(a)(iv) as discussed above, no specific response to Issue B-29 is necessary from either GE or the COL applicant. On this basis, the staff concludes that DFSER

COL Action Item 20.1-1 was not warranted and need not be addressed.

20.1.24 Issue B-32: Ice Effects on Safety-Related Water Supplies

This issue addressed the potential effects associated with extreme cold weather and ice buildup on the reliability of various plant water supplies. Of particular concern are events that could affect safety-related water systems and affect the ability of the plant operations staff to safely shut down the plant and provide adequate core cooling.

The DFSER contained a discussion of interface requirements for the UHS necessary to the reliability of this water source. The evaluation concluded that GE needed to require the COL applicant to use site weather conditions to establish the severe weather design envelope for the site. This was identified in the DFSER as COL Action Item 20.1-2.

The staff's evaluation in the DFSER also identified that the interface requirements for the UHS did not explicitly include the RSW system, portions of which have exposed piping that may be vulnerable to the adverse effects of ice, and required GE to address this concern. This was identified in the DFSER as Open Item 20.1-2. The DFSER stated that subject to the acceptable resolution of DFSER Open Item 20.1-2, the staff could conclude that the ABWR design addresses the concerns of Issue B-32.

In the DFSER, the staff also required GE to establish a requirement for the COL applicant to address the capability of the plant-specific RSW system and UHS designs to address the concerns of Issue B-32. This was identified in the DFSER as COL Action Item 20.1-3.

NUREG-0933 shows that Issue B-32 was subsumed by Issue 153 (discussed in Section 20.2.32 of this report). As such, Issue B-32 is not required to be considered separately in meeting the requirements of 10 CFR 52.47(a)(1)(iv). The UHS design and the out-of-scope portions of the RSW will be provided by the COL applicant as part of the COL application in accordance with the interface requirements discussed in Sections 9.2.5 and 9.2.15 of this report. The UHS design and the out-of-scope portions of the RSW will be reviewed at the COL application stage. This approach is acceptable to the staff. In addition, GE has included a COL action item in SSAR Section 2.3.2.14 for the COL applicant to demonstrate that safety-related facilities and water supply are not affected by ice flooding or blockage.

Because Issue B-32 was subsumed by Issue 153, upon further consideration of this information, the staff

Generic Issues

determined that no specific response to Issue B-32 is necessary from either GE or the COL applicant. On this basis, the staff concludes that

- (1) DFSEER Open Item 20.1-2 was not warranted and need not be addressed. Therefore, DFSEER Open Item 20.1-2 is resolved.
- (2) DFSEER COL Action Items 20.1-2 and 20.1-3 were not warranted and need not be addressed.

See Section 20.2.32 of this report for the staff's evaluation of Issue 153.

20.1.25 Issue B-36: Develop Design, Testing, and Maintenance Criteria for Atmosphere Cleanup System Air Filtration and Adsorption Units for Engineered Safety Features Systems and Normal Ventilation Systems

This issue involves developing revisions to current guidance and staff technical positions regarding ESF and normal ventilation system air filtration and adsorption units. Any technological advances leading to better methods and/or standards for the design, testing, and maintenance for these systems in light water-cooled nuclear power plants need to be documented for NRC staff guidance and technical positions.

Guidance on controlling the release of gaseous radioactive effluents to the environment is contained in Revision 1 to RG 1.140, "Design, Testing, and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," and Revision 2 to RG 1.52, "Design, Testing and Maintenance Criteria for Postaccident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Plants." RG 1.52 provides guidance relating to the design, testing, and maintenance of post-accident ESFs, whereas RG 1.140 applies to the normal ventilation exhaust features.

GE provided the information pertinent to this issue in SSAR Sections 6.4 and 6.5. SSAR Section 6.5.1 states that the filter systems required to perform safety-related functions following a design-basis accident are the standby gas treatment system (SGTS) and the control room portion of the HVAC system.

The SGTS has the capacity to filter the gaseous effluent from the primary containment or from the secondary containment, when required, to limit the discharge of radioactivity to the environment to meet the guidelines of 10 CFR Part 100. GE described the SGTS power generation design basis, safety design basis, system design,

SGTS operation (automatic and manual), and the design evaluation. Compliance with RG 1.52 is also discussed.

SSAR Sections 9.4.1 and 6.4 and Appendices 6A, 6B, 9C, and 9D describe the HVAC system pertinent to control room habitability. The system is designed in conformance with the requirements of GDC 19 and guidelines of RG 1.52 related to the design, testing, and maintenance of post-accident ESFs and ASME N509 and N510.

The radwaste building incinerator offgas exhaust is directed to a separate monitored vent, as described in SSAR Section 9.4.6.5.3. The COL applicant is to provide conformance with RG 1.140 for the radwaste building HVAC system.

Based on this information, the staff concludes that GE adequately addressed this issue for the ABWR design.

20.1.26 Issue B-55: Improved Reliability of Target Rock Safety-Relief Valves

The majority of valves in BWR pressure-relief systems are Target Rock SRVs. A significant number of failures of these valves have occurred, which include valves that (1) failed to open properly on demand, (2) opened spuriously and then failed to reseal properly, and (3) opened properly and then failed to reseal properly. The performance of these valves is under continual surveillance and the consequences of their failures are subject to review.

SSAR Section 19B.2.22 states that Target Rock SRVs are not to be used in the ABWR design. The SRVs to be used in the ABWR do not have a pilot stage such as that present in the Target Rock pilot-operated SRVs. The ABWR will use a direct-acting SRV design. Therefore, the mechanisms that cause the pilot valve to fail to open properly have been eliminated from the ABWR design.

Based on this information, the staff concludes that GE adequately addressed this issue for the ABWR design.

20.1.27 Issue B-56: Diesel Reliability

If a loss of normally available ac power from the offsite preferred systems occurs at a nuclear power plant, redundant onsite standby ac power sources (diesel generators) provide power for necessary safety functions, which include reactor core decay heat removal, emergency core cooling, and containment heat removal. Therefore, the reliability of the diesel generators is a major factor in ensuring acceptable plant safety.

This issue was promulgated by a review of licensee event reports (LERs) which indicated that the diesel generators

at operating plants were demonstrating an average starting reliability of about 0.94 per demand. The NRC's goal for new plants, as expressed in Regulatory Guide 1.108, "Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants," is a diesel generator starting reliability of 0.99 per demand. As part of the resolution of Issue A-44 on station blackout (see Section 20.1.17 of this report), diesel reliability was considered one of the main factors affecting plant risk from station blackout. Thus, attaining and maintaining high diesel generator reliability was a necessary part of the resolution of Issue A-44. The Federal Register notice (53 FR 23217) which accompanied 10 CFR 50.63 (station blackout rule) indicated that the resolution of Issue B-56 would provide specific guidance for diesel generator reliability consistent with the resolution of Issue A-44. The specific guidance (the resolution of B-56) has been included as part of Revision 3 of RG 1.9, "Selection, Design, and Qualification of Diesel-Generator Units Used as Standby (Onsite) Electric Power Systems at Nuclear Power Plants" and as part of RG 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." RG 1.160 endorses the guidelines in NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants" (May 1993) as an acceptable method for meeting the requirements of 10 CFR 50.65 (the maintenance rule). With regard to systems and components such as diesel generators, 10 CFR 50.65 requires that:

- the performance or condition of diesel generators be monitored against established goals in a manner sufficient to provide reasonable assurance that the diesel generators are capable of fulfilling their intended functions,
- goals be established commensurate with safety and to take into account industry wide operating experience,
- appropriate corrective action be taken when the performance or condition of the diesel generators do not meet established goals,
- diesel generator performance and condition monitoring activities and associated goals and preventive maintenance activities be periodically evaluated, taking into account industry wide operating experience, and
- adjustments be made where necessary to ensure that the objective of preventing failure of diesel generators through maintenance is appropriately balanced against the objective of minimizing unavailability of diesel generators due to monitoring or preventive maintenance.

Revision 3 of RG 1.9 integrated into a single regulatory guide guidance previously addressed in other documents such as Revision 2 of RG 1.9, Revision 1 of RG 1.108, and Generic Letter 84-15.

With respect to diesel generator reliability (compliance with Issue B-56), the ABWR SSAR indicates the following:

- a. Diesel generators will be required to be capable of reaching full speed and voltage within 20 seconds after the signal to start (Section 8.3.1.1.8.2(4) of SSAR Amendment 34).
- b. For the COL license, the COL applicant will be required to have appropriate plant procedures for periodic testing of diesel generator start capability (Section 8.3.4.36 of SSAR Amendment 33). In particular,
 - Appropriate plant procedures will include the requirement that the interval between periodic start test for diesel generators will be no longer than 31 days (Section 6.5.1 of IEEE 387-1984), and
 - Diesel generator start testing may, once per 6 months, be replaced with a modified diesel generator start involving idling and gradual acceleration to synchronous speed as recommended by the manufacturer (Technical Specification guidelines in Section 16.3.8 (SR 3.8.1.2) of SSAR Amendment 34).
- c. As part of the COL license, the COL applicant will be required to demonstrate the start reliability of the diesel generators (Section 8.3.4.2 of SSAR Amendment 34). Specifically,
 - The preoperational test program will demonstrate the required reliability by means of 25 start demands without failure on each installed diesel generator unit (Section 14.2.12.1.45.3(l) of SSAR Amendment 34 and Position C.2.3.1 of RG 1.9 (Rev. 3)).
 - Periodic testing at intervals of no longer than 31 days will commence within 31 days following completion of preoperational testing for diesel generator start reliability (Position C.2.3.2.1 of RG 1.9, Rev. 3)).
 - Performance criteria and goals for diesel generator start and loading reliability will be set at 0.975 per demand as part of the COL licensee's program for

Generic Issues

assuring diesel generator reliability. The SSAR indicates:

- Diesel generator start reliability of 0.986 per demand can be achieved as shown by industry experience (Section 19B.2.23 of SSAR Amendment 34).
- A diesel generator reliability of 0.975 per demand was assumed for station blackout considerations (Table 1C-1 of SSAR Amendment 32).
- A diesel generator reliability of 0.975 per demand was used in the ABWR PRA analysis (Table 1C-1 of SSAR Amendment 32).
- Performance criteria for both diesel generator reliability and unavailability will be established to assure that the performance or condition of the diesel generators is being effectively controlled through the performance of appropriate preventive maintenance, such that the diesel generator remains capable of performing its intended function (Section B of RG 1.160). The SSAR states:
 - Performance criteria for reliability will be met by the absence of a maintenance-preventable failure, or the occurrence of a single maintenance-preventable failure, followed by appropriate root cause determination and corrective action.
 - Performance criteria for unavailability will be met by having fewer unavailable hours, on a rolling 1-year basis, than required by the established performance criteria.
- If performance criteria is not met or a second diesel generator maintenance-preventable failure occurs, diesel generator performance goals will be established and the performance or condition of the diesel generators will be monitored in a manner sufficient to provide reasonable assurance that the diesel generators are capable of fulfilling their intended functions consistent with an appropriate balance between diesel generator reliability and unavailability (Section B of RG 1.160).
- Periodic adjustments will be made where necessary to ensure that the objective of preventing failures of the diesel generator through maintenance will be appropriately balanced against the objective of minimizing unavailability of the diesel generator due to monitoring or preventive maintenance (Section B of RG 1.160).

The staff concludes that a diesel generator testing and reliability program which meets the above described commitments will assure an acceptable level of diesel reliability in accordance with the objectives of Issues B-56 and A-44 and will meet the above defined requirements of 10 CFR 50.65. Because COL applicants will be required to have a diesel generator test and reliability program for their license which meets the above described commitments, or an acceptable alternative method, the staff concludes that GE adequately addressed this issue for the ABWR design.

20.1.28 Issue B-61: Allowable ECCS Equipment Outage Periods

This issue concerns establishing surveillance test intervals and allowable equipment outage periods, using analytically based criteria and methods for TS. The present TS allowable equipment outage intervals and test intervals were determined primarily on the basis of engineering judgement. Studies performed by the NRC on operating reactors indicated that from 30 to 80 percent of the ECCS system unavailability was a result of testing, maintenance, and allowed outage periods. Therefore, by optimizing the allowed outage period and the test and maintenance interval, the equipment unavailability and public risk can be reduced.

The review criteria for this issue consist of the techniques and methods available and the modeling from the Interim Reliability Evaluation Program (IREP) and the National Reliability Evaluation Program (NREP), the optimum equipment test intervals and allowable equipment downtimes. Also, since the ABWR evolved primarily from the BWR/6 design, most of the ABWR TS that control the ECCS outage periods, were modeled after NUREG-1433, "Standard Technical Specifications General Electric Plants, BWR/6." Furthermore, the criteria for resolution of this issue incorporated the accumulated experience from currently operating light water reactors.

SSAR Section 19B.2.24 states that the ABWR design incorporates many design enhancements to improve the operation and safety of the plant, and the most significant advances are in the area of ESFs. Based on the review of the information provided in the SSAR, the staff agreed with GE that the ABWR design includes redundancy for the ESF systems beyond that for currently operating BWR plants. This added redundancy allows for extending the associated completion times (CTs) beyond those specified for ESF systems on currently operating BWR plants, thereby facilitating maintenance on certain subsystems. This relaxation ranges from 8 hours to 14 days, and is based on probabilistic risk analysis (PRA), engineering

evaluation, operating experience, and judgement for various components and combinations of components.

The PRA performed by GE used a system fault tree approach to quantify system accident sequences that result in severe core damage. Data related to the ESF used in the quantification included:

- Component failure rates
- Component repair times and maintenance frequencies
- Component inspection and test times and frequencies
- Allowable equipment completion times

The data used were in accordance with the guidance in NUREG/CR-2815, "Probabilistic Safety Analysis Procedures Guide," and basic failure rate data were obtained from the EPRI ALWR Requirements Document, supplemented by other nuclear sources. The staff reviewed the ABWR TS and concluded that the CTs contained therein are considered appropriate by the staff. The staff concludes that GE adequately addressed this issue for the ABWR design.

20.1.29 Issue B-63: Isolation of Low-Pressure Systems Connected to the Reactor Coolant Pressure Boundary

These generic issues, combined with Issue 96, "RHR Suction Valve Testing," (subsumed by Issue 105, "Interfacing Systems LOCA at LWRs") concern the common area of the ABWR, namely, the systems connected to the RCS pressure boundary that are considerably below the RCS operating pressure. The NRC has required that valves forming the interface between these high and low systems and the associated piping have sufficient redundancy to assure that the low-pressure systems are not subjected to pressure that exceeds their design limits.

Issue B-63 was resolved by the staff and implemented by MPA B-45, "Event V, Primary Coolant System Pressure Isolation Valves," which required leak testing of these check valves in accordance with plant-specific technical specifications. For the ABWR, SSAR Section 19B.2.25 states that test requirements for these valves are in SSAR Section 3.9.6, "Testing of Pumps and Valves." SSAR Table 3.9-9 provides a list of all the ABWR reactor coolant system pressure isolation valves. Section 3.9.6 states that all of the valves in this table will be periodically leak tested in accordance with the surveillance requirements in SSAR Chapter 16, "Technical Specifications." As discussed in Section 3.9.6.2.4 of this report, the staff reviewed the applicable information in SSAR Section 3.9.6 and concluded that it is acceptable.

On the basis the above discussion, the staff concludes that GE adequately addressed these issues for the ABWR.

20.1.30 Issue B-66: Control Room Infiltration Measurements

This issue addresses the concerns that the control room may not be in a safe, habitable condition under accident conditions and may not provide adequate protection for the plant operators against the effects of accidental releases of airborne radioactivity and toxic gases. The rate of air infiltration into the control room is a significant factor in maintaining habitability, and the NRC measured air exchange rates in selected operating reactor plant control rooms to improve the data base for evaluating its effects. No new design requirements were established by the NRC as a result of this and other work related to control room habitability in an accident. However, more specific review procedures were incorporated in SRP Sections 6.4, 6.5.1, and 9.4.1, including the habitability review provisions of TMI Action Plan Item III.D.3.4 (see Section 20.4.87 of this report) regarding analyses of toxic gas concentrations and operator exposures from airborne radioactive material and direct radiation, to ensure more effective implementation of existing requirements.

The review criterion for the resolution of this issue is that the control room ventilation and air conditioning systems be designed to maintain the room's environment within acceptable limits for the operation, testing, and maintenance of the unit controls and the uninterrupted safe occupancy during normal and accident conditions.

SSAR Sections 9.4.1, 6.4, and 19B.2.26 state that the systems incorporated in the design of the control room ventilation and air conditioning will meet the intent of the guidance given in SRP Sections 6.4, 6.5.1, and 9.4.1. More specifically, these systems are designed to meet the intent of the guidance given in SRP Sections 6.4, 6.5.1, 9.4.1, and 15.6.5.5 (all Rev. 2). Under normal operation, the control room HVAC provides HVAC functions and pressurization inside the main control area envelope (MCAE) using a combination of filtered outdoor air and recirculated indoor air.

The emergency recirculation system consists of an electric heating coil, a prefilter, pre-high-efficiency particulate (HEPA) filter, charcoal absorber, post-HEPA, and two 100-percent capacity circulating fans. Independent and separate discharge to and return from the MCAE to each filtration unit is provided. All control room HVAC equipment, including the ductwork (which is termed ESF),

Generic Issues

and surrounding structures are seismic Category I design and operable during loss of offsite power supply.

ABWR design features regarding control room habitability and the HVAC are described in the SSAR Sections 9.4.1 and 6.4. The filtration units conform to the guidance of RG 1.52, and the requirements of ANSI ASME N509 and ASME N510, "Testing of Nuclear Air-Cleaning Systems." SSAR Appendices 9C and 9D provide detailed information regarding ABWR conformance with RG 1.52 and SRP Table 6.5.1-1.

In Amendment 32, GE revised SSAR Sections 9.4.1.1.4 and 9.4.1.1.5 to state that the galvanized steel (American Society for Testing and Materials (ASTM) A526 or A527) is used for outdoor air intake and exhaust ducts and all other ducts (CRHS HVAC system) are welded, black steel ASTM A570, Grade A or Grade D. In Amendment 34, GE revised SSAR Section 9.4.1.1.5 to state that unfiltered in-leakage testing will be periodically performed on all ductworks and housing outside the MCAE in accordance with ASME N510. Based on the information provided in SSAR Sections 19B.2.26, 9.4.1, and 6.4 and Appendices 9C and 9D that the appropriate plant-specific procedures will be developed to address MCR infiltration measurements, to preclude any unfiltered in-leakage not credited in dose analysis, in accordance with the above requirements.

On the basis of this information, the staff concludes that GE adequately addressed this issue for the ABWR design.

20.1.31 Issue C-1: Assurance of Continuous Long-Term Capability of Hermetic Seals on Instrumentation and Electrical Equipment

This issue concerns the long-term capability of hermetically sealed instruments and equipment that must function in post-accident conditions. More specifically, certain classes of instrumentation that are equipped with seals are sensitive to steam and vapor. If the seals become defective as a result of personnel error in the maintenance of such equipment, such errors could lead to the loss of a seal and of equipment functionality. The objective of this issue is to establish confidence that sensitive equipment has an effective seal for the lifetime of the plant.

The review criterion for this issue is compliance with the review criteria of SRP Section 3.11 for environmental qualification of electrical equipment.

SSAR Section 19B.2.17 refers to SSAR Section 3.11, "Environmental Qualification of Safety-Related Mechanical and Electrical Equipment," which defines the environmental conditions with respect to limiting design

conditions for all safety-related mechanical and electrical equipment. Safety-related equipment located in a harsh environment must perform its proper safety function during normal, abnormal, test, DBA, and post-accident environments, as applicable. A list of all safety-related electrical and mechanical equipment required for safe shutdown that is located in a harsh environment area will be included in the Environmental Qualification Document as stated in SSAR Section 3.11.6.1.

Environmental conditions for the zones where safety-related equipment is located are calculated for normal, abnormal, test, accident, and post-accident conditions and are documented in SSAR Appendix 3I, "Equipment Qualification Environmental Design Criteria." Environmental conditions are tabulated by zones contained in the referenced building arrangements.

Safety-related electrical equipment that is located in a harsh environment is qualified by test or other methods, as described in IEEE 323 and permitted by 10 CFR 50.49(f). The qualification methodology is described in detail in the NRC-approved report, NEDE-24326-1-P. This report also addresses compliance with the applicable portions of 10 CFR Part 50, Appendix A and the quality assurance criteria of 10 CFR Part 50, Appendix B. Additionally, the report describes conformance to NUREG-0588 and the RGs and IEEE standards referenced in SRP Section 3.11.

Based on the above discussion, since safety-related electrical equipment for the ABWR will be qualified in accordance with applicable guidance, including NUREG-0588, the staff concludes that GE adequately addressed this issue for the ABWR design.

Details on the staff's evaluation of environmental qualification of safety-related electrical equipment are provided in Section 3.11 of this report.

20.1.32 Issue C-8: Main Steam Line Leakage Control Systems

Dose calculations in 1975 by NRC's former Accident Analysis Branch indicated that operation of the main steam isolation valve leakage control system (MSIVLCS) required for some BWRs could result in higher offsite accident doses than if the system were not used and the integrity of the steam lines and condenser was maintained. This issue was initiated to investigate whether the MSIVLCS recommended in RG 1.96, "Design of Main Steam Isolation Valve Leakage Control Systems for Boiling Water Reactor Nuclear Power Plants," was desirable. This issue was resolved without the establishment of new requirements. The main steam system of BWR designs is expected to meet GDC 54. RG 1.96 describes a means

acceptable to the staff for implementing GDC 54 with regard to the design of an MSIVLCS that ensures that total radiological effects do not exceed the guidelines of 10 CFR Part 100 in the event of a postulated design-basis LOCA.

Since this issue was resolved without the establishment of new requirements, it does not need to be addressed in the ABWR design certification review. A discussion of the ABWR approach to the MSIVLCS issue is provided here for information purposes only.

The ABWR design does not incorporate a MSIVLCS, but utilizes an alternative approach as permitted by RG 1.96. RG 1.96 indicates that an LCS is not required if the main steam line leakage path can be relied on to remain intact and capable of providing significant dose reduction factors for postulated accident conditions. A description and the staff's evaluation of the ABWR's method of containing and holding up MSIV leakages following a design-basis LOCA are addressed in Section 10.3.1 of this report. Evaluation of the seismic and radiological analyses of the main steam and condenser systems' capability to perform this post-LOCA function is provided in Sections 3.2.1 and 15.4.4.2, respectively, of this report.

20.1.33 Issue C-10: Effective Operation of Containment Sprays in a LOCA

This issue deals with the effectiveness of various containment sprays to remove airborne radioactive materials that could be present within the containment following a LOCA. This concern includes the possible damage to equipment located inside the containment because of inadvertent actuation of the sprays.

The review criteria for this issue are that the containment spray system will be designed to meet the requirements of GDC 41, 42, and 43 of 10 CFR Part 50, Appendix A, related to fission product removal, periodic inspection, and functional testing, respectively, by conforming to the guidance of SRP Section 6.5.2, Revision 2.

SSAR Section 19B.2.28 states that the RHR system provides two independent containment spray cooling systems (on loops B and C), each having a common header in the wetwell and a common spray in the drywell, and sufficient capacity for containment depressurization by removing heat and condensing steam in both the drywell and wetwell air volumes following a LOCA. All components of the RHR containment spray system can be inspected and tested during normal plant operation or during refueling and maintenance outages. The ABWR design does not take credit for any fission product removal provided by the drywell and wetwell spray portion of the RHR system. The removal of fission products is

controlled by the SGTS, which has the redundancy and capability to filter the gaseous effluent from the primary and secondary containment. However, the drywell sprays off the RHR system also function to provide removal of fission products during a LOCA, as well as in the event of failure of the drywell head. The drywell spray is initiated by operator action post-LOCA in the presence of high drywell pressure and is terminated by operator action. It is also terminated automatically as the RHR injection valve starts to open.

The water in the 304L stainless-steel-lined suppression pool is maintained at high purity by the suppression pool cleanup (SPCU) system. The pH range is maintained between 5.3 and 8.6 to minimize any corrosive attack on the pool liner over the life of the plant. The protective epoxy coatings applied on the carbon steel containment liner, internal steel structures and equipment inside the drywell and wetwell meet the guidance of RG 1.54, "Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants," and are qualified using the tests in accordance with ANSI N101.4, "Quality Assurance for Protective Coatings Applied to Nuclear Facilities."

On the basis of this information, the staff concludes that GE adequately addressed this issue for the ABWR design.

20.1.34 Issue C-17: Interim Acceptance Criteria for Solidification Agents for Radioactive Solid Wastes

This issue concerns lack of criteria for acceptability of solidification agents for radioactive solid wastes.

The review criteria for this issue are contained in 10 CFR Part 61, which was published in the Federal Register on December 27, 1982, and includes Section 61.56, which addresses waste characteristics. Also, BTP ETSB-11-3, "Design Guidance for Solid Radioactive Waste Management Systems Installed in Light-Water-Cooled Nuclear Power Plants," was developed under TMI Action Plan Item IV.C.1 before the issuance of 10 CFR 61.56. Evaluation of this issue is based on these documents.

SSAR Section 19B.2.29 states that the ABWR design will comply with the requirements of the 10 CFR 61.56 regarding waste characteristics. The establishment and implementation of a process control program (PCP) for solidifying the evaporator concentrates (the applicable plant waste will be solidified) using an approved solidification agent is dependent on the as-procured equipment for the ABWR standard design. Therefore, the staff will review the PCP on a plant-specific basis. The staff verified that GE established a COL action item in SSAR

Generic Issues

Section 11.4.3.1 to provide a PCP for solidifying the evaporator concentrates using an approved solidification agent and to demonstrate that the wet solidification process will result in a product that complies with 10 CFR 61.56 regarding waste characteristics. The staff finds this approach acceptable, and therefore, concludes that GE adequately addressed this issue for the ABWR design.

20.1.35 Other Issues

20.1.35.1 Issue A-43: Containment Emergency Sump Performance

This issue concerns the availability of adequate recirculation cooling water following a LOCA when long-term recirculation of cooling water (from the PWR containment sump or the BWR RHR system suction intake) must be initiated to prevent core melt. This water must be sufficiently free of LOCA-generated debris and potential air ingestion so that pump performance is not impaired, thereby seriously degrading long-term recirculation flow capability. Issue A-43 was resolved with the issuance of RG 1.82 (Rev. 1), "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident."

NUREG-0933, Appendix B indicates that this issue is not applicable to BWRs, therefore, GE did not specifically address it in SSAR Appendix 19B. However, the staff evaluated GE's compliance with RG 1.82 (Rev. 1) and supplemental staff positions and found it to be acceptable. Refer to Sections 6.2.1.9 and 6.2.2 of this report for the staff's evaluation.

20.2 New Generic Issues

This section addresses staff evaluation of GSIs that are categorized as "new generic issues" in NUREG-0933. All the following issues, with the exception of Issue 130 (which was evaluated in the DFSER) are relevant to the ABWR design. Issue 130 is discussed here for continuity only.

20.2.1 Issue 15: Radiation Effects on Reactor Vessel Supports

This issue addresses the potential problem of radiation embrittlement of reactor vessel support structures. Neutron damage of structural materials causes embrittlement that may increase the potential for propagation of flaws that might exist in the materials. The potential for brittle fracture of these materials is typically measured in terms of the material's nil ductility transition temperature (NDTT), which is the lowest temperature at which the material would not be susceptible to failure by

brittle fracture. As long as the operating environment in which the materials are used has a higher temperature than the material's NDTT, no failure by brittle fracture would be expected. Recent studies of steel materials indicate that the NDTT may shift upwards (towards the operating temperature) as a result of exposure to neutron irradiation.

For a structural material to be susceptible to a brittle failure, several conditions must be met simultaneously. These conditions are that (a) there must be a flaw of critical size, (b) there must be a load that develops a tensile stress, and (c) the service temperature must be at or below the NDTT of the material. This last condition is affected by the neutron flux to which the element is exposed. Thus, the evaluation of the resolution of this issue depends on the assessment of the conditions stated above.

The ABWR RPV support consists of a support skirt bolted to the support pedestal. The skirt is located below the core beltline and slightly below the core support plate. It follows that the skirt is in a low neutron flux area because it is located below the core beltline. Additional shielding is provided by water flow between the vessel shroud and the vessel wall. In this situation, the effects of neutron flux would be negligible. Based on this information, the staff determined that this issue is not applicable to the ABWR design and concludes that GE adequately addressed this issue for the ABWR design.

20.2.2 Issue 23: Reactor Coolant Pump Seal Failures

This issue concerns the high rate of reactor coolant pump (RCP) seal failures that challenge the makeup capacity of the ECCS in pressurized water reactors (PWRs) which could result in a small-break loss-of-coolant-accident (SBLOCA) and possibly in core damage. RCP seal failures in BWRs occur at a frequency similar to that experienced in PWRs, but the operating experience indicates that the problem in BWRs is mitigated by smaller leak rate, larger RCIC, high-pressure coolant injection (HPCI), and feedwater makeup capabilities.

The review criteria for RCP seal acceptability is to limit the possibility of an SBLOCA (which might lead to core damage) resulting from an RCP shaft seal failure. In particular, susceptibility of the auxiliary systems to failure because of an SBO should be addressed.

SSAR Section 1A.2.30 states that the ABWR wet motor RIPs do not include seals. During a loss of preferred power (LOPP), the RIPs shutdown automatically. There are no shaft seals which require cooling water restoration. Based on this information, the staff determined that this

issue does not apply to the ABWR design and concludes that GE adequately addressed this issue for the ABWR design.

20.2.3 Issue 25: Automatic Air Header Dump on BWR Scram System

This issue concerns the slow loss of control air pressure in the scram system of BWRs. Air pressure dropping at a certain rate will first allow some of the control rod drive (CRD) scram outlet valves to open slightly, thus filling the scram discharge volume (SDV) with water, but allowing little or no control rod movement. Eventually, the rods will try to scram but the scram will be impaired. Meanwhile, the dropping air pressure may cause a transient (e.g., via controller lockup), which would normally call for a scram.

The review criteria for this issue are stated in an NRC memorandum for G. Lainas, et al., from P. Check, "BWR Scram Discharge System Safety Evaluation," dated December 1, 1980.

SSAR Section 19B.2.32 states that the ABWR is different from other BWRs in that there is no SDV employing the locking piston control rod drive in this design. For the ABWR fine motion control rod drive (FMCRD) design, scram water is discharged directly into the reactor vessel, instead of the discharge volume, as was done in previous BWR designs. The ABWR has no SDV as does previous BWR designs.

Two other features are incorporated in the design of the ABWR:

- A scram air header low-pressure alarm to alert the operator of a low pressure in the header. The pressure in the header is maintained higher than that at which the scram valves start to open and the set point is set at a sufficient margin to alert the operator to take corrective action.
- A rod block and alarm initiated by low pressure and a scram initiated by low-low pressure in the common header supplying the charging water to the scram accumulators. The accumulators have sufficient water volume to scram the associated control rods as long as the CRD system pump maintains the pressure in the charging header above the minimum required accumulator charging pressure. If pressure in the header drops below the acceptable level, the instrumentation located in the charging header will initiate an immediate scram. Thus, the accumulator

charging header low pressure causes the automatic shutdown before the accumulator is depleted.

The ABWR design incorporates two features to prevent the loss or impairment of the scram function because of slow loss of control air in the system: (1) a low pressure alarm to alert the operator to trouble in the scram air header and (2) an accumulator charging header low pressure scram to automatically shut down the plant before the accumulator is depleted. Based on this information, the staff concludes that GE adequately addressed this issue for the ABWR design.

20.2.4 Issue 29: Bolting Degradation or Failure in Nuclear Power Plants

This issue addresses degradation of bolts in nuclear power plants, especially those constituting an integral part of the primary pressure boundary, such as closure studs and bolts on reactor vessels, reactor coolant pumps, and other safety-related equipment and components. Failure of these bolts or studs could result in a loss of reactor coolant and thus jeopardize the safe operation of a plant. There is also a concern regarding bolts used as component supports or embedment anchor bolts that are essential for withstanding transient loads resulting from abnormal or accident conditions. This issue was resolved without the establishment of any new requirements.

The review criteria used for evaluating this issue are taken from NUREG-1339, "Resolution of Generic Safety Issue 29: Bolting Degradation or Failure in Nuclear Power Plants." More specifically, proven bolting designs and selection of proper materials and fabrication techniques should be employed to assure acceptable performance of bolts and studs used in vital areas of nuclear power plants.

In SSAR Section 19B.2.62, GE states that for the ABWR designs only proven materials for the specific application and environment will be employed. The materials will be selected after evaluation of the potential for corrosion wastage and IGSCC. Also, the RCPB components and their integral bolts, including the reactor vessel, reactor coolant pumps, and piping will be fabricated, tested, and installed in accordance with the requirements of the ASME Code, Sections III and XI. The ABWR design will also comply with the guidelines of NUREG-1339; EPRI NP-5769, "Degradation and Failure of Bolting in Nuclear Power Plants;" and GL 91-17, "Generic Safety Issue 29, 'Bolting Degradation or Failure in Nuclear Power Plants'."

This approach is acceptable to the staff and, therefore, the staff concludes that GE adequately addressed this issue for the ABWR design.

Generic Issues

20.2.5 Issue 40: Safety Concerns Associated With Pipe Breaks in the BWR Scram System

This issue is concerned with a possibility of a break or leak in the scram discharge volume (SDV) during a reactor scram. If such a leak or break develops, it would result in the release of water and steam at 100 °C (212 °F) into the reactor building at a maximum flow rate of 2,082 Lpm (550 gpm) and it is postulated to result in 100-percent relative humidity in the reactor building. This could be mitigated by closing the scram exhaust valves that are located on the hydraulic control units, but this depends upon ability to reset the scram, which cannot be absolutely assured immediately following the scram. Therefore, a rupture of the SDV could result in an unisolable break outside of primary containment. This break is postulated to threaten emergency core cooling equipment by flooding areas in which this equipment is located and by causing ambient temperature and relative humidity conditions for which this equipment is not qualified.

The review criteria for this issue are stated in the NUREG-0803, "Generic Safety Evaluation Report Regarding Integrity of BWR Scram System Piping," which provides guidance to ensure pipe integrity, detection capability, mitigation capability, and qualification of the emergency equipment to the expected environment.

This issue is not applicable to the ABWR design. SSAR Sections 4.6 and 19B.2.33 state that for the FMCRD design, scram water is discharged through the drive directly into the reactor vessel. There are no CRD withdraw lines or SDV as used in previous BWR designs employing the locking piston control rod drive. Consequently, the safety concerns associated with pipe break are not applicable to the ABWR. The staff concludes that GE adequately addressed this issue for the ABWR design.

20.2.6 Issue 45: Inoperability of Instrumentation Due to Extreme Cold Weather

This issue was raised after an event at Arkansas Nuclear One, Unit 2, in which all four refueling water storage tank (RWST) instrumentation channels were lost when the level transmitters froze. The system heat-tracing circuit was deenergized because the main line fuse was removed. This situation would have prevented the automatic changeover of the ECC from the injection to the recirculation mode under LOCA conditions, that is, a loss of safety function could have occurred. Typical safety-related systems employ pressure and level sensors that use small bore instrumentation lines. Most operating plants contain

safety-related equipment and systems, parts of which are exposed to the ambient environment. These lines contain liquid that is susceptible to freezing. Where systems or components and their associated instrumentation are exposed to subfreezing temperatures, heat tracing and/or insulation should be used to mitigate the effects of cold temperatures.

The review criterion for this issue is that the fluid in safety-related instrument sensing lines must be protected from freezing and maintained above the precipitation point. Guidance on the design of protective measures against freezing in instrument lines of safety-related systems are stated in the RG 1.151, "Instrument Sensing Lines." RG 1.151 endorses and augments the Instrument Society of America (ISA) standard ISA-S67.02 (1980), "Nuclear-Safety-Related Instrument Sensing Line Piping and Tubing Standards for Use in Nuclear Power Plants." RG 1.151 augments ISA-S67 by indicating that freezing temperatures be added to the environmental and installation conditions of the standard and that special considerations in the design of instrument sensing lines include freezing temperatures. Further guidance is provided in SRP Sections 7.1, Revision 3; Section 7.1, Appendix A, Revision 1; Section 7.5, Revision 3; and Section 7.7, Revision 3. SRP Section 7.1 provides for identification of the I&C systems important to safety. The acceptance criteria consist of the GDC (identified in SRP Table 7.1) and IEEE 279, "Criteria for Protection Systems for Nuclear Power Generating Stations." SRP Section 7.5 provides that the information systems important to safety provide the operator with the information on the status of the plant to allow manual safety actions to be performed when necessary. SRP Section 7.7 provides that the control systems used for normal operation, that are not relied upon to perform safety functions, but which control plant processes having a significant impact on plant safety, be acceptable and meet the relevant requirements of GDC 13 and 19.

As a proposed resolution of this issue, SSAR Section 19B.2.34 states that all safety-related systems and components used in the ABWR design, including instrument sensing lines, will be located in temperature-controlled environments. These environments will be maintained above the freezing (or precipitation) point of the contained fluid. SSAR Appendix 3I demonstrates that the temperature of these environments is not expected to be below 10° C (50 °F). The section also states that the ABWR will meet the guidance of RG 1.151.

Based on this information, the staff concludes that GE adequately addressed this issue for the ABWR design.

20.2.7 Issue 51: Proposed Requirements for Improving the Reliability of Open Cycle Service Water Systems

This issue addresses the subject of service water system (SWS) fouling at operating plants by aquatic bivalves. The SWS is the UHS that, during an accident or transient, cools the reactor building component cooling water heat exchangers, which in turn cool the RHR heat exchangers, as well as provide cooling for safety-related pumps and area cooling coils. Fouling of the safety-related SWS either by mud, silt, corrosion products, or aquatic bivalves has led to plant shutdowns, reduced power operation for repairs and modifications, and degraded modes of operation.

The review criteria for this issue consist of elimination of the possible effects of fouling of SWS and UHSs.

SSAR Section 19B.2.35 indicates that the COL applicant is given specific requirements and guidance on achieving this goal, including instructions to consider designs and new requirements that further mitigate the fouling effects. Additionally, the COL applicant is directed to investigate the problem with ice as a flow blockage mechanism and to dispose of and/or dissolve such ice, as required.

The staff studied the conditions that allow fouling and compared alternative surveillance and control programs to minimize SWS fouling. The staff's technical findings were published in NUREG/CR-5210, "Technical Findings Document for Generic Issue 51: Improving the Reliability of Open-Cycle Service-Water Systems."

SSAR Section 19B.2.35 states that the design basis for the SWS is in accordance with the EPRI ALWR Requirements Document and RG 1.27, "Ultimate Heat Sink for Nuclear Power Plants" (Rev. 2). The specifics of the design basis are given in SSAR Section 19B.2.35 as follows. Section 19B.2.35 states that the direct service water will not be used for component cooling. A closed-loop component cooling system will be used to transfer heat from the component heat loads via a heat exchanger to the SWS and ultimate heat sink. This design will minimize the number of pieces of equipment that could be in contact with the problem-causing service water. Additionally, the COL applicant will treat raw service water to reduce the effects of mud, silt and/or organisms, will select the materials for piping, pumps, and heat exchangers to offer greater resistance to the probable water chemistry, and will provide for inspections and replacements of piping during plant life. The COL applicant will also provide sufficient redundancy of makeup pumps for the UHS to allow for malfunction of one of them according to the guidelines of RG 1.27 (Rev. 2) and provide the safety-related portions

of the systems to meet the design bases during a loss of power. These systems will be designed to perform their cooling function assuming a single active failure in any mechanical or electrical system.

GE identified this issue as a COL applicant action in SSAR Section 9.2.15.2.2, "Power Generation Design Bases (Interface Requirements)," and in SSAR Section 9.2.15.2.3, "System Description (Conceptual Design)." The staff verified that GE established a COL action item (Item 9-12) in SSAR Table 1.9-1. The staff also verified that SSAR Section 9.2.17.2, "COL License Information, Reactor Service Water System Requirements," addresses measures that will be used to prevent organic fouling, erosion, and corrosion. This approach is acceptable to the staff. However, the staff will review the COL applicant's proposed resolution of this issue on a case-by-case basis.

Based on this information, the staff concludes that GE adequately addressed this issue for the ABWR design.

20.2.8 Issue 57: Effects of Fire Protection System Actuation on Safety-Related Equipment

This issue addresses fire protection system (FPS) actuations that have resulted in adverse interactions with safety-related equipment at operating nuclear power plants. Events have shown that safety-related equipment subjected to FPS water spray could be rendered inoperable and that numerous spurious actuations of the FPS have been initiated by operator testing errors or by maintenance activities, steam, or high humidity in the vicinity of FPS detectors. The NRC issued Office of Inspection and Enforcement (IE) Information Notice 83-41, "Actuation of Fire Suppression System Causing Inoperability of Safety-Related Equipment," to alert licensees and provide recent examples in which FPS actuations caused damage or inoperability of systems important to safety. In addition, the staff is considering the need for modifying FPS requirements or licensing review procedures.

The review criterion for this issue is to verify, per GDC 3 of 10 CFR Part 50, Appendix A, that a fire detection and fighting system of appropriate capacity shall be provided and designed to minimize the adverse effects of fires on structures, systems, and components important to safety. It further states that fire fighting systems shall be designed to assure that their rupture or inadvertent operation does not significantly impair the safety capability of those structures and components. Criteria in BTP CMEB 9.5-1, Section C.7.1, state that automatic fire suppression should be installed to combat any diesel generator or lubricating oil fires; such systems should be designed for operation when the diesel is running without affecting the diesel.

Generic Issues

SSAR Section 19B.2.36 states that the ABWR incorporates design features that prevent the inadvertent actuation of fire protection systems and limit the effects of water spray onto safety-related equipment. The automatic fire suppression systems protecting the safety-related equipment are of the preaction automatic type that require the detection of a fire by infrared and/or rate-of-rise heat detectors, and that require the opening of the fusible link sprinkler heads. Furthermore, each division has its own dedicated detection and actuation equipment for the control of the automatic closed head sprinklers in that divisional area. The first of the two actuation signals required to initiate the fire suppression system will annunciate an alarm to alert the operator to any potential problem. The operator has the capability of terminating the flow of fire suppressant locally by manual isolation valve.

The safety-related equipment that could be damaged because of flooding discharge of a sprinkler system is further protected by being elevated and by providing adequate drainage.

Based on this information, the staff concludes that GE adequately addressed this issue for the ABWR design.

20.2.9 Issue 67.3.3: Steam Generator Staff Actions - Improved Accident Monitoring

This issue addresses several weaknesses in accident monitoring that were observed at the Ginna Nuclear Plant. More precisely, they included (1) non-redundant monitoring of RCS pressure, (2) failure of the position indication for the steam generator relief and safety valves, and (3) the limited range of the charging pump flow indicator for monitoring charging flow during accidents. Under these conditions, it is difficult for the operating personnel to decide what corrective action they should take in situations when such an action is needed.

The review criteria used for this issue are those contained in the RG 1.97, (Rev. 3), "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions and Environs During and Following an Accident." RG 1.97 describes the methods acceptable to the NRC staff for complying with the Commission's requirements to provide instrumentation to monitor plant variables and systems during and following an accident in a nuclear plant. It describes the parameters that are necessary for the operating personnel to be monitored so that an appropriate corrective action could be taken. It also sets the requirements for the instrumentation to be functional in case of an emergency.

SSAR Section 19B.2.37 states that the ABWR has implemented into its basic design the guidance of RG 1.97

and the TMI Action Plan requirements of NUREG-0737 and NUREG-0737, Supplement 1. Section 19B.2.37 refers also to Section 7.5.1.1, Table 7.5.2, and SSAR Section 18.2. Section 7.5.1.1 describes the design features of the plant that indicate its conformance to the provisions of RG 1.97. Plant variables are defined in conformance to the definitions contained in the RG and consist of five "types" and three "categories," also according to the RG. SSAR Table 7.5-3 lists 12 Type A "variables," such as neutron flux, RPV water level, and RPV pressure.

SSAR Section 18.2 states that during all phases of normal plant operation, abnormal events, and emergency conditions, the ABWR will be operable by two reactor operators. In addition, the operating crew will include one assistant control room shift supervisor, one control room shift supervisor, and two or more auxiliary equipment operators. Four licensed operators will be on shift at all times, consistent with the staffing requirements of 10 CFR 50.54(m). The main control room staff size and roles will be evaluated and implemented by the COL applicant. The staff verified that GE established a COL action item in SSAR Section 18.8.2 regarding the adequacy of control room staffing. The design acceptance criteria (DAC)/ITAAC will further ensure compliance with the HFE design and implementation process with regard to the control room and the remote shutdown station.

Based on this information, the staff concludes that GE adequately addressed this issue for the ABWR design.

20.2.10 Issue 75: Generic Implications of ATWS Events at the Salem Nuclear Plant

On two occasions in 1983, Salem Unit 1 failed to scram automatically because both reactor trip breakers (RTBs) failed to open on receipt of an actuation signal. In both cases, the unit was successfully tripped manually. The failure of the breakers was attributed to excessive wear from improper maintenance of the undervoltage relays that receive the trip signal from the protection system and result in the breaker's opening mechanically.

Three separate actions were initiated to address this problem. One was plant specific and was addressed before restart of Salem Unit 1. The second action was an investigation of the Salem events and the circumstances leading to them. The third action was the formation of an NRC task force to study the overall generic implications of this event. The results of the task force's work were reported in NUREG-1000, Volume 1, "Generic Implications of ATWS Events at the Salem Nuclear Power Plant."

The results of investigations of the Salem Generic Implications Task Force published in NUREG-1000, Volume 1, were later outlined in proposed actions for licensees, applicants, and the NRC staff in SECY-83-248, "Generic Actions for Licensees and Staff in Response to the ATWS Events at Salem Unit 1." Furthermore, in July 1983, NRC issued the required actions for licensees and applicants in GL 83-28, "Required Actions Based on Generic Implications of Salem ATWS Events," and the internal staff actions as NUREG-1000, Volume 2.

SSAR Section 19B.2.38 describes acceptable criteria pertinent to the issue that would satisfy the regulatory requirements. These criteria state that:

- The plant must have a program for a post-trip review of unscheduled reactor shutdowns.
- The plant must have a program for safety-related equipment classification and vendor interface.
- The plant must have a program for post-maintenance operability testing.
- The plant must have a program to control vendor-related modifications, preventive maintenance, and surveillance for reactor trip breakers.

SSAR Section 19B.3.1 indicates that the COL applicant is responsible for providing resolutions of issues identified as "COL Applicant" in the "Safety Issues Index" of SSAR Appendix 19B. The SSAR indicates that issues are identified for COL action because they pertain to operating personnel issues, operating procedures, and other topics beyond the scope of the ABWR design certification review. SSAR Section 19B.1.1 lists specific documentation the COL applicant is to provide for resolution of such issues.

GE identified this issue for COL applicant action in the "Safety Issues Index." The staff verified that GE established a COL action item (Item 19-28) in SSAR Table 1.9-1 to address unresolved generic and TMI safety issues. This approach is acceptable to the staff. The staff will review the COL applicant's proposed resolution of this issue on a case-by-case basis.

20.2.11 Issue 78: Monitoring of Fatigue Transient Limits for Reactor Coolant System

This issue concerns the fact that repeated thermal cycling of RCS components produces some degree of fatigue degradation of the material that could lead to failure, thereby increasing the likelihood of a LOCA. The staff expressed the concern that for many older operating reactors, no TS requirements exist for monitoring the

actual number of transient occurrences. For newer operating reactors, it is required that the licensees keep account of the number of transient occurrences to ensure that transient limits, based on design assumptions, are not exceeded. Additionally, the staff determined that the fatigue curves used in ASME Code, Section III, may not be adequate in taking into account environmental effects. Recent data indicated that the existing code fatigue curves may have less margin than originally intended when the effects of fatigue induced by the operating environment were considered.

The review criteria for this issue are that plants implement TS to monitor plant transients and environmental effects on the fatigue life of ASME Code, Section III, Class I carbon steel piping. For ASME Code, Class 2 and 3 or Quality Group D components that are subjected to cyclic loading, an appropriate analysis is required.

SSAR Section 19B.2.39 states that the ABWR TS 5.7.2.9 requires that the monitoring of plant transients be performed to ensure that RCPB components are maintained within design limits, and that environmental effects will be included in the design bases for materials. The calculated core damage frequency (CDF) includes the environmental effects on fatigue resistance of materials.

The tentative procedure to evaluate the environmental effects on material fatigue that is currently used for a foreign BWR plant design was presented to the staff during an audit at the GE offices in San Jose, California, on March 23 through 26, 1992. In SSAR Section 3.9.3.1.1.7, GE commits to perform additional evaluations for environmental effects on the fatigue design of ASME Code, Section III, Class 1 carbon steel piping in accordance with GE document 408HA414 (non-proprietary). As discussed in Section 3.12.5.7 of this report, the staff found that the conditions and the methodology proposed by GE constitute supplemental guidelines that enhance the design margins beyond the requirements of the ASME Code, Section III, for fatigue evaluation and are acceptable.

Based on this information, the staff concludes that GE adequately addressed this issue for the ABWR design.

20.2.12 Issue 82: Beyond Design-Basis Accidents in Spent Fuel Pools

A typical spent fuel storage pool with high-density storage racks can hold about five times the fuel in the core. If the pool were to be drained of water, the discharged fuel from the last two refuelings might still be "fresh" enough to melt under decay heat. Additionally, the zircaloy cladding of this fuel could be ignited during the heatup. The

Generic Issues

resulting fire, in a pool equipped with high-density storage racks, could spread to most or all of the fuel in the pool. This could cause a release of fission products from the fuel matrix.

The review criteria for this issue are contained in SRP Sections 9.1.2 through 9.1.5 and RG 1.13, "Spent Fuel Storage Facility Design Basis." The highlights of the guidelines contained in these documents are as follows:

- The spent fuel pool should be a seismic Category I structure.
- The spent fuel pool should be designed to withstand heavy load drops without pool leakage that would uncover the top of the fuel. The spent fuel pool will be arranged to prevent cask movement over the pool.
- There should be no connections to the pool that could allow the pool to be drained below the minimum level over the spent fuel.

Although the likelihood of the complete draining of the spent fuel pool is low, the use of high-density storage racks does increase the probability of a zircaloy cladding fire as compared with the use of low-density or open frame racks. The use of low-density storage racks, for the most recently off loaded fuel, as a minimum, is justified by a favorable value-impact ratio for new designs.

In SSAR Section 19B.2.63, GE listed several design features that have been incorporated into the ABWR to provide the degree of safety for the spent fuel pool required for resolution of the issue. These features are as follows:

- The spent fuel pool is located inside the reactor building, a seismic Category I structure, and is, therefore, protected against seismic loads, tornadic winds and the associated missiles, as well as turbine missiles. It is also protected from other type of missiles because of the absence of non-seismic systems, high- or moderate-energy piping, and rotating machinery in the vicinity of the spent fuel pool. Also, connections from the RHR system to the fuel pool cooling and cleanup (FPC) and suppression pool cleanup (SPCU) systems provide a seismic Category I, safety-related makeup capability to the spent fuel pool. In SSAR Section 9.1.3.3, GE states that following an accident or seismic event, the filter-demineralizers are isolated from the cooling portion of the fuel pool cooling and cleanup system and the SPCU system by two block valves in series at both the inlet and outlet of the common filter-demineralizer portion. Seismic Category I Quality Group C bypass lines are provided

on both the fuel pool cooling and cleanup system and SPCU system to allow continued flow of cooling and makeup water to the spent fuel pool.

- Spent fuel is protected against heavy loads, including the fuel cask, by means of interlocks that prevent travel of the reactor building crane over the spent fuel storage pool.
- The SGTS limits the potential release of radioactive iodine and other radioactive materials because it is located inside the reactor building.
- No inlets, outlets, or drains are provided that might permit the pool to be drained below a safe shielding level.

In addition to the above features, SSAR Section 9.1.3 states the following:

- Fire protection is provided by means of standpipes in the reactor building and the water supplies of which are seismically designed. GE states that an analysis indicates that under the maximum abnormal heat load with the pool gates closed and no pool cooling taking place, the pool temperature will reach about 100 °C (212 °F) in about 16 hours. This provides sufficient time for the operator to hook up fire hoses for the pool makeup.

Based on the above information, the staff determined that the ABWR meets the guidelines or the pertinent regulatory documents listed above. Based on this information, the staff concludes that GE adequately addressed this issue for the ABWR design.

20.2.13 Issue 83: Control Room Habitability

This issue emphasizes the significant discrepancies found during a survey of existing plant control rooms. These discrepancies highlighted deficiencies in the maintenance and testing of ESFs designed to maintain control room habitability. They also provided examples of design and installation errors, including degradation of control room leak tightness, and pointed out a shortage of NRC and licensee personnel knowledgeable about HVAC systems and nuclear air-cleaning technology.

Loss of control room habitability following an accidental release of external airborne toxic or radioactive material or smoke can impair or cause loss of the control room operator's capability to safely control the reactor and could lead to a core damaging accident. Use of the remote shutdown station outside the control room following such events is unreliable since this station has no emergency

habitability or radiation protection provisions similar to the control room's.

The review criterion for this issue is to verify that the control room is designed to provide adequate protection to the operating personnel during and following an accident. The design must meet the guidance given in the SRP Sections 6.4, 6.5.1, 9.4.1, and 15.6.5.5. The design must be in accordance with the requirements of GDC 2, 4, and 19 of 10 CFR Part 50, Appendix A, and ASME AG-1, "Code on Nuclear Air and Gas Treatment" (1991) and the ASME AG-1a-92 Addenda, ASME 509, and ASME N510.

SSAR Sections 9.4.1 and 6.4 describe control room habitability. The control room is designed to withstand the effects of natural phenomena, missiles, and postulated accidents in accordance with GDC 2 and 4. Design of the ambient conditions (HVAC system) permits safe occupancy during abnormal conditions. Radiation exposure of control room personnel during any of the postulated design-basis accidents does not exceed the requirements of GDC 19, that is, 0.05 Sv (5 rem) whole-body radiation exposure. Smoke and toxic gas protection will be provided by the use of noncombustible materials, purging by the HVAC, individual respirators, and site-specific considerations of potential chemical releases. SSAR Section 9.4.1.1.7 states that ESF filter trains comply with the design, testing, and maintenance provisions of RG 1.52. SSAR Appendices 9C and 9D provide information on how the ABWR design meets RG 1.52 and SRP Table 6.5.1-1, respectively. The staff's review of this section found that the ABWR conforms with the design, testing, and maintenance provisions of RG 1.52.

Based on this information, the staff finds that the ABWR is adequately designed to ensure the habitability of the control room under normal and accident conditions and meets the requirements of GDC 2 regarding the systems being capable of withstanding the effects of earthquakes; GDC 4 as it relates to maintaining environmental conditions in the control room compatible with the design limits of essential equipment located therein during normal, transient, and accident conditions; and GDC 19 with regard to providing adequate protection to permit access and occupancy of the control room under accident conditions. Therefore, the staff concludes that GE adequately addressed this issue for the ABWR design.

Three other issues are related to this issue: B-36, B-66, and III.D.3.4. They are discussed in Sections 20.1.25, 20.1.30, and 20.4.87, respectively, of this report.

20.2.14 Issue 86: Long-Range Plan for Dealing with Stress Corrosion Cracking in BWR Piping

This issue addresses leaks that were detected in the heat-affected zones of the safe-end-to-pipe welds in two of the 71-cm (28-in.) diameter recirculation loop safe ends at Nine Mile Point Unit 1. Subsequent ultrasonic examination revealed extensive cracking at many weld joints in the recirculation system. The cause of the cracking was determined to be IGSCC. Addressing existing power plants, this issue offers four possible solutions for preventing or mitigating the effects of IGSCC. These recommendations are contained in NUREG-0313, Revision 2, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping."

The ABWR design meets RG 1.44, "Control of the Use of Sensitized Stainless Steel," and NUREG-0313. This will ensure that significant sensitization in ABWR components will be avoided. Further, the ABWR design does not utilize recirculation piping and, therefore, the issue as it relates to cracking of the recirculation loop piping in BWRs does not apply to the ABWR. Based on this information, the staff concludes that GE adequately addressed this issue for the ABWR design.

20.2.15 Issue 87: Failure of High Pressure Coolant Injection Steam Line Without Isolation

This issue addresses a postulated break in the HPCI steam supply line and the uncertainty regarding the operability of the HPCI steam supply line isolation valves under those conditions. A similar situation can occur in the RWCU system. The HPCI steam supply line has two containment isolation valves in series (one inside the containment and one outside), both of which are normally open in most plants (two plants do operate with the outboard isolation valve normally closed). An HPCI supply valve located adjacent to the turbine and the turbine stop valve are normally closed. The RWCU system also has two normally open containment isolation valves that must remain open if the system is to function.

At the valve manufacturers' facilities, only the opening characteristics are tested under operating conditions (because of flow limitations). Although the operation of the valves is tested periodically without steam, the capability of the valves to close against the forces created by the steam flow resulting from a downstream line break has not been demonstrated. The valve type is not under

Generic Issues

GE's (BWR vendor) scope of control, but is selected by the plant architect-engineer. This results in a diversity of valves and valve types (Y-type globe valves and gate valves) and increases the difficulty of demonstrating valve operating capability.

This issue is addressed in GL 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance," dated June 28, 1989, with Supplements 1-4. The staff's specific concern is the adequacy of testing, maintenance, and inspection of motor-operated valves (MOVs) so as to assure that they will remain functional when subjected to the design-basis conditions that are to be considered during both normal operation and abnormal events within the design basis of the plant.

SSAR Section 3.9.6.2.2 provides commitments for the design, qualification, testing, and inspection of MOVs. The details of the staff's evaluations are contained in Section 3.9.6.2.2 of this report. The staff's evaluation concludes that the commitments as described in SSAR Section 3.9.6.2.2 provide reasonable assurance for demonstrating the adequacy of the MOV isolation capability for the design-basis conditions and are, therefore, acceptable. Furthermore, SSAR Section 3.9.7.3 also states that the COL applicant will address the design qualification and testing for MOVs as discussed in SSAR Section 3.9.6.2.2 prior to plant startup.

Based on this information, the staff concludes that GE adequately addressed this issue for the ABWR design.

20.2.16 Issue 89: Stiff Pipe Clamps

Stiff pipe clamps were installed because of requirements for piping systems to withstand dynamic loads such as SRV discharges to suppression pools, LOCA-induced loads, and seismic loads. A preloading of pipe clamp U-bolts or straps, which imposes a constant compressive load on the piping, is necessary to prevent stiff pipe clamps from lifting under dynamic loading. In addition to the large preloading of the clamps, four other design features were identified as requiring additional analyses because of their differences from conventional pipe clamps. Those were (1) use of high-strength or non-ASME-approved materials; (2) local surface contact on the pipe; (3) uncommonly thick and/or wide design of clamp; and (4) clamp applications to piping components other than straight pipe, such as pipe elbows.

It was found that piping designers often assumed that the clamp effects on piping systems were negligible and did not warrant any explicit consideration. Although this assumption was acceptable for most clamp applications, in some cases, piping systems coupled with specific pipe

clamp design requirements could experience interaction effects that need to be evaluated to determine the magnitude of pipe stresses induced.

The ASME Code, Section III, requires that the effects of attachments in producing thermal stresses, stress concentrations, and restraints on pressure-retaining members be taken into consideration in checking for compliance with stress criteria. The review criteria for this issue are that the effects of stiff clamps on piping stresses should be included in the piping system design. That is, in designing a piping system, cumulative stress contribution, such as due to thermal expansion of the pipe and clamp, discontinuity stresses in the pipe, stresses from thermal gradient through the pipe wall, and the external loads from dynamic events such as earthquake should also include the stresses attributed to the stiff clamps.

SSAR Section 19B.2.43 states that a study was performed in 1980 for typical stiff pipe clamps on BWR main steam and recirculation piping systems. For each system, the stiff clamps were installed on straight pipe or on bends with a radius of at least five pipe diameters. The purpose of these analyses was to evaluate the additional stresses at clamp locations resulting from the following:

- Differential thermal expansion of the pipe and clamp
- Discontinuity stresses in the pipe from internal pressure restraint
- Thermal gradient through the pipe wall in the vicinity of the pipe clamp
- External loads resulting from dynamic events such as earthquakes.

The results of these calculations showed that the total primary and secondary stresses, including clamp-induced stresses, were less than 70 percent of code-allowable stresses. GE states that the governing stress locations occurred at pipe branch connections, elbows, and shear lugs, and that they did not occur at stiff clamp locations. GE says this indicates that the stress intensification that occurs at elbows, branch connections, and shear lugs is greater than that occurring at stiff pipe clamps. Based on these calculations GE concluded, and the staff concurred, that explicit consideration of clamp-induced piping stresses is not required when the clamps are installed on straight pipe or on bends with a radius of at least five pipe diameters.

GE states in the SSAR section referred to above that the stiff clamp pipe analysis described above will be extended to the ABWR design and the pipe design specifications on other than NSSS piping will also consider these stress requirements.

Based on this information, the staff concludes that GE adequately addressed this issue for the ABWR design.

20.2.17 Issue 103: Design for Probable Maximum Precipitation

This issue concerns the use of the most recent National Oceanic and Atmospheric Administration (NOAA) procedures for determining probable maximum precipitation (PMP). More specifically, the issue was centered upon the NRC use of NOAA Hydrometeorological Reports (HMRs) Nos. 51 and 52, published in June 1978 and August 1982, respectively. The PMP values are used in estimating design flood levels at reactor sites. Objections were raised against use of these reports because by using HMR-52, higher values of flood levels were obtained than those when earlier reports were used. That would constitute unauthorized backfit under NRC procedures.

The procedures for estimating PMP and, therefore, the probable maximum flood (PMF) acceptable to the NRC are given in Appendices A and B of RG 1.59, "Design Basis Floods for Nuclear Power Plants," (Appendix A has since been superseded by ANSI N170-1976, "Standards for Determining Design Basis Flooding at Power Reactor Sites." ANSI N170-1976 defines the PMF as a hypothetical flood that is considered to be the most severe reasonably possible, based on comprehensive hydrometeorological application of PMP and other hydrologic factors favorable for maximum flood runoff. Thus, PMP is an integral component of PMF determination.

GDC 2 of 10 CFR Part 50, Appendix A, requires that the design bases for floods reflect consideration of the most severe historical flood data, with sufficient margin for the limited accuracy, quantity, and period of time in which the data have been accumulated. Additional guidance for estimating PMP and PMF is contained in SRP Sections 2.4.2, Revision 3, and 2.4.3 Revision 3, and in GL 89-22, "Potential for Increased Roof Loads and Plant Area Flood Runoff Depth at Licensed Nuclear Plants Due to Recent Change in Probable Maximum Precipitation Criteria Developed by the National Weather Service."

SSAR Section 19B.2.44 states that the ABWR design meets the requirements of GDC 2. In SSAR Table 2.0-1, GE provides the envelope of the ABWR site design parameters, among which is the maximum precipitation (for roof design) expressed in terms of a maximum rainfall rate of 49.3 cm/hr (19.4 in./hr), and a maximum snow load of 2.354 kPa (.341 psi). In Section 4.5.2.2 of the EPRI Evolutionary SER (NUREG-1242, "NRC Review of Electric Power Research Institute's Advanced Light Water Reactor Utility Requirements Document"), the staff stated

that a 5-minute PMP is reasonable and that the 49.3 cm/hr (19.4 in./hr) PMP, together with 2.6 km² (1.0 mi²), 5-minute PMP of 15.7 cm (6.2 in.) appears to be acceptable to the staff. However, that might exclude a number of sites in the Great Lakes area. GE further states that the ABWR meets the intent of SRP Sections 2.4.2, Revision 3 and 2.4.3, Revision 3 and GL 89-22. Detailed site characteristics based upon historical site environmental data will be provided by the site owner-operator for any specific applications. The staff verified that GE established a COL action item in SSAR Section 2.3 to provide site characteristics information. This approach is acceptable to the staff.

Based on this information, the staff determined that the ABWR is designed in accordance with the requirements of GDC 2 for the most severe environmental conditions, including floods, tornadoes, and hurricanes, that are expected and meets the intent of SRP Section 2.4.2 Revision 3, SRP Section 2.4.3 Revision 3, and GL 89-22. Therefore, the staff concludes that GE adequately addressed this issue for the ABWR design.

20.2.18 Reserved.

20.2.19 Issue 105: Interfacing Systems LOCA at LWRs

To protect against an intersystem LOCA (ISLOCA), designers of future ALWR plants should reduce the possibility of a LOCA outside containment by designing, to the extent practicable, all systems and subsystems connected to the RCS to an ultimate rupture strength (URS) at least equal to the normal RCS operating pressure.

Enhancements of isolation capability or the number of intersystem barriers (e.g., three isolation valves) are not considered to be adequate alternatives in systems that can be practically designed to the URS criteria. For example, piping runs should be designed to meet the URS criteria, as should all associated flanges, connectors, and packing, including valve stem seals, pump seals, heat exchanger tubes, valve bonnets, and RCS drain and vent lines. The designer should also make every effort to minimize the pressure loading experienced by each system and subsystem connected to the RCS should an ISLOCA occur. The staff does recognize, however, that all systems must eventually interface with atmospheric pressure and that it would be difficult or prohibitively expensive to design certain large tanks and heat exchangers to an URS equal to normal RCS operating pressure.

Applicants must provide justification demonstrating that it is not practicable to reduce the pressure challenge any

Generic Issues

further for each interfacing system and component that does not meet the RCS URS. This justification must be based upon an engineering feasibility analysis and not solely on the ratio of risk to benefit. Applicants must also demonstrate a compensating isolation capability for each interface for which it submits acceptable justification on the impracticability of normal RCS operating pressure capability. This would include a discussion of how the degree and quality of isolation or the reduced severity of the pressure challenges compensate for the low-pressure design of the interfacing system or component. The vendor may also need to consider the adequacy of pressure relief and piping of relief back to primary containment. In SECY-90-016, the staff stated that systems that have not been designed to full RCS pressure must include the following protection measures:

- the capability for leak testing of the pressure isolation valves,
- valve position indication that is available in the control room when isolation valve operators are deenergized, and
- high-pressure alarms to warn control room operators when rising RCS pressure approaches the design pressure of the attached low pressure systems and both isolation valves are not closed.

In the DFSER, the staff reported on its review of GE's preliminary evaluation of the RHR, HPCF, SLC, and RCIC systems and indicated that it would complete its review of Issue 105 upon receiving and reviewing additional information from GE. This was identified in the DFSER as Open Item 20.2-3.

In the original ABWR design, low-pressure piping that indirectly interfaces with the RCS (such as the RHR system suction piping) was designed to 1,277 kPa (200 psig). This design pressure was not considered to be acceptable since the piping was susceptible to pipe break because of ISLOCA. The staff required GE to upgrade the pressure rating of the piping that interfaces with the RCS. GE provided its implementation of the issue resolution in a submittal dated October 8, 1992, from J. Fox to C. Poslusny (NRC), "Proposed Resolution of ISLOCA Issue for ABWR." The staff reviewed the GE submittal and concluded that the design pressure for the low-pressure piping systems that interface with the RCS pressure boundary should be equal to 0.4 times the normal reactor operating pressure of 6,965 kPa (1025 psig), that is, 2,786 kPa (410 psig), and the minimum wall thickness of the low-pressure piping should be no less than that of a standard weight pipe. The staff concluded that this minimum pressure will ensure reasonable protection

against burst failure should the low-pressure system be subjected to full pressure.

GE agreed with the staff position and made the necessary design changes as described below. All the low-pressure piping was changed to 2,786 kPa (410 psig). Furthermore, the staff will continue to require periodic surveillance and leak rate testing of the pressure isolation valves per TS requirements as a part of the ISI program. The details of the staff position on this issue is given in Section 3.9.3.1.1 of this report.

The following items form the basis of what constitutes practicality and set forth the test of practicality used to establish the boundary limits of URS for ABWR.

It is impractical to construct large tank structures to the URS design pressure that are vented to the atmosphere and have a low design pressure. Tanks included in this category are:

- Condensate storage tank
- SLCS main tank
- Low-conductivity waste receiving tank
- High-conductivity waste receiving tank
- FPC skimmer surge tank
- FPC spent fuel storage pool and cask pit
- Condensate hotwell.

Also included as impractical to upgrade were the suppression pool and primary containment. The suppression pool provides a low-pressure sink, approximately -96.31 kPa (0.75 psig) for its interfacing systems, and the first closed valve is rated to at least 2,786 kPa (410 psig). The ABWR containment is designed to 209 kPa (45 psig) and is designed to seismic Category I.

GE upgraded the design pressure of the following small tanks as a result of the review:

- SLCS test tank
- RCIC turbine barometric condenser tank.

Based on the staff guidance described above, GE evaluated in SSAR Appendix 3MA the following systems that interface with the RCS to verify that they are designed for an ISLOCA "to the extent practicable":

- RHR system
- HPCF system
- Standby liquid control system
- RCIC
- Control rod drive (CRD) system
- Reactor water cleanup system
- FPC system

- Nuclear boiler system (NBS)
- Reactor recirculation system (RRS)
- Makeup water (condensate) (MUWC) system
- Makeup water (purified) (MUWP) system
- Radwaste system
- Condensate and feedwater (CFS) system
- Sampling (SAM) system

The pressure of each system piping boundary was reviewed to identify where changes were needed to provide the URS protection. Where low-pressure piping interfaces with higher-pressure piping that is connected to the RCS, design pressure values were increased to 2,786 kPa (410 psig). The low-pressure piping boundaries were upgraded to URS pressures and extended to the last closed valve connected to the piping interfacing a low-pressure sink, such as the suppression pool or the condensate storage tank. For some systems, with low-pressure piping and normally open valves, the valves were changed to locked-open valves to ensure a pathway from the last URS boundary to the tank or low-pressure sink. Also, the minimum wall thickness for all the piping was changed to the standard weight pipe.

It is the staff's position that components such as heat exchangers, flanges, and pump seals should also be designed to a pressure of 2,786 kPa (410 psig). These changes have been implemented and are indicated in the SSAR by revised boundary symbols in the P&IDs. A stated parameter (e.g., design pressure) of a boundary symbol on the P&ID applies to all the piping and components on the P&ID that extend away from the boundary symbol, including any branch line, until another boundary symbol occurs on the P&ID.

GE upgraded the design pressure of piping in 14 systems that interface with the RCS and changed the design pressure of two tanks as the result of the ISLOCA review. Based on the above, the staff concludes that using the staff guidance, GE has modified the ABWR systems design "to the extent practicable" and, hence, has adequately addressed this issue for the ABWR design. Therefore, DFSEER Open Item 20.2-3 is resolved.

Based on the above discussions, the staff concludes that the ABWR design meets the criteria of SECY-90-016 regarding ISLOCA prevention and mitigation.

20.2.20 Issue 106: Piping and Use of Highly Combustible Gases in Vital Areas

Issue 106 addresses the risk associated with the use of hydrogen and other combustible gases, such as propane and acetylene, during normal plant operation. This issue does not cover the use of large quantities of liquid

hydrogen at hydrogen water chemistry (HWC) installations at BWRs or liquified petroleum gases (which are covered under Licensing Issue 136).

The review criteria for this issue are taken from EPRI Report NP-5283-SR-A, "Guidelines for Permanent BWR Hydrogen Chemistry Installations," and SRP Section 9.5.1. The current SRP Section 9.5.1, with the BTP CMEB 9.5-1, Part C.5.d(5), should be modified as follows: "Hydrogen lines in safety-related areas should follow the guidance of RG 1.29, "Seismic Design Classification," Section C.2. The lines should (1) be equipped with an excess flow valve or equivalent protection located outside the building so that in case of a line break, the hydrogen concentration in affected areas does not exceed 2 percent volume or (2) be sleeved with the outer pipe vented directly to the outside." The criteria should be applied to systems that supply hydrogen for cooling of the electric generators.

The system design should comply with the following general guidance:

- Design features and administrative controls should be provided to prevent inadvertent bypass of small or normally isolated hydrogen supplies.
- Flow limiting devices should be used to limit hydrogen releases to a leak or pipe break.
- Equipment and controls to mitigate the consequences of a hydrogen fire or explosion should be accessible and remain functional during an event.
- Design features and administrative controls should be provided to isolate the hydrogen supply if normal building ventilation is lost.
- Backflow to a leak or line break of hydrogen contained in components (e.g., generator) should be considered in evaluating the consequences of leaks or breaks and measures taken to mitigate these consequences.
- Threaded joints in the hydrogen distribution lines within safety-related areas should be back welded.

EPRI Report NP-5283-SR-A provides guidelines for HWC installations. With the exception of information dealing specifically with the HWC application (e.g., certain trips, injection points, and main steam line radiation), most of the EPRI guidelines dealing with hydrogen are applicable to these other uses. The guidelines give a number of system design features and administrative controls that are in addition to, or more restrictive than, those in SRP Section 9.5.1. In addition, safety-related equipment should

Generic Issues

not be located in the turbine building because of the hazards associated with hydrogen fires or explosions and large oil fires and the large uncertainties in estimating the consequences.

According to SSAR Section 9.3.9.1.2, hydrogen is used in the ABWR design to reduce the dissolved oxygen in the reactor water in order to mitigate the potential for IGSCC of sensitized austenitic stainless steels. The amount of hydrogen required is in the range of 1.0 to 1.5 ppm, but the exact amount, which depends on many factors, including for instance, incore recirculation rates, will be determined by tests performed during initial operation of the plant.

SSAR Section 19B.2.48 states that the ABWR design uses hydrogen for HWC and the main generator bulk hydrogen supply system. These systems are non-nuclear and non-safety-related and are located in the turbine building, which is a non-safety-related structure in a nonvital area.

SSAR Section 10.2.2.1 states that there are no safety-related systems or components located within the turbine building, hence any local failure associated with the turbine-generator (T-G) unit will not affect any safety-related equipment. It further states that failure of T-G equipment cannot preclude safe shutdown of the RCS. SSAR Sections 9.3.9.1.2 and 10.2.2.2 state that the HWC system and the bulk hydrogen system, respectively, utilize the guidelines given in EPRI Report NP-5283-SR-A with respect to those portions of the guidelines involving hydrogen which do not deal specifically with the HWC system. Specifically, the bulk hydrogen system and HWC system piping and components will be located to reduce risk from their failures. The bulk hydrogen storage is located outside but near the turbine building. The arrangement of buildings at the facility and location of building doors and the bulk hydrogen storage tanks will be designed to ensure that damage to buildings containing safety-related equipment due to combustion of hydrogen or an explosion is unlikely.

Hydrogen lines are provided with a pressure-reducing station before the piping enters the turbine building which limits the maximum flow of hydrogen to less than 100 standard cubic meters per minute (3530 scfm). The hydrogen piping inside the turbine building will be designed in accordance with the guidance of RG 1.29, Position C.2, regarding the seismic design of non-safety-related equipment whose failure could affect safety-related equipment. This is in order to comply with the modified BTP CMEB 9.5-1, Part C.5.d(5). Additionally, all threaded joints in the hydrogen distribution piping will be back welded. Equipment and controls used to mitigate the

consequences of a hydrogen fire or explosion will be designed to be accessible and remain functional during the postulated post-accident condition. Design features and/or administrative controls will be provided to ensure that the hydrogen supply is isolated when normal building ventilation is lost.

Based on this information, the staff concludes that GE adequately addressed this issue for the ABWR design.

20.2.21 Issue 113: Dynamic Qualification Testing of Large Bore Hydraulic Snubbers

Large bore hydraulic snubbers (LBHSs) have a load rating greater than 22.68 tonnes (50 kips). They are active mechanical devices used to restrain safety-related piping and equipment during seismic or other dynamic events (e.g., high-energy line breaks), yet also allow sufficient piping and component flexibility to accommodate system expansion and contraction resulting from thermal transients, such as normal plant heatups and cool downs. Dynamic qualification testing and periodic functional testing are important to verify that the LBHSs are properly designed and maintained for the life of the plant. Issue 113 addresses the need for requirements for dynamic testing of LBHSs.

SSAR Section 19B.2.64 states that for the ABWR design, LBHSs will only be used for piping systems when dynamic supports are required at locations where large thermal displacements prohibit the use of rigid supports. They will not be used in applications other than piping restraints. Section 19B.2.64 refers the reader to SSAR Section 3.9.3.4.1(3) for information on design, testing, installation, and pre-service examination of mechanical and hydraulic snubbers, including LBHSs. To assure snubber functionality under various normal and abnormal conditions, snubbers are to be designed in accordance with provisions of the ASME Code, Section III, Subsection NF. This design requirement includes analysis for normal, upset, emergency, and faulted loads. These calculated loads are then compared with the allowable loads to verify that the stresses are below the code-allowable limits.

GE described the tests that the snubbers will be subjected to, and they will consist of the following:

- Force or displacement versus time loading at frequencies within the range of significant modes of the piping system.
- Dynamic cyclic tests to determine the operational characteristics of the snubber control valve.

- Displacement tests to determine the specified performance characteristics.

The snubbers will be tested for various abnormal environmental conditions. Upon completion of the abnormal environmental transient test, the snubbers will be tested dynamically at a frequency within a specified frequency range.

Based on this information, the staff concludes that GE provided acceptable commitments to dynamically test LBHSs, and therefore, adequately addressed this issue for the ABWR design.

20.2.22 Issue 118: Tendon Anchorage Failure

An inspection of a PWR prestressed concrete containment structure showed that three lower vertical tendon anchor heads were broken. The failures appeared to be caused by stress corrosion cracking. Quantities of water ranging from a few milliliters (oz.) to about 5.7 liters (1.5 gals.) were found in the grease caps.

A reinforced concrete containment for BWRs will be used. Since a prestressed concrete containment is not specified, no specific requirements for tendon anchorage are provided.

SSAR Sections 3.8.1 and 19B.2.48 state that the primary containment of the ABWR standard plant is designed as a reinforced concrete structure. Since the technique of prestressed concrete design has not been used by GE, the issue of tendon anchorage failure is not applicable for the containment structure design of the ABWR standard plant. Therefore, the staff concludes that GE adequately addressed this issue for the ABWR design.

20.2.23 Issue 120: On-Line Testability of Protection Systems

During its 1985 review of several plant TS, the staff discovered that the design of protection systems of some plants did not provide as complete a degree of on-line, at-power surveillance testing capability as did other plants undergoing staff review and evaluation at that time. This raised questions about the on-line testability of protection systems and the possibility that some nuclear power plants might not provide complete testing capability. Issue 120 was established to examine these questions. Protection systems consist of the reactor protection system (RPS) and the engineered safety features actuation system (ESFAS). The main concern of this issue, however, is the on-line testability of the actuation subgroup (slave) relays in the ESFAS.

The requirements for at power testability of components are included in GDC 21 of 10 CFR Part 50, Appendix A. RGs 1.22, "Periodic Testing of Protection System Actuation Functions," and 1.118, "Periodic Testing of Electric Power and Protection Systems," and IEEE 338-1977, "Criteria for the Periodic Testing of Nuclear Power Generating Station Safety Systems," provide supplemental guidance. This guidance is intended to ensure that protection systems (including logic, actuation devices, and associated actuated equipment) will be designed to permit testing while a plant is operating at power without adversely affecting the plant's operation. The scope of testing covered consists of functional tests, checks, calibration verifications, and time-response measurements. Criteria are provided for determining system operational availability, status, and necessary documentation, and for establishing test intervals and test procedures during operation.

The review criteria applied for this issue included an assessment of the capability for periodic functional testing of the systems. This periodic testing should be manually initiated, but automatically performed once initiated, and should meet the guidance of RGs 1.22 and 1.118 and IEEE 338. Automatic initiation of periodic testing may be provided where the testing does not degrade the system functionality. Built-in, automated test features are expected to be provided for periodic, functional testing, as necessary, to eliminate physical reconfiguration of systems (e.g., adding jumpers, lifting leads, swapping cables) to perform the required tests. The safety-related systems are to have automatic test features that are sufficient to meet TS requirements for periodic surveillance of the system's functionality as defined by RGs 1.22 and 1.118 and IEEE 338.

This issue was resolved during the staff's USI-GSI review for the ABWR without the establishment of new requirements. Nevertheless, the staff evaluated the capability of the ABWR for continuous on-line self-testing of hardware and system integrity as detailed above.

SSAR Sections 7.1.2.1.6 and 19B.2.49 indicate that the ABWR design's RPS and ESFAS can be tested at power during reactor operation by six separate tests. The first five tests are primarily manual tests and although each individually is a partial test, combined with the sixth test they constitute a complete system test. The sixth test is the self-test of the system logic and control that automatically tests the complete system excluding sensors and actuators.

The sixth test is an integrated self-test provision built into the microprocessors within the safety system logic and control (SSLC). It consists of on-line, continuously operating, self-diagnostic, monitoring network and an off-

Generic Issues

line semi-automatic (operator initiated, but automatic to completion), end-to-end surveillance program. This testing includes the following:

- On-line continuous testing, which is a self-diagnostic program monitoring each signal-processing module from input to output (actuation of the trip functions is not performed during this test)
- Off-line semi-automatic end-to-end testing, which exercises the trip outputs of the SSLC logic processors.

All testing features adhere to the single-failure criterion.

Since the design of the systems permit periodic testing of their functioning when the reactor is in operation, including a capability to test channels independently to determine failures and losses of redundancy that may have occurred, the staff determined that the ABWR meets the requirements of GDC 21 of 10 CFR Part 50, Appendix A. Therefore, the staff concludes that GE adequately addressed this issue for the ABWR design.

20.2.24 Issue 121: Hydrogen Control for Large, Dry PWR Containments

As a result of the TMI-2 accident, the Commission promulgated regulatory requirements on hydrogen control in 10 CFR 50.34(f) and 50.44. 10 CFR 50.34(f) requires a hydrogen control system based on 100-percent, fuel-clad metal-water reaction and hydrogen concentration limit of 10 percent on uniformly distributed hydrogen in the containment or on a post-accident atmosphere that will not support hydrogen combustion.

The review criterion for this issue is that the control of hydrogen generated in the containment in a degraded core accident shall meet the requirements of 10 CFR 50.34(f) on limiting the distributed hydrogen concentration to 10 percent, on limiting the combustible concentration, and on maintaining safe-shutdown equipment and containment integrity.

This issue is relevant to large, dry containments, such as those used in PWRs. Therefore, this issue does not apply to the ABWR design. Additionally, the ABWR primary containment is inerted and is, therefore, protected from hydrogen combustion regardless of the amount or rate of hydrogen generation.

Based on this information, the staff concludes that GE adequately addressed this issue for the ABWR design. See also the staff's evaluation of 10 CFR 50.34(f)(2)(ix) in Section 20.5.21 of this report.

20.2.25 Issue 124: Auxiliary Feedwater System Reliability

In 1985, it was observed by the NRC staff and industry that the auxiliary feedwater systems (AFWSs) continued to fail at a high rate. These studies also indicated that plants with similar AFWS reliabilities (as calculated in accordance with the SRP guidance) did not necessarily exhibit similar AFWS availabilities. Based on these studies and on engineering judgement, the staff concluded that the PWR AFW system reliabilities calculated in accordance with the SRP guidance may have represented the relative reliability of AFW system hardware configurations for various plants, but did not represent the real availability of the system.

A function of the AFWS in most cases is to supply water to the secondary side of the steam generator during system fill, normal plant startup, normal plant standby, and normal plant cold shutdown. The AFWS also functions following loss of normal feedwater flow, including loss because of offsite power supply failure, and provides emergency feedwater (EFW) following such postulated events as main feedwater line break or main steam line break.

The AFWS reliability criterion has been specified in SRP Section 10.4.9. For the ABWR, SSAR Section 19B.2.51 states that the acceptance criteria for resolution of this issue will be that the AFWS will be designed for a high degree of reliability (that is, using reliability analyses the system will attain 0.0001 to 0.00001 unavailability per demand).

As it has been pointed out above, the function of the AFWS in most cases is to supply water to the secondary side of the steam generator under various conditions. Since this condition does not exist in case of the ABWR, this issue is not applicable to the ABWR.

20.2.26 Issue 128: Electrical Power Reliability

Concerns have been raised regarding the dependence on Class 1E power, especially dc power, of the decay heat removal systems required for long-term heat removal. Failure of one division would generally result in a reactor scram, which would then require removal of decay heat. The frequency of reported failures of single dc divisions gives rise to the concern that the second dc division may not be available. This issue combines three interrelated issues, namely, A-30, "Adequacy of Safety-Related DC Supplies;" 48, "LCOs for Class 1E Vital Instrument Buses in Operating Reactors;" and 49, "Interlocks and LCOs for Redundant Class 1E Tie Breakers."

Issue A-30 involves the power source to the inverters for the vital 120-Vac instrument power supplies that are related to Issues 48 and 49. It is also related to Issue 76, "Instrumentation and Control Power Interactions," because a loss of 120-Vac vital instrument power could challenge emergency safeguards systems and could cause reactor trips, loss of feedwater, loss of emergency core and containment cooling systems, and loss of post-accident monitoring instrumentation.

Issue A-48 concerns the fact that in some plants, there are no administrative controls governing operational restrictions for their Class 1E 120-Vac vital instrument buses (VIBs) and associated inverters. Without such restrictions, these power sources could be out of service indefinitely and might place certain safety systems in a situation where they could not meet the single-failure criterion. This is of particular concern during the period before the start and load of diesel generators following a loss of offsite power. In this condition, some VIBs may be subjected to power failure modes that may not have been considered during the safety analysis of the plant.

Issue A-49 arises from the fact that in some plants there is lack of adequate procedural and administrative controls that are used to monitor and provide assurance that the tie breakers between redundant Class 1E divisions of electrical power and multi-units are always open during plant operation. Such controls are necessary to provide assurance that the Class 1E power buses are not compromised. There is also a related concern, when a tie breaker is closed, involving electrical interlocks to prevent out-of-synchronization interconnections of a diesel generator to either the off-site power source or another diesel generator.

The review criteria for this issue are contained in NUREG/CR-5414, "Technical Findings for Proposed Integrated Resolution of Generic Issue 128, Electric Power Reliability." Generally, this document states that the plant design should provide for the separation and isolation of electrical power systems to preclude interactions that could adversely affect such functions as diesel generator loading and offsite to diesel generator power transfers.

Each division of the ESFs requiring electric power will be provided with an onsite source of ac and dc power. At least two separate and independent connections will be provided to offsite power sources capable of starting and running all Class 1E loads required for safe shutdown. The specified functions of the ESFs will be met by the use of redundant divisions and that the divisions will be totally independent and separated both mechanically and electrically. The DHR systems will be redundant and safety grade.

The plant designs will have three independent divisions for the core coolant inventory control and DHR systems and each division will have its own independent ac and dc power source.

Separation of electrical power systems will be such as to preclude interactions that could adversely affect the functioning of the dc power systems. Specifically, the use of bus ties between safety divisions is prohibited.

Non-safety-related loads will be placed on power supplies that are completely separate from those on which safety-related loads are placed.

The loss of any plant battery or dc bus concurrent with a single independent failure in any other system required for shutdown cooling will not result in a total loss of reactor cooling capability.

Each reactor protection channel will be normally powered from a dedicated Class 1E source that is normally independent of other dc sources.

SSAR Section 19B.2.52 describes the ABWR design features pertinent to the resolution of Issues 128, A-30, 48, and 49.

For Issue A-30, the dc buses for the safety-related dc power system meet the acceptance criteria because of the following:

- The safety-related dc power system does not supply power to any non-Class 1E loads.
- Consists of four separate and independent dc battery systems.
- Does not contain any direct bus tie between dc battery systems. However, it does contain two standby battery charges, each of which is capable of supplying one of the divisional dc systems. Redundant key-locked breakers are provided to prevent manual paralleling between divisions, and no automatic connections are provided between dc divisions.

The ABWR design meets the acceptance criteria for the resolution of Issue 48 by the system design and TS. The ABWR design consists of four separate and independent Class 1E 120-Vac vital instrument buses with their respective inverters. TS contain the appropriate operating restrictions, to assure the onsite Class 1E ac and dc power distribution system availability and thus, an uninterruptable power source for safety-related systems and components. The TS contain specific requirements regarding a periodic evaluation of the onsite power system bus condition which

Generic Issues

addresses such availability items as correct breaker and bus alignment and bus voltage.

The ABWR meets the acceptance criteria with regards to Issue 49. The ABWR Class 1E system design does not contain bus tie breakers between Class 1E divisions, but it is possible to manually cross-connect the Class 1E diesel buses through the CTG connections, since the power to each diesel bus can be provided from the CTG. Each diesel generator is provided with a synchronizing equipment for paralleling offsite supplies. The normal and alternate offsite feeder breakers to Class 1E buses are interlocked to prevent paralleling offsite circuits.

Based on this information, the staff concludes that GE adequately addressed this issue for the ABWR design.

20.2.27 Issue 130: Essential Service Water Pump Failures at Multiplant Sites

This issue addresses shared essential service water (ESW) systems between PWR units located on the same site. Specifically, this issue deals with multiplant sites that have only two ESW pumps per plant with cross-tie capability and the impact of such sharing on the availability of the ESW pumps. The safety concern is that the needed ESW pumps may not be available during all possible operating conditions for the multiple units, thereby increasing the core melt and radiological risks at the site. Other multiplant and single plant configurations may also contain similar ESW system vulnerabilities. Therefore, the concern of this issue is equally applicable to other multiplant and single plant PWR sites. According to Appendix B to NUREG-0933, Issue 130 is not applicable to BWR vendors.

Issue 130 was resolved with the issuance on September 19, 1991, of GL 91-13, "Request for Information Related to the Resolution of Generic Issue 130, 'Essential Service Water System Failures at Multi-Unit Sites,' Pursuant to 10 CFR 50.54(f)." This letter contained TS and emergency procedures improvements for seven specific multiplant PWR sites.

Although this issue is not applicable to BWR plants, the staff included an evaluation of the concerns as related to the ABWR RSW design in the DFSER. In the DFSER, the staff said that the SSAR stated that the ABWR has been designed as a single unit with no specific consideration of possible shared systems. (Note that in all system analysis reviews, the requirements of GDC 5 have been identified as not being applicable to the ABWR design.) Shared systems are, therefore, not a concern for the ABWR design.

The staff reported in the DFSER that SSAR Appendix 19B, "Resolution of Applicable Unresolved Safety Issues and Generic Safety Issues," Sections 19B.2.10 and 19B.3.5, allowed for the possible modification of the ABWR design by plant-specific applicants. The guidelines provided by GE to assure that the modified design would address the concerns of Issue 130 included reference to two design requirements from the EPRI ALWR Requirements Document. The first requirement was for limiting the number of shared systems to the "test programs which have been made" and further states that these systems "shall incorporate the pertinent results into the design of the ALWR." The second requirement can be interpreted to address shared systems at multiplant sites stated that each division of any ESW system must have two heat exchangers and two pumps sized so that each division can provide the capacity to absorb the system heat loads generated by the plant during all operational (normal and emergency) modes.

The staff stated in the DFSER that the requirements mentioned above did not provide sufficient guidance to an applicant referencing the ABWR design for a multiplant site to design ESW systems for the units that reflect an acceptable resolution of the concern of Issue 130. Specifically, the staff indicated that the first EPRI requirement as identified by GE is confusing. The subject requirement as stated in the Requirements Document (see Volume 2, Chapter 1, Section 6.2.B.1) limits the number of shared systems to auxiliary support systems such as sewer, auxiliary steam, or site security. Further, another EPRI requirement (see Requirements Document, Volume 2, Chapter 8, Item B.1.4.2) clarifies the above EPRI requirement by specifically ruling out sharing of ESW pumps between divisions and between units of a multiplant site and by requiring the ESW system for each unit to be designed to the same requirements as for a single unit. The EPRI requirement as identified by GE, on the other hand, implies that the ESW system can be shared between the units, provided certain conditions are satisfied. However, the staff related that the ABWR SSAR provided little or no guidance regarding whether the ESW system pumps will be shared between the units and if so, how they will be shared, the number of shared as well as non-shared ESW pumps for each unit, the capacity of each ESW pump, and the operational limitations required to minimize system misoperation (human errors associated with the wrong train or the wrong plant identification). In addition, the staff said that GE should address the applicable reliability concerns of Issue 130 for single-plant units.

In the DFSER, the staff required GE to address all the above concerns and modify the two identified interface requirements accordingly to provide sufficient guidance to the applicants referencing the ABWR design. The staff

stated that such guidance is needed so that each plant-specific applicant will be able to provide supporting documentation and analysis to justify modifications to the ABWR essential cooling water systems if a multi-unit site is to incorporate shared cooling water systems in the modified designs. The staff indicated that the analysis and documentation should demonstrate that the revised plant design meets the concerns identified in Issue 130, as well as the requirements of GDC 5 with regard to the sharing of structures, systems, and components at multi-unit sites. This was identified in the DFSE as Open Item 20.2-4.

Further, the staff required in the DFSE that the COL applicant address the plant-specific resolution of Issue 130. This was identified as COL Action Item 20.2-3. The staff stated that it would review the plant-specific response to the concerns of Issue 130 and that the review would include, among other things, the available redundancy in ESW pumps within a division to achieve any licensing-basis requirement, available flexibility for needed recovery actions, and specific measures to preclude potential operator errors.

GE revised the SSAR to delete the discussion of Issue 130, because it is not applicable to BWR plants. The SSAR discussions and EPRI requirements that are the subject of DFSE Open Item 20.2-4 no longer exist in SSAR Sections 19B.2.10 and 19B.3.5. SSAR Section 1.1.6 still states that the ABWR has been designed as a single-unit plant, therefore, the staff need not be concerned with specific consideration of possible shared systems and with addressing Issue 130. Because Issue 130 is not applicable to BWR plants and based on the subsequent revisions to the SSAR discussed above, the staff concludes that DFSE Open Item 20.2-4 is resolved.

Because this issue is not applicable to the ABWR and based on the above discussion of the ABWR design as a single-unit plant without any shared systems, the staff concludes that the COL applicant need not prepare a plant-specific response to the issue. On this basis, the staff concludes that DFSE COL Action Item 20.2-3 was not warranted and need not be addressed.

20.2.28 Issue 142: Leakage Through Electrical Isolators in Instrumentation Circuits

This issue addresses electrical isolators used to maintain electrical separation between safety-related and non-safety-related electrical systems in nuclear power plants, preventing malfunctions in the non-safety-related systems from degrading the performance of safety-related circuits. The primary concern is that the amount of energy that could pass through certain types of isolation devices (and be transmitted to safety-related circuitry) during certain

electrical transients might damage or seriously degrade the performance of Class 1E components. Or, this energy could cause the isolation devices to give false output, or the electrically generated noise on the circuit might cause the isolation device to give a false output.

The review criteria for this issue are contained in a letter from T. Murley (NRC) to R. Artigas (GE), "Advanced Boiling Water Reactor Licensing Review Bases," dated August 7, 1987. This letter contains the staff's expectations regarding isolation devices. The letter describes the design and environmental qualifications that insulators must satisfy. It requires that description of tests be provided, as well as other requirements, as appropriate. The letter also addresses guidance for fiber-optic cable, and states that the staff is working to develop comprehensive guidance on this subject, and that it will be based on the existing IEEE cable standards, such as IEEE 323 and 384, and applicable ANSI standards.

Therefore, the review criteria for this issue must contain guidance for:

- Inspection and testing of all electrical insulation devices between Class 1E and non-Class 1E systems.
- Replacement or repair of isolators that fail the tests, including description of acceptable hardware fixes to the isolators.
- Implementation of an annual program to inspect and test all electrical isolators between Class 1E and non-Class 1E systems.

SSAR Appendix 7A and Section 19B.2.53 state that the isolating devices used in the ABWR design are similar to those in Group 1 referred to in NUREG/CR-3453, "Electronic Isolators Used in Safety Systems of U.S. Nuclear Power Plants." Review of this reference confirmed that test results demonstrated that only minor amounts of high-frequency energy pass through the barrier during testing. These isolators, though they provide the best isolation and are recommended as the safest units, they are the most susceptible to damage. The report indicates that they are so fragile that some were damaged during the electromagnetic interference (EMI) tests.

GE indicates that the ABWR design will use a fiber-optic system for electrical isolation of logic and analog signals between protection divisions and from protection divisions to non-safety equipment. This selection of isolation devices appears to be acceptable, in view of the statement in the referenced report that "optical-fiber isolators are the newest of the isolators and consist of both analog and digital types."

Generic Issues

Another positive feature of the design is the fact that the ABWR RPS and ESF functions are not supplied directly from a commercial power source, but from different plant power sources. Thus, the low voltage dc supplies fed from these sources are regulated and filtered. This is important, because as the reference states, when power is supplied from commercial sources to the input power supply through the common ac power line, as it is in most PWRs, power surge appears to be high.

The advance SER stated that as part of the resolution of this issue, there should be a requirement stated in the acceptance criteria in SSAR Section 19B.2.53 that the following will be implemented:

- Annual inspection and testing of all electrical isolation devices between Class 1E and non-Class 1E systems.
- Repair or replacement of insulators that fail the tests.

These changes were incorporated in SSAR Amendment 34.

In the advance SER, the staff concluded that GE provided an acceptable system of isolators to resolve this issue, assuming that the testing, inspection, and replacement of isolators, when needed, would be incorporated in resolution of this issue. The staff verified that GE established a COL action item in SSAR Section 19B.3.2 and Table 1.9-1 addressing testing, inspection, and replacement of isolators. On the basis of this information, the staff concludes that GE adequately addressed this issue for the ABWR design.

20.2.29 Issue 143: Availability of Chilled Water Systems

This issue concerns partial or complete loss of the HVAC system. Many of these problems exist because of the desire to provide increased fire protection and the need to avoid severe temperature changes in equipment control circuits. The improvements in this area, which started after the Browns Ferry fire, consist mainly of enclosing the affected equipment in small, isolated rooms. This resulted in a significant reduction in room cooling rate. Plant control and safety have improved with the introduction of electronic integrated circuits. However, these circuits are more susceptible to damage from severe changes in temperature caused by the loss of room cooling.

It is believed that failures of air cooling systems for areas housing key components, such as RHR pumps, switchgear, and diesel generators, could contribute significantly to core-melt probability in certain plants. Because corrective measures are often taken at the affected plants once such failures occur, the impact of these failures on the proper

functioning of air cooling systems may not have been considered. Thus, plants with similar inherent deficiencies may not be aware of these problems.

Operability of some safety-related components is dependent upon operation of HVAC and chilled water systems to remove heat from the rooms containing the components. If chilled water and HVAC systems are unavailable to remove heat, the functionality of the equipment within the rooms may be destroyed.

The review criterion for this issue is to assure that the equipment can be functional during the period of loss of room cooling.

SSAR Section 19B.2.54 states the following criteria as the acceptable ABWR design:

- An evaluation of the dependencies or nondependencies of safety-related equipment on HVAC cooling will be performed. This evaluation will include assessment of room heat load and heatup rates, and establish equipment operating conditions. Equipment ability to withstand these conditions without loss of function will be established.
- For equipment found to be significantly dependent on the HVAC cooling, an assessment of the HVAC system reliability will be performed. PRA analyses will be carried out to assess plant risk and determine whether any modifications are necessary.
- Corrective design measures will be identified where necessary to reduce plant risk.

SSAR Section 19B.2.54 states that the following features have been incorporated into the ABWR design to satisfy the criteria stated above:

- RCIC pump and turbine are designed to operate for at least 8 hours without room cooling. This system will provide core cooling during a prolonged loss of HVAC cooling.
- Operation of other injection systems (HPCF, LPFL, RHR) is more dependent on the availability of room cooling. However, these systems are designed to operate for at least 10 minutes without room cooling. The equipment in question is designed to be operational at the highest temperature expected during that time.
- Detailed design specifications for ABWR safety-related equipment will specify the room conditions under which equipment must operate without room cooling.

Room heat assessment will be performed to establish environmental conditions for equipment specification.

- Potential modifications including procedure changes or hardware changes will be evaluated through PRA analyses to ensure acceptable risk.

GE also listed the safety-related HVAC systems that will provide room cooling under most circumstances. These include the secondary containment safety-related HVAC, reactor building safety-related electrical equipment HVAC, reactor building safety-related diesel generator HVAC, and HVAC emergency cooling water (three divisions).

The staff concludes that GE adequately addressed this issue for the ABWR design.

20.2.30 Issue 145: Actions to Reduce Common Cause Failures

This issue concerns the fact that common cause failures (CCFs) can be a major cause of a system failure.

Testing of equipment has its limitations; in fact, testing can be an important cause of CCF that occur when the testing does not reflect true demands of the equipment under operating conditions. For example, MOVs may work during a test but not during a true demand when there exists a high differential pressure across them. Much design-basis testing cannot be performed in situ. Prototypical testing, on the other hand, is expensive and the application of it to equipment in plants is sometimes not practical. Effective maintenance is important to ensure that design assumptions and margins in the original design bases are maintained. In the design of nuclear plants, an important safety margin is the redundancy of equipment to perform safety functions. This redundancy, however, can be degraded by CCFs. Thus, measures are needed to identify CCF precursors before they occur so that corrective measures can be taken.

The review criterion for the resolution of Issue 145 is to demonstrate compliance with the maintenance rule, 10 CFR 50.65, which requires that a program of performance and condition monitoring activities be evaluated at least every refueling cycle (provided the interval between evaluations does not exceed 24 months) and that industry-wide experience be incorporated in the program. When monitoring and preventive maintenance activities are performed, an assessment of the total plant equipment that is out of service should be taken into account to determine the overall effect on performance of safety functions.

Implementation of 10 CFR 50.65 is an effective and practical way to prevent or reduce CCFs.

SSAR Section 19B.2.55 states that compliance with 10 CFR 50.65 will be responsibility of the COL applicant. This approach is acceptable to the staff.

In addition, the ABWR capability to respond to system interactions and CCFs is described in the SSAR Section 19.2.3.4. Five factors are considered and incorporated in the analysis of system interactions and CCFs:

- Component commonality at the system level, such as common initiating signal.
- Common divisional services, such as common electric power buses or common service water loops.
- System dependency, such as automatic depressurization system (ADS) dependency on the operability of at least one of the five (two high-pressure and three low-pressure) ECCS pumps.
- Past experience of losing onsite power.
- Human errors.

Actions to reduce CCFs fall into the Owner/Operator's Reliability Assurance Process (O-RAP), described in SSAR Section 17.3.10 and which has been reviewed by the staff and found to be acceptable as discussed in Sections 17.3.9 and 17.3.10 of this report. The COL applicant will specify the policy and implementation procedures for the O-RAP and submit it for staff review. The staff verified that GE established a COL action item in SSAR Section 17.9.13 to make use of information provided by GE to help the owner/operator determine activities that should be included in the O-RAP. This approach is acceptable to the staff.

Based on this information, the staff concludes that GE adequately addressed this issue for the ABWR design. The staff will evaluate compliance with 10 CFR 50.65 as part of its inspection activities during plant operations.

20.2.31 Issue 151: Reliability of Recirculation Pump Trip During an ATWS

This issue concerns reliability of breakers used to trip the recirculation pumps at high pressure or low-water-level signals during ATWS mitigation in BWRs.

If a plant transient requiring a reactor scram occurs and the scram function does not occur, then an ATWS event

Generic Issues

exists. To lessen the effects of an ATWS event, negative reactivity must be added to the reactor core by tripping the recirculation pumps. Negative reactivity is added as a result of the ensuing steam voiding in the core area as the core flow decreases, thereby decreasing the power generation and limiting the power or pressure disturbance. If the recirculation pump trip (RPT) fails to trip on demand, the reactor could experience continued power generation resulting in a high suppression pool temperature.

Plants equipped with GE AKF-25 circuit breakers have experienced failures of the field breakers in the RPT system that were caused by binding of the trip latch mechanism and misadjustment of the breakers' mechanical linkage.

The review criterion for this issue is determining the use of reactor RPT hardware or a RPT method that is more reliable than the previously used AKF-2-25 breaker hardware or RPT method.

Since the design for the ABWR reactor recirculation system and RPT method is completely different from the previously designed BWR reactor recirculation systems and RPT trip methods, resolution of this issue was evaluated on the basis of the ABWR RPT information about the new design for the recirculation pump system provided by GE in SSAR Sections 7.7.1.3(7) and 7.7.1.3(8). A summary of the important design changes is provided in the following paragraphs.

GE states that the ABWR reactor recirculation system and the RPT design is completely different from the previously designed BWR reactor recirculation systems and RPT methods. They state that it is more diverse and more redundantly reliable. The ABWR uses 10 pumps and multiple pump and RPT trip logic, circuits and hardware, rather than only two recirculation pumps, as has been generally used in BWRs. The recirculation flow control (RFC) system consists of three redundant process controllers, adjustable-speed drives (ASDs), switches, sensors, and alarm devices provided for operational manipulation of the 10 RIPs and the surveillance of the associated equipment. RFC is achieved either by manual operation or by automatic operation if the power level is above 70 percent of rated power.

In the event of (a) a turbine trip or generator load rejection when reactor power is above a predetermined level, (b) the reactor pressure exceeds the high dome pressure set point, or (c) the reactor water level drops below the Level 3 set point, the RPT logic will automatically trip off a group of four RIPs. If the reactor water level continues to drop and reaches Level 2 after the first group of four RIPs have

been tripped, the remaining six RIPs will be tripped. The implementation of the second RPT function is similar to that of the first RPTs, using 2-out-of-4 confirmation logic.

It is known that plants with GE AKF-25 circuit breakers have experienced failures of the field breakers in the RPT system. In the ABWR design, instead of using AKF-25 breaker switching hardware to provide an RPT, RFC controller switching and ADS gate inverter turn-off circuit hardware provide the RPT.

The staff determined that the system described above appears to be more reliable than the previously used AKF-2-25 breaker hardware and method and, therefore, provides reasonable assurance that in the event of an ATWS, the RIPs will be tripped, thus lessening the effect of the ATWS. Based on this determination, the staff concludes that GE adequately addressed this issue for the ABWR design.

20.2.32 Issue 153: Loss of Essential Service Water in LWRs

This issue concerns reliability of the ESW supply that is critical in the transfer of heat from various safety-related and non-safety-related systems and equipment to the UHS. The ESW is needed in every phase of plant operations and, under accident conditions, supplies adequate cooling water to systems and components that are important to safe plant shutdown or to mitigate consequences of an accident. Under normal operating conditions, the ESW provides component and room cooling (mainly via the component cooling water system). During shutdowns, it also ensures that the residual heat is removed from the reactor core, cooling towers, and water treatment systems at a plant. A complete loss of the ESW system could lead to a core-melt accident, posing a significant risk to the public.

Loss of ESW can be caused by a number of reasons: various fouling mechanisms (sediment decomposition, biofouling, corrosion and erosion, foreign materials, and debris intrusion), ice effects, single failures and other design deficiencies, flooding, multiple equipment failures, and personnel and procedural errors. Additionally, the design and operational characteristics of the ESW system differ significantly from plant to plant within each reactor type. For these reasons, it is practically impossible to formulate generic ESW design criteria that would be universally applicable. The design bases of the system will conform to the requirements of GDC 2 relating to the ability of structures, systems, and mechanisms to withstand the effects of earthquakes; GDC 4 regarding the protection of safety-related equipment from the effects of internally generated missiles, pipe whip, and environmental conditions resulting from high and moderate energy line

breaks and the dynamic effects associated with flow instabilities and loads (e.g., water hammer); GDC 5 relating to shared systems and components; and GDC 44 as it relates to transferring heat from structures, systems, and components important to safety, to an ultimate heat sink. More specific review criteria are contained in SRP Section 9.2.1.

The ABWR RSW system removes heat from the reactor building cooling water (RCW) and transfers it to the UHS. The RSW system is provided in three divisions. Each division has two pumps that send cooling water to three heat exchangers. Normally, one pump and two heat exchangers are operating in each division. The remaining pump and the third heat exchanger are automatically put in operation when heat-removal requirements increase.

In case of failure of any of the three RSW divisions, the remaining two divisions are sufficient to meet safe shut down requirements for the plant. The ABWR RSW system is protected from common-cause effects by the fact that the three divisions are separated both physically and electrically from each other.

Degradation of the RSW system is prevented by periodic inspection and testing to ensure integrity and functional capability. All three divisions of the RSW system are designed to allow periodic inservice inspection of all the system components. This testing capability consists of structural and leak-tightness visual inspection, entire system operability, and system component operability and performance.

SSAR Section 19B.2.57 lists the design features for the portions of system that are not within the ABWR standard plant scope that will be provided by the COL applicant.

Based on the above, the staff concludes that GE adequately addressed this issue for the ABWR design.

20.2.33 Issue 155.1: More Realistic Source Term Assumptions

This issue is a result of the study conducted by the TMI-2 Safety Advisory Board, and is one of the seven recommendations forwarded to the NRC. The subject issue deals with the fact that during the TMI-2 accident, fission products did not behave as predicted by the analytical methods and assumptions used in the licensing process at the time and delineated in RG 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Boiling Water Reactors," RG 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors," and TID-14844,

"Calculation of Distance Factors for Power and Test Reactor Sites." Contrary to the original predictions that during the TMI-2 accident major core damage had occurred, NRC determined that approximately 50 percent of the core was in a molten state, and only about 55 percent of the highly volatile fission products and noble gases were released from the reactor vessel with a major portion retained in the reactor building. There is also evidence that less than 5 percent of the medium- and low-volatile fission products were released from the reactor vessel.

The review of this issue consists of determining that the plant is designed to ensure that the dose commitment to the public, in the event of a licensing design-basis accident, will be within those limits prescribed by existing regulations based upon the guidelines of 10 CFR Part 100.

SSAR Section 19B.2.58 states that in view of lack of adequate guidance as to acceptance methods and conditions, i.e., revised RGs and SRP, it is premature for the ABWR to use the revised source terms. It also states that the ABWR design has been analyzed in accordance with the current RGs, SRP, and the GDC, all of which are based on TID-14844. The staff evaluated offsite radiological consequences using the TID-14844 source term procedures that are consistent with the guidelines provided in the applicable SRP sections and RGs. (See Section 15.4.3 of this report.) Two deviations from the current staff position were noted: (1) credit was given for radioactive iodine removal in the main steam lines and in the main condensers by holdup for decay and deposition and (2) the staff accepted the ABWR design without an MSIV leakage control system.

The advance SER indicated that in response to the staff's concern, the BWR Owners' Group (BWROG) performed further studies to determine the consequences, if any, of the ABWR design regarding radioactive offsite releases. The staff subsequently evaluated the results of the BWROG and GE proposals and found them to be acceptable. In particular, in SSAR Chapter 15, GE performed radiological consequence assessments of certain DBAs and concluded that the ABWR design, using TID-14844 source terms with the two deviations stated above, will meet the dose reference values established in 10 CFR Part 100 and the dose limits given in GDC 19 of 10 CFR 50, Appendix A. To verify GE's conclusion, the staff independently assessed the radiological consequences resulting from DBAs, also using TID-14844 and the deviations discussed earlier, and found GE's assessments to be acceptable. The staff assessments and conclusion are discussed further in Section 15.4 of this report.

Based on this information, the staff concludes that GE adequately addressed this issue for the ABWR design.

Generic Issues

20.3 Human Factors Issues

This section addresses staff evaluation of GSIs categorized as "human factors issues" in NUREG-0933. All the following issues are relevant to the ABWR design.

20.3.1 Issue HF 1.1: Staffing and Qualifications - Shift Staffing

The purpose of this issue is to assure that the number and capabilities of the staff at nuclear power plants are adequate to provide safe operation. To meet this goal, consideration should be given to: (1) the number and functions of the staff needed to safely perform all required plant operations, maintenance, and technical support for each operational mode; (2) the minimum qualifications of plant personnel in terms of education, skill, knowledge, training experience, and fitness for duty; and (3) appropriate limits and conditions for shift work including overtime, shift duration, and shift rotation. More specifically, this issue refers to determination of the minimum appropriate shift crew staffing composition.

The review criteria for this issue are contained in the 10 CFR 50.54, SRP Section 13.1.2-13.1.3, and RG 1.114, "Guidance on Being Operator at the Controls of a Nuclear Power Plant."

This issue is beyond the scope of the ABWR design certification review and the COL applicant will be responsible for addressing it. The staff verified that GE established a general COL action item (Item 18.1.1 in SSAR Section 18.8) to conduct the detailed HFE design according to design and implementation. The staff considers this to include the resolution of Issue HF 1.1. This approach is acceptable to the staff as discussed in Section 18.7.2.2 of this report.

20.3.2 Issue HF 4.4: Procedures - Guidelines for Upgrading Other Procedures

The objective of this issue is to provide assurance that plant procedures are adequate and can be used effectively and to guide operators in maintaining plants in a safe state under all operating conditions. This latter includes the ability to control upset conditions without first having to diagnose the specific initiating event. This objective is to be met by: (1) developing guidelines for preparing and criteria for evaluating emergency operating procedures (EOPs), normal operating procedures, and other procedures that affect plant safety and (2) upgrading the procedures, training the operators in their use, and implementing the upgraded procedures.

The review criteria for this issue are contained in SRP Sections 13.5.1 and 13.5.2 and NRC Information Notice 86-64, "Deficiencies in Upgrade Programs for Plant Emergency Operating Procedures."

The development of detailed procedures and training materials is beyond the scope of the ABWR design certification review and the COL applicant will be responsible for addressing this generic issue. The staff verified that GE established a COL action item in SSAR Section 13.5.3 for procedure development. SSAR Section 13.5.3.1 indicates that the methods and criteria for the development, verification and validation (V&V), implementation, maintenance, and revision of procedures will include considerations of Issues I.C.1, I.C.5, and I.C.9. The staff considers this to also include the resolution of Issue HF 4.4. This approach is acceptable to the staff as discussed in Section 13.5 of this report.

20.3.3 Issue HF 5.1: Man-Machine Interface - Local Control Stations

The objective of this issue is to ensure that the man-machine interface is adequate for the safe operation and maintenance of nuclear power plants. The regulatory guidance has been limited to the control room and the remote shutdown panel. Further guidance is necessary regarding local control stations and auxiliary operator interfaces. To accomplish this task, analyses of control room crew and local control activities should be conducted to establish and describe communication and control links between the control room and the auxiliary control stations.

The review criteria for this issue are contained in SRP Section 18.2, Appendix A.

This issue is beyond scope of the ABWR design certification review and the COL applicant will be responsible for addressing it. The staff verified that GE established a COL action item (Item 18.8.11 in SSAR Section 18.8) to analyze this issue. This approach is acceptable to the staff as discussed in Section 18.7.2.2 of this report.

20.3.4 Issue HF 5.2: Man-Machine Interface - Review Criteria for Human Factors Aspects of Advanced Controls and Instrumentation

With the outcome of advanced technologies utilizing improved annunciator systems, guidelines for evaluation of these longer-term annunciator improvements are necessary. These guidelines will be based upon evaluations of results from advanced concept activities being performed by the

Government and commercially sponsored research programs.

The existing HFE guidelines for nuclear power plant control rooms primarily address the control, display, and information concepts and technologies that are now being used in process control systems. While these guidelines were adequate in the past, they may not be sufficient for advanced and developing technologies that are being introduced in more advanced designs.

The review criteria for this issue are contained in SRP Section 18.2, Appendix A.

In SSAR Section 18.3, GE described an integrated design implementation process to incorporate HFE principles. The process includes an integrated design of control and instrumentation systems and HSI of the ABWR. The design implementation process facilitates selection of design features that satisfy the top level requirements and goals of individual systems and the overall plant.

In its review of GE's detailed control room design process, the staff found that this issue is beyond the scope of the ABWR design certification review and that it is to be addressed by the COL applicant as part of the detailed design implementation. The staff verified that GE established a general COL action item in SSAR Section 18.8 (Item 18.8.1) to conduct the detailed HFE design according to the design and implementation as defined by the HFE ITAAC and DAC described in ABWR CDM Table 3.1 and SSAR Appendix 18E. The staff considers this to include the analysis of this generic issue. This approach is acceptable to the staff as discussed in Section 18.7.2.2 of this report.

20.4 TMI Action Plan Items

This section addresses staff evaluation of GSIs that are categorized as "TMI Action Plan items" in NUREG-0933. Except as noted, all the following issues are relevant to the ABWR design.

20.4.1 Issue I.A.1.1: Operating Personnel -- Operating Personnel and Staffing - Shift Technical Advisor

This issue requires the provision of an on-shift technical advisor, with specific education and training, to the shift supervisor.

SSAR Section 19B.3.1 states that the COL applicant is responsible for providing resolutions of issues identified as "COL Applicant" in the "Safety Issues Index" of SSAR Appendix 19B. The SSAR states that issues are identified

for COL action because they pertain to operating personnel issues, operating procedures, and other topics beyond the scope of the ABWR design certification review. SSAR Section 19B.1.1 lists specific documentation the COL applicant is to provide for resolution of such issues.

GE identified this issue for COL applicant action in the "Safety Issues Index." The staff verified that GE established a COL action item (Item 19-28) in SSAR Table 1.9-1 to address unresolved generic and TMI safety issues. This approach is acceptable to the staff. The staff will review the COL applicant's proposed resolution of this issue on a case-by-case basis.

20.4.2 Issue I.A.1.2: Operating Personnel -- Operating Personnel and Staffing - Shift Supervisor Administrative Duties

This issue requires review of the administrative duties of the shift supervisor by the senior officer responsible for plant operations. It also requires that when administrative functions detract from or are subordinate to the management responsibility for assuring the safe operation of the plant, they are to be delegated to other operations personnel not on duty in the control room.

SSAR Section 19B.3.1 indicates that the COL applicant is responsible for providing resolutions of issues identified as "COL Applicant" in the "Safety Issues Index" of SSAR Appendix 19B. The SSAR indicates that issues are identified for COL action because they pertain to operating personnel issues, operating procedures, and other topics beyond the scope of the ABWR design certification review. SSAR Section 19B.1.1 lists specific documentation the COL applicant is to provide for resolution of such issues.

GE identified this issue for COL applicant action in the "Safety Issues Index." The staff verified that GE established a COL action item (Item 19-28) in SSAR Table 1.9-1 to address unresolved generic and TMI safety issues. This approach is acceptable to the staff. The staff will review the COL applicant's proposed resolution of this issue on a case-by-case basis.

20.4.3 Issue I.A.1.3: Operating Personnel -- Operating Personnel and Staffing - Shift Manning

This issue requires adherence to the shift manning and overtime requirements for normal plant operation established by the NRC.

SSAR Section 19B.3.1 indicates that the COL applicant is responsible for providing resolutions of issues identified as "COL Applicant" in the "Safety Issues Index" of SSAR

Generic Issues

Appendix 19B. The SSAR indicates that issues are identified for COL action because they pertain to operating personnel issues, operating procedures, and other topics beyond the scope of the ABWR design certification review. SSAR Section 19B.1.1 lists specific documentation the COL applicant is to provide for resolution of such issues.

GE identified this issue for COL applicant action in the "Safety Issues Index." The staff verified that GE established a COL action item (Item 19-28) in SSAR Table 1.9-1 to address unresolved generic and TMI safety issues. This approach is acceptable to the staff. The staff will review the COL applicant's proposed resolution of this issue on a case-by-case basis.

20.4.4 Issue I.A.1.4: Operating Personnel -- Operating Personnel and Staffing - Long-Term Upgrading

The purpose of this issue was to develop changes to 10 CFR 50.54 concerning shift staffing with licensed operators and their working hours.

The resolution of this issue is beyond the scope of the ABWR design certification review and the COL applicant will be responsible for addressing it. This was identified in the DFSER as COL Action Item 18.7.2.2-8. The staff verified that GE established a general COL action item (Item 18.1.1 in SSAR Section 18.8) to conduct the detailed HFE design during design and design implementation. The staff considers this to include the resolution of Issue I.A.1.4. This approach adequately addresses DFSER COL Action Item 18.7.2.2-8 as discussed in Section 18.7.2.2 of this report.

20.4.5 Issue I.A.2.1: Operating Personnel -- Training and Qualification of Operating Personnel - Immediate Upgrading of Operator and Senior Operator Training and Qualifications

This issue requires that, effective December 1, 1980, all senior reactor operator (SRO) applicants must have been a licensed operator for at least 1 year.

SSAR Section 19B.3.1 states that the COL applicant is responsible for providing resolutions of issues identified as "COL Applicant" in the "Safety Issues Index" of SSAR Appendix 19B. The SSAR indicates that issues are identified for COL action because they pertain to operating personnel issues, operating procedures, and other topics beyond the scope of the ABWR design certification review. SSAR Section 19B.1.1 lists specific documentation the COL applicant is to provide for resolution of such issues.

GE identified this issue for COL applicant action in the "Safety Issues Index." The staff verified that GE established a COL action item (Item 19-28) in SSAR Table 1.9-1 to address unresolved generic and TMI safety issues. This approach is acceptable to the staff. The staff will review the COL applicant's proposed resolution of this issue on a case-by-case basis.

20.4.6 Issue I.A.2.3: Operating Personnel -- Training and Qualifications of Operating Personnel - Administration of Training Programs

This issue requires that, subject to the accreditation of training institutions, licensees and applicants assure that training center and facility instructors who teach systems, integrated responses, transients, and simulator courses demonstrate SRO qualifications and be enrolled in appropriate requalification programs.

SSAR Section 19B.3.1 states that the COL applicant is responsible for providing resolutions of issues identified as "COL Applicant" in the "Safety Issues Index" of SSAR Appendix 19B. The SSAR indicates that issues are identified for COL action because they pertain to operating personnel issues, operating procedures, and other topics beyond the scope of the ABWR design certification review. SSAR Section 19B.1.1 lists specific documentation the COL applicant is to provide for resolution of such issues.

GE identified this issue for COL applicant action in the "Safety Issues Index." The staff verified that GE established a COL action item (Item 19-28) in SSAR Table 1.9-1 to address unresolved generic and TMI safety issues. This approach is acceptable to the staff. The staff will review the COL applicant's proposed resolution of this issue on a case-by-case basis.

20.4.7 Issue I.A.2.6(1): Operating Personnel -- Training and Qualifications of Operating Personnel - Long-Term Upgrading of Training and Qualifications; Revise Regulatory Guide 1.8

This issue required NRC development of a revised RG 1.8 to incorporate recommendations on upgrading personnel training and qualifications.

SSAR Section 19B.3.1 states that the COL applicant is responsible for providing resolutions of issues identified as "COL Applicant" in the "Safety Issues Index" of SSAR Appendix 19B. The SSAR indicates that issues are identified for COL action because they pertain to operating personnel issues, operating procedures, and other topics beyond the scope of the ABWR design certification review.

SSAR Section 19B.1.1 lists specific documentation the COL applicant is to provide for resolution of such issues.

GE identified this issue for COL applicant action in the "Safety Issues Index." The staff verified that GE established a COL action item (Item 19-28) in SSAR Table 1.9-1 to address unresolved generic and TMI safety issues. This approach is acceptable to the staff. The staff will review the COL applicant's proposed resolution of this issue on a case-by-case basis, evaluating compliance with the version of RG 1.8 current at the time of the review.

20.4.8 Issue I.A.3.1: Operating Personnel -- Licensing and Requalification of Operating Personnel - Revise Scope of Criteria for Licensing Examinations

This issue requires the inclusion of simulator examinations as part of the licensing examinations.

SSAR Section 19B.3.1 states that the COL applicant is responsible for providing resolutions of issues identified as "COL Applicant" in the "Safety Issues Index" of SSAR Appendix 19B. The SSAR states that issues are identified for COL action because they pertain to operating personnel issues, operating procedures, and other topics beyond the scope of the ABWR design certification review. SSAR Section 19B.1.1 lists specific documentation the COL applicant is to provide for resolution of such issues.

GE identified this issue for COL applicant action in the "Safety Issues Index." The staff verified that GE established a COL action item (Item 19-28) in SSAR Table 1.9-1 to address unresolved generic and TMI safety issues. This approach is acceptable to the staff. The staff will review the COL applicant's proposed resolution of this issue on a case-by-case basis.

20.4.9 Issue I.A.4.1(2): Operating Personnel -- Simulator Use and Development - Initial Simulator Improvement; Interim Changes in Training Simulators

This issue requires the following capabilities for simulators: modeling saturation conditions; providing multiple-failure accident training, including incorrect instrument responses; providing training for both active and passive failure of ESF components; providing training on natural circulation operation under solid water conditions; and other simulator weaknesses that may have been identified under I.A.2.6 and I.A.4.2.

SSAR Section 19B.3.1 states that the COL applicant is responsible for providing resolutions of issues identified as "COL Applicant" in the "Safety Issues Index" of SSAR

Appendix 19B. The SSAR states that issues are identified for COL action because they pertain to operating personnel issues, operating procedures, and other topics beyond the scope of the ABWR design certification review. SSAR Section 19B.1.1 lists specific documentation the COL applicant is to provide for resolution of such issues.

GE identified this issue for COL applicant action in the "Safety Issues Index." The staff verified that GE established a COL action item (Item 19-28) in SSAR Table 1.9-1 to address unresolved generic and TMI safety issues. This approach is acceptable to the staff. The staff will review the COL applicant's proposed resolution of this issue on a case-by-case basis.

20.4.10 Issue I.A.4.2: Operating Personnel -- Simulator Use and Development - Long-Term Training Upgrade

Refer to the evaluation of 10 CFR 50.34(f)(2)(i) in Section 20.5.13 of this report.

20.4.11 Issue I.C.1: Operating Procedures - Short-Term Accident Analysis and Procedures Revision

The objective of this issue was to improve the analysis of design-basis and off-normal transients and accidents and the procedures for handling them. Actions to address this issue include the performance of analyses of small-break LOCAs, inadequate core cooling, transients, and accidents; preparation of emergency procedure guidelines (EPGs); implementation of appropriate emergency procedures; and training of operators.

The development of detailed procedures and training materials is beyond the scope of the ABWR design certification review and the COL applicant will be responsible for addressing this TMI item. The staff verified that GE established a COL action item in SSAR Section 13.5.3 for procedure development. SSAR Section 13.5.3.1 states that the methods and criteria for the development, V&V, implementation, maintenance, and revision of procedures will include considerations of I.C.1. This approach is acceptable to the staff as discussed in Section 13.5 of this report.

20.4.12 Issue I.C.2: Operating Procedures - Shift Relief and Turnover Procedures

This issue requires that plant procedures include provisions to assure that shift and relief turnover is adequately prescribed to ensure that each oncoming shift is aware of critical plant status information and system availability prior to assuming duties.

Generic Issues

SSAR Section 19B.3.1 states that the COL applicant is responsible for providing resolutions of issues identified as "COL Applicant" in the "Safety Issues Index" of SSAR Appendix 19B. The SSAR states that issues are identified for COL action because they pertain to operating personnel issues, operating procedures, and other topics beyond the scope of the ABWR design certification review. SSAR Section 19B.1.1 lists specific documentation the COL applicant is to provide for resolution of such issues.

GE identified this issue for COL applicant action in the "Safety Issues Index." The staff verified that GE established a COL action item (Item 19-28) in SSAR Table 1.9-1 to address unresolved generic and TMI safety issues. This approach is acceptable to the staff. The staff will review the COL applicant's proposed resolution of this issue on a case-by-case basis.

20.4.13 Issue I.C.3: Operating Procedures - Shift Supervisor Responsibilities

This issue requires review and revision of plant procedures and directives to assure that duties, responsibilities, and authority are properly defined to establish a definite line of command and clear delineation of the command decision authority of the supervisor in the control room relative to other plant management personnel. It also requires training programs for shift supervisors to emphasize and reinforce the responsibility for safe operation and the management function of the shift supervisor to assure safe operation of the plant.

SSAR Section 19B.3.1 states that the COL applicant is responsible for providing resolutions of issues identified as "COL Applicant" in the "Safety Issues Index" of SSAR Appendix 19B. The SSAR states that issues are identified for COL action because they pertain to operating personnel issues, operating procedures, and other topics which are beyond the scope of the ABWR design certification review. SSAR Section 19B.1.1 lists specific documentation the COL applicant is to provide for resolution of such issues.

GE identified this issue for COL applicant action in the "Safety Issues Index." The staff verified that GE established a COL action item (Item 19-28) in SSAR Table 1.9-1 to address unresolved generic and TMI safety issues. This approach is acceptable to the staff. The staff will review the COL applicant's proposed resolution of this issue on a case-by-case basis.

20.4.14 Issue I.C.4: Operating Procedures - Control Room Access

This issue requires that the authority and responsibilities of the person in charge of control room access and clear lines

of authority and responsibility in the control room in the event of an emergency be established in conformance with item 2.2.2.a of NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations," dated July 1979.

SSAR Section 19B.3.1 states that the COL applicant is responsible for providing resolutions of issues identified as "COL Applicant" in the "Safety Issues Index" of SSAR Appendix 19B. The SSAR states that issues are identified for COL action because they pertain to operating personnel issues, operating procedures, and other topics beyond the scope of the ABWR design certification review. SSAR Section 19B.1.1 lists specific documentation the COL applicant is to provide for resolution of such issues.

GE identified this issue for COL applicant action in the "Safety Issues Index." The staff verified that GE established a COL action item (Item 19-28) in SSAR Table 1.9-1 to address unresolved generic and TMI safety issues. This approach is acceptable to the staff. The staff will review the COL applicant's proposed resolution of this issue on a case-by-case basis.

20.4.15 Issue I.C.5: Operating Procedures - Procedures for Feedback of Operating Experience to Plant Staff

Refer to the evaluation of 10 CFR 50.34(f)(3)(i) in Section 20.5.41 of this report.

20.4.16 Issue I.C.6: Operating Procedures - Procedures for Verification of Correct Performance of Operating Activities

This issue requires review and revision, as necessary, of procedures to assure that an effective system of verifying the correct performance of operating activities is provided as a means of reducing human errors and improving the quality of normal operations. Such a verification system may include automatic system status monitoring and human verification of operations and maintenance activities independent of the people performing the activity.

SSAR Section 19B.3.1 states that the COL applicant is responsible for providing resolutions of issues identified as "COL Applicant" in the "Safety Issues Index" of SSAR Appendix 19B. The SSAR states that issues are identified for COL action because they pertain to operating personnel issues, operating procedures, and other topics beyond the scope of the ABWR design certification review. SSAR Section 19B.1.1 lists specific documentation the COL applicant is to provide for resolution of such issues.

GE identified this issue for COL applicant action in the "Safety Issues Index." The staff verified that GE established a COL action item (Item 19-28) in SSAR Table 1.9-1 to address unresolved generic and TMI safety issues. This approach is acceptable to the staff. The staff will review the COL applicant's proposed resolution of this issue on a case-by-case basis.

20.4.17 Issue I.C.7: Operating Procedures - NSSS Vendor Review of Procedures

This issue requires that applicants for near-term operating licenses (NTOLs) obtain the NSSS vendor's review of the low-power and power-ascension test and emergency procedures to further verify the adequacy of the procedures.

SSAR Section 19B.3.1 states that the COL applicant is responsible for providing resolutions of issues identified as "COL Applicant" in the "Safety Issues Index" of SSAR Appendix 19B. The SSAR states that issues are identified for COL action because they pertain to operating personnel issues, operating procedures, and other topics beyond the scope of the ABWR design certification review. SSAR Section 19B.1.1 lists specific documentation the COL applicant is to provide for resolution of such issues.

GE identified this issue for COL applicant action in the "Safety Issues Index." The staff verified that GE established a COL action item (Item 19-28) in SSAR Table 1.9-1 to address unresolved generic and TMI safety issues. This approach is acceptable to the staff. The staff will review the COL applicant's proposed resolution of this issue on a case-by-case basis.

20.4.18 Issue I.C.8: Operating Procedures - Pilot Monitoring of Selected Emergency Procedures for NTOL Applicants

This issue requires an interdisciplinary and interoffice NRC task force review of emergency procedures received from NTOL applicants and of the training related to the symptoms of the postulated transients.

SSAR Section 19B.3.1 states that the COL applicant is responsible for providing resolutions of issues identified as "COL Applicant" in the "Safety Issues Index" of SSAR Appendix 19B. The SSAR indicates that issues are identified for COL action because they pertain to operating personnel issues, operating procedures, and other topics beyond the scope of the ABWR design certification review. SSAR Section 19B.1.1 lists specific documentation the COL applicant is to provide for resolution of such issues.

GE identified this issue for COL applicant action in the "Safety Issues Index." The staff verified that GE established a COL action item (Item 19-28) in SSAR Table 1.9-1 to address unresolved generic and TMI safety issues. This approach is acceptable to the staff. However, the COL applicant's responsibility for resolution of this issue extends only to providing the necessary procedures to the task force, accommodating the task force review, and addressing the review findings to the satisfaction of the NRC.

20.4.19 Issue I.C.9: Operating Procedures - Long-Term Program Plan Procedures for Upgrading of Procedures

Refer to the evaluation of 10 CFR 50.34(f)(2)(ii) in Section 20.5.14 of this report.

20.4.20 Issue I.D.1: Control Room Design - Control Room Design Reviews

Refer to the evaluation of 10 CFR 50.34(f)(2)(iii) in Section 20.5.15 of this report.

20.4.21 Issue I.D.2: Control Room Design - Plant Safety Parameter Display Console

Refer to the evaluation of 10 CFR 50.34(f)(2)(iv) in Section 20.5.16 of this report.

20.4.22 Issue I.D.3: Control Room Design - Safety System Status Monitoring

Refer to the evaluation of 10 CFR 50.34(f)(2)(v) in Section 20.5.17 of this report.

20.4.23 Issue I.D.5(2): Control Room Design - Plant Status and Post-Accident Monitoring

The objective of this issue is to improve the ability of nuclear power plant control room operators to prevent, diagnose, and properly respond to accidents and concentrates on the operator's information needs.

The review criteria for this issue are contained in RG 1.97 (Rev. 3). This document provides guidance for the design of instrumentation to help the operators (1) to determine the nature of an accident and whether the reactor trip and engineered safety features are functioning properly, (2) to provide information regarding the potential for breaching the barriers to radioactivity release, and (3) furnish data for deciding on the need to take manual action if an engineered safety feature malfunctions.

Generic Issues

SSAR Section 19B.2.65 states that the ABWR information system provides information for manual initiation and control of safety systems. These systems provide information sufficient for the operators to take an appropriate action when needed. Section 19B.2.65 refers to SSAR Section 7.5, "Information Systems Important to Safety," which describes safety-related display systems that provide information for the safe operation of the plant during normal operation, anticipated operational occurrences, and accidents. SSAR Section 7.5.2.1, "Post Accident Monitoring System," describes Type A Variables, which are plant-specific parameters needed to alert the control room operators to take actions manually, initiating a system or function that otherwise would not be automatically initiated in the course of an event. In conformance with the guidelines provided in RG 1.97, SSAR Table 7.5-2 lists post-accident monitoring variables which are common to BWR designs. The staff concludes that the features discussed above adequately address this issue for the ABWR design.

20.4.24 Issue I.D.5(3): Control Room Design - On-Line Reactor Surveillance System

This issue addresses noise surveillance and diagnostic techniques associated with the on-line reactor surveillance system. More specifically, it focuses on neutron noise monitoring in BWRs to detect the impact of instrument tubes against fuel channel boxes or detect other loose internal reactor parts.

The review criteria for this issue are addressed in RG 1.133, "Loose Parts Detection Program for the Primary System of Light-Water Cooled Reactor." SSAR Section 4.4.4, "Loose-Parts Monitoring System," describes the ABWR design features to provide detection of loose metallic parts within the RPV. The loose-parts monitoring system (LPMS) is designed to provide detection and operator warning of loose parts in the RPV to avoid or mitigate damage to or malfunction of reactor components. Additional design considerations provide for the inclusion of electronic features to minimize operator interfacing requirements during normal LPMS operation. These electronic features improve the LPMS capability when operator action is required. GE provided a general description of the LPMS, including the design bases, system description, system operation, safety evaluation, test, inspection, and application. The LPMS includes sensors (accelerometers located at neutral loose parts collection regions, e.g., steam outlet nozzle, feedwater inlet nozzle, control rod drive housings), signal conditioning, signal analysis, alarms, and calibration. The staff reviewed the LPMS description and concluded in Section 4.4.4.2 of this report that it conforms with RG 1.133

and, therefore, concludes that GE adequately addressed this issue for the ABWR design.

20.4.25 Issue I.F.1: Quality Assurance (QA) - Expand QA List

Refer to the evaluation of 10 CFR 50.34(f)(3)(ii) in Section 20.5.42 of this report.

20.4.26 Issue I.F.2: Quality Assurance - Develop More Detailed QA Criteria

Refer to the evaluation of 10 CFR 50.34(f)(3)(iii) in Section 20.5.43 of this report.

20.4.27 Issue I.G.1: Preoperational and Low-Power Testing - Training Requirements

The objective of this issue is to increase the capability of shift crews to operate facilities in a safe and competent manner by assuring that training for plant changes and off-normal events is conducted.

The review criterion for this issue is the definition of training plans prior to fuel loading and the conduct of training prior to full-power operation for each operating shift. The resolution of this issue is beyond the scope of the ABWR design certification review, and the COL applicant will be responsible for addressing it. SSAR Section 13.2 discusses training requirements for reactor operators. The staff verified that GE established a COL action item in SSAR Section 13.2.3.2 to include training requirements for preoperational and low-power testing activities. This approach is acceptable to the staff as discussed in Section 13.2 of this report.

20.4.28 Issue I.G.2: Preoperational and Low-Power Testing - Scope of Test Program

The objective of this issue is to review the comprehensiveness of test programs to identify anomalies in a plant's response to transients.

The review criteria for resolution of this issue are specified in SRP Chapter 14, "Initial Test Program - Final Safety Analysis Report," and RG 1.68, "Initial Test Programs for Water-Cooled Reactor Power Plants." SRP Section 14.2 sets forth the acceptable test procedures to establish the degree of conformance with the applicable tests identified in RG 1.68. RG 1.68, in turn, describes the general scope and depth of initial test programs acceptable to the NRC staff for light-water cooled reactors. RG 1.68, Appendix A provides a representative listing of plant structures, systems, and components design features and

performance capability tests that should be demonstrated during the initial test program.

The staff reviewed the test program proposed by GE for the ABWR and concludes in Section 14.2 of this report that it conforms to SRP Chapter 14 and RG 1.68. Therefore, the staff concludes that GE adequately addressed this issue for the ABWR.

20.4.29 Issue II.B.1: Consideration of Degraded or Melted Cores in Safety Review - Reactor Coolant System Vents

Refer to the evaluation of 10 CFR 50.34(f)(2)(vi) in Section 20.5.18 of this report.

20.4.30 Issue II.B.2: Consideration of Degraded or Melted Cores in Safety Review - Plant Shielding to Provide Access to Vital Areas and Protect Safety Equipment for Post-Accident Operation

Refer to the evaluation of 10 CFR 50.34(f)(2)(vii) in Section 20.5.19 of this report.

20.4.31 Issue II.B.3: Consideration of Degraded or Melted Cores in Safety Review - Post-Accident Sampling

Refer to the evaluation of 10 CFR 50.34(f)(2)(viii) in Section 20.5.20 of this report.

20.4.32 Issue II.B.4: Consideration of Degraded or Melted Cores in Safety Review - Training for Mitigating Core Damage

This issue requires development and implementation of a training program to teach the use of installed equipment and systems to control or mitigate accidents in which the core is severely damaged.

SSAR Section 19B.3.1 indicates that the COL applicant is responsible for providing resolutions of issues identified as "COL Applicant" in the "Safety Issues Index" of SSAR Appendix 19B. The SSAR states that issues are identified for COL action because they pertain to operating personnel issues, operating procedures, and other topics which are beyond the scope of the ABWR design certification review. SSAR Section 19B.1.1 lists specific documentation the COL applicant is to provide for resolution of such issues.

GE identified this issue for COL applicant action in the "Safety Issues Index." The staff verified that GE established a COL action item (Item 19-28) in SSAR Table 1.9-1 to address unresolved generic and TMI safety

issues. This approach is acceptable to the staff. The staff will review the COL applicant's proposed resolution of this issue on a case-by-case basis.

20.4.33 Issue II.B.8: Consideration of Degraded or Melted Cores in Safety Review - Rulemaking Proceeding on Degraded Core Accidents

Refer to the evaluations of 10 CFR 50.34(f)(1)(i), 50.34(f)(2)(ix), 50.34(f)(3)(iv), and 50.34(f)(3)(v) in Sections 20.5.1, 20.5.21, 20.5.44, and 20.5.45, respectively, of this report.

20.4.34 Issue II.D.1: Reactor Coolant System Relief and Safety Valves - Testing Requirements

Refer to the evaluation of 10 CFR 50.34(f)(2)(x) in Section 20.5.22 of this report.

20.4.35 Issue II.D.3: Reactor Coolant System Relief and Safety Valves - Relief and Safety Valve Position Indication

Refer to the evaluation of 10 CFR 50.34(f)(2)(xi) in Section 20.5.23 of this report.

20.4.36 Issue II.E.1.1: System Design -- Auxiliary Feedwater System - Auxiliary Feedwater System Evaluation

This issue is not applicable to the ABWR as discussed in the evaluation of 10 CFR 50.34(f)(1)(ii) in Section 20.5.2 of this report.

20.4.37 Issue II.E.1.2: System Design -- Auxiliary Feedwater System - Auxiliary Feedwater System Automatic Initiation and Flow Indication

This issue is not applicable to the ABWR as discussed in the evaluation of 10 CFR 50.34(f)(2)(xii) in Section 20.5.24 of this report.

20.4.38 Issue II.E.1.3: System Design -- Auxiliary Feedwater System - Update the Standard Review Plan and Develop Regulatory Guidance

This issue requires the NRC to update SRP Section 10.4.9 and issue a regulatory guide on AFWs.

The staff determined that this issue is not applicable to the ABWR, since a BWR plant does not incorporate an AFWs. Therefore, this issue is not technically relevant to the ABWR design and does not need to be addressed.

Generic Issues

20.4.39 Issue II.E.3.1: System Design -- Decay Heat Removal - Reliability of Power Supplies for Natural Circulation

This issue is not applicable to the ABWR as discussed in the evaluation of 10 CFR 50.34(f)(2)(xiii) in Section 20.5.25 of this report.

20.4.40 Issue II.E.4.1: System Design -- Containment Design - Dedicated Penetrations

Refer to the evaluation of 10 CFR 50.34(f)(3)(vi) in Section 20.5.46 of this report.

20.4.41 Issue II.E.4.2: System Design -- Containment Design - Isolation Dependability

Refer to the evaluation of 10 CFR 50.34(f)(2)(xiv) in Section 20.5.26 of this report.

20.4.42 Issue II.E.4.4: System Design -- Containment Design - Purging

Refer to the evaluation of 10 CFR 50.34(f)(2)(xv) in Section 20.5.27 of this report.

20.4.43 Issue II.E.5.1: System Design -- Design Sensitivity of Babcock & Wilcox (B&W) Reactors - Design Evaluation

This issue is not applicable to the ABWR as discussed in the evaluation of 10 CFR 50.34(f)(2)(xvi) in Section 20.5.28 of this report.

20.4.44 Issue II.E.6.1: System Design -- In-Situ Testing of Valves - Test Adequacy Study

The objective of this issue is to establish the adequacy of current requirements for safety-related valve testing. It recommends a study that would result in recommendations for alternate means of verifying performance requirements. This issue was divided into four parts during its resolution: (1) pressure isolation valves (PIVs), (2) check valves, (3) reevaluation of thermal-overload protection provisions of RG 1.106, "Thermal Overload Protection for Electric Motors on Motor-Operated Valves," for MOVs, and (4) in-situ testing of MOVs.

Relative to in-situ testing of PIVs, SSAR Section 3.9.6 requires that all the PIVs listed in Table 3.9-9 be leak tested in accordance with the ABWR TS. This approach is an acceptable way of addressing this part of the issue.

Regarding in-situ testing of check valves, SSAR Section 3.9.6.2.1 indicates that the COL applicant is responsible for performing in-situ full-flow testing of check valves, in addition to the ASME Section XI inservice testing requirements. Advanced, nonintrusive techniques will be used to assess degradation and performance characteristics of the check valves. In addition, the COL applicant is to develop a program to establish the frequency and extent of disassembly and inspection of check valves. This approach is responsive to the applicable guidelines of SECY-90-016 regarding inservice testing of pumps and valves and is an acceptable means of addressing this part of the issue.

For reevaluation of MOV thermal-overload protection, SSAR Tables 1.8-20 and 1.8-22 state that the ABWR design complies with RG 1.106 and NUREG-1296, "Thermal Overload Protection for Electric Motors on Motor-Operated Valves - Generic Issue II.E.6.1," respectively. In addition, SSAR Section 3.9.6.2.2 states that the guidelines of GL 89-10 will be implemented. Since the staff determined that the guidelines of RG 1.106 adequately address thermal-overload protection and GL 89-10 addresses control switch settings, which include thermal overload, the staff concludes that GE's approach to this part of the issue is acceptable.

With respect to in-situ testing of MOVs, SSAR Section 3.9.6.2.2 discusses implementation of the guidelines of GL 89-10 in sufficient detail for the staff to conclude in Section 3.9.6.2 of this report, that the staff's positions in SECY-90-016 as an applicable regulation regarding inservice testing of pumps and valves are addressed and that GE took an acceptable approach to this part of the issue.

20.4.45 Reserved.

20.4.46 Issue II.F.1: Instrumentation and Controls - Additional Accident Monitoring Instrumentation

Refer to the evaluation of 10 CFR 50.34(f)(2)(xvii) in Section 20.5.29 of this report.

20.4.47 Issue II.F.2: Instrumentation and Controls - Identification of and Recovery from Conditions Leading to Inadequate Core Cooling

Refer to the evaluation of 10 CFR 50.34(f)(2)(xviii) in Section 20.5.30 of this report.

20.4.48 Issue II.F.3: Instrumentation and Controls - Instruments for Monitoring Accident Conditions

Refer to the evaluation of 10 CFR 50.34(f)(2)(xix) in Section 20.5.31 of this report.

20.4.49 Issue II.G.1: Electrical Power - Power Supplies for Pressurizer Relief Valves, Block Valves, and Level Indicators

This issue is not applicable to the ABWR as discussed in the evaluation of 10 CFR 50.34(f)(2)(xx) in Section 20.5.32 of this report.

20.4.50 Issue II.J.3.1: General Implications of TMI for Design and Construction Activities -- Management for Design and Construction - Organization and Staffing to Oversee Design and Construction

Refer to the evaluation of 10 CFR 50.34(f)(3)(vii) in Section 20.5.47 of this report.

20.4.51 Issue II.J.4.1: General Implications of TMI for Design and Construction Activities - Revise Deficiency Reporting Requirements

This issue requires the NRC to improve the event-reporting requirements of 10 CFR 50.55(e) and 10 CFR Part 21 to ensure that all reportable items are reported promptly and that information submitted is complete.

SSAR Section 19B.3.1 states that the COL applicant is responsible for providing resolutions of issues identified as "COL Applicant" in the "Safety Issues Index" of SSAR Appendix 19B. The SSAR states that issues are identified for COL action because they pertain to operating personnel issues, operating procedures, and other topics beyond the scope of the ABWR design certification review. SSAR Section 19B.1.1 lists specific documentation the COL applicant is to provide for resolution of such issues.

GE identified this issue for COL applicant action in the "Safety Issues Index." The staff verified that GE established a COL action item (Item 19-28) in SSAR Table 1.9-1 to address unresolved generic and TMI safety issues. This approach is acceptable to the staff. However, the COL applicant's responsibility for resolution of this issue extends only to complying with the current regulations. The staff will review the COL applicant's proposed resolution of this issue on a case-by-case basis, evaluating compliance with the versions of 10 CFR 50.55(e) and 10 CFR Part 21 current at the time of the review.

20.4.52 Issue II.K.1(5): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- IE Bulletins - Safety-Related Valve Position Description

The objective of this issue is to have plants (1) review all valve positions and positioning requirements and positive controls, along with all related test and maintenance procedures, to assure proper ESF functioning, if required, and (2) verify that AFW valves are in the open position.

The verification that AFW valves are in the open position is applicable to PWRs only, since a BWR design does not include an AFWs. Therefore, this portion of the issue it is not technically relevant to the ABWR design and does not need to be addressed.

The review of valve positions and positioning requirements and positive controls is beyond the scope of the ABWR design certification review, and the COL applicant will be responsible for addressing this portion of the TMI item. This was identified in the DFSE as COL Action Item 18.7.2.2-7. The staff verified that GE established a COL action item (Item 18.8.7) in SSAR Section 18.8 for an evaluation of the indication of local valve position. This approach adequately addresses DFSE COL Action Item 18.7.2.2-7 as discussed in Section 18.7.2.2 of this report.

20.4.53 Issue II.K.1(10): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- IE Bulletins - Review and Modify Procedures for Removing Safety-Related Systems from Service

Between April 1, 1979, and July 26, 1979, the former NRC Office of Inspection and Enforcement (IE) issued nine bulletins to various operating plant licensees. This issue requires compliance with the requirements of the IE Bulletins related to operability determination and criteria needed to be met before removing safety-related equipment from service.

The DFSE reported that the SSAR stated that the COL applicant will review all maintenance and test procedures during the preoperational test phase. It also stated that the COL applicant will ensure that the maintenance and test procedures require verification of operability of redundant safety-related systems before removing the safety system from service. The SSAR also stated that the COL applicant will verify the operability of safety-related systems after performing maintenance or tests as part of the test to restore a system to service. The staff concluded that these requirements satisfied this TMI item, but stated

Generic Issues

that compliance with the TMI item was DFSER COL Action Item 20.3.1-2.

The staff verified that GE established a COL action item in SSAR Section 1A.3.2 for reviewing and modifying, as required, the procedures for removing safety-related systems from and restoring them to service to assure that their operability status is known. (The staff will verify that the procedures satisfy these requirements, which correspond to Item 8 of IE Bulletin 79-08, while reviewing the preoperational testing.) The staff concludes that this approach adequately addresses DFSER COL Action Item 20.3.1-2 and GE adequately addressed this TMI item for the ABWR design.

20.4.54 Issue II.K.1(13): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- IE Bulletins - Proposed Technical Specification Changes Reflecting Implementation of All Bulletin Items

Between April 1, 1979, and July 26, 1979, IE issued nine bulletins to various operating plant licensees. This issue requires operating plants to propose TS changes reflecting implementation of all bulletin items, as required.

Since 1969, there has been a trend to include in TS not only those requirements derived from safety analyses and evaluations, but also many other Commission requirements governing the operation of nuclear power reactors. Therefore, to make TS more operator oriented and to focus on the more important requirements, in its interim "Commission Policy Statement on Technical Specifications Improvements for Nuclear Power Plants," (52 FR 3788 dated February 6, 1987), the Commission established criteria to determine which requirements should remain in TS and which requirements could be relocated to licensee-controlled documents. Based on these criteria and with industry input, the staff developed improved standard technical specifications (STS). The bulletin items covered by this TMI item were considered in the development of the improved STS.

Future TS, including those for advanced reactors such as the ABWR, are to be based on the improved STS. Consequently, this approach supersedes the need to specifically address the bulletin items covered by this issue for the ABWR. Section 16 of this report discusses the development and acceptability of the ABWR TS.

20.4.55 Issue II.K.1.(22): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- IE Bulletins - Describe Automatic and Manual Actions for Proper Functioning of Auxilliary Heat Removal Systems When Feedwater System Not Operable

Refer to the evaluation of 10 CFR 50.34(f)(2)(xxi) in Section 20.5.33 of this report.

20.4.56 Issue II.K.1(23): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- IE Bulletins - Describe Uses and Types of RV Level Indication for Automatic and Manual Initiation Safety Systems

Between April 1, 1979, and July 26, 1979, IE issued nine bulletins to various operating plant licensees. This issue requires the performance of systems reliability analyses and changes in EOPs and operator training to improve the capability of plants to mitigate the consequences of the SBLOCAs and loss-of-feedwater events.

The review criterion for this issue is that the reactor water level must be known to the operators under all normal and abnormal conditions. The instrumentation that serves this purpose must be functional under all conditions, and must provide the operators all the information necessary to assess the state of the plant and what corrective action to be taken when needed.

SSAR Section 1A.2.21, describes the instrumentation that give the operator the information necessary to assess plant status. It provides information for the following conditions:

- Shutdown water level range - used to monitor the reactor water level during shutdown conditions when the reactor is flooded for maintenance and head removal.
- Narrow water level range - RPV taps at the elevation above the main steam outlet nozzle and a tap at an elevation near the bottom of the dryer skirt. This range is used for the water-level control and indication inputs of the feedwater control system.

- Wide water level range - RPV taps at the elevation above the main steam outlet nozzle and taps at the elevation near the top of the active fuel. These instruments provide inputs to various safety systems and ESFs.
- Fuel zone, water level range - RPV taps at the elevation above the main steam outlet nozzle and taps just above reactor internal pump deck. These instruments provide input to water-level indication only.

The instrumentation described above will improve the capability of the plant to mitigate the consequences of the SBLOCAs and loss-of-feedwater events, which is the objective of this issue. Therefore, the staff concludes that GE adequately addressed the requirements of this TMI item for the ABWR design.

20.4.57 Issue II.K.2(9): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Commission Orders on B&W Plants - Analysis and Upgrading of Integrated Control System

This issue is not applicable to the ABWR as discussed in the evaluation of 10 CFR 50.34(f)(2)(xxii) in Section 20.5.34 of this report.

20.4.58 Issue II.K.2(10): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Commission Orders on B&W Plants - Hard-Wired Safety-Grade Anticipatory Reactor Trips

This issue is not applicable to the ABWR as discussed in the evaluation of 10 CFR 50.34(f)(2)(xxiii) in Section 20.5.35 of this report.

20.4.59 Issue II.K.2(16): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Commission Orders on B&W Plants - Impact of RCP Seal Damage Following SBLOCA With Loss of Offsite Power

This issue is not applicable to the ABWR as discussed in the evaluation of 10 CFR 50.34(f)(1)(iii) in Section 20.5.3 of this report.

20.4.60 Issue II.K.3(2): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Report on Overall Safety Effect of Power-Operated Relief Valve (PORV) Isolation

This issue is not applicable to the ABWR as discussed in the evaluation of 10 CFR 50.34(f)(1)(iv) in Section 20.5.4 of this report.

20.4.61 Issue II.K.3(3): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Report Safety and Relief Valve Failures Promptly and Challenges Annually

This issue requires all operating plants and operating license applicants to report safety and relief valve failures promptly and challenges annually.

In the DFSER, the staff reported that SSAR Section 1.9 committed the COL applicant to report the failures of safety and relief valves in the annual report to the NRC in accordance with the requirement of this TMI item and that this approach was acceptable. COL applicant compliance with the requirement was identified as DFSER COL Action Item 20.3.1-3. The staff verified that GE established a COL action item in SSAR Section 1A.3.4. This approach adequately addresses DFSER COL Action Item 20.3.1-3.

20.4.62 Issue II.K.3(11): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Control Use of PORV Supplied by Control Components, Inc., Until Further Review Complete

This issue requires all plants to justify the use of PORVs supplied by Control Components, Inc., that had failed during testing.

SSAR Chapter 15 demonstrates the ABWR's capability to respond to the full spectrum of line breaks and loss-of-feedwater accidents without loss of containment or significant core damage. SSAR Section 5.2 describes the

Generic Issues

overpressure protection provided by the SRVs performing an overpressure relief valve function, an overpressure safety valve function, or an ADS function. SSAR Section 19B.2.70 states that the SRV for the ABWR is not a PORV by Control Components, Inc. For the safety valve function, the ABWR will use a spring-loaded safety valve with a pneumatic cylinder or piston for power operation in the ADS and relief function. Further, SSAR Section 3.9.3.2.4.2 describes the qualification by type test of the SRVs to IEEE 344, "Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations," for operability during a dynamic event.

The staff concludes that since the ABWR design does not use Control Components, Inc. PORVs and that the safety valve for the ABWR design will be appropriately qualified, this issue is adequately addressed for the ABWR design.

20.4.63 Issue II.K.3(13): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Separation of HPCI and RCIC System Initiation Levels

Refer to the evaluation of 10 CFR 50.34(f)(1)(v) in Section 20.5.5 of this report.

20.4.64 Issue II.K.3(15): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Modify Break Detection Logic to Prevent Spurious Isolation of HPCI and RCIC Systems

The HPCI and RCIC systems use differential pressure sensors on elbow taps in the steam lines to their turbine drives to detect and isolate pipe breaks in the systems. In NUREG-0737, the staff stated that the pipe-break-detection circuitry has resulted in spurious isolation of the HPCI and RCIC systems because of the pressure spike that accompanies the actuation of the systems. This TMI item requires applicants to modify the pipe-break-detection circuitry so that pressure spikes resulting from HPCI and RCIC system initiation will not cause inadvertent system isolation.

SSAR Section 1A.2.23 states that the ABWR design will maintain the high-pressure inventory using the motor-driven HPCF system rather than the turbine-driven HPCI system. Therefore, this TMI item only applies to the turbine-driven RCIC system of the ABWR.

SSAR Section 1A.2.23 states that the ABWR high leak detection and isolation system processes the differential pressure signals that isolate the RCIC turbine. Spurious trips are avoided because the RCIC has a bypass startup system controlled by valves F037 and F045. Upon receiving RCIC start signals, bypass valve F045 opens to pressurize the line downstream and accelerate the turbine. The bypass line through F045 is small (diameter of 1 in.) and naturally limits the initial flow surge to prevent a differential pressure spike in the upstream pipe.

After approximately 5 to 10 seconds, steam supply valve F037 opens to admit full steam flow to the turbine. At this stage, the line downstream is already pressurized. This design feature will reduce the possibility that a pressure spike would occur during any phase of the normal startup process. In the DFSER, the staff concluded that the ABWR design adequately addresses the requirements of this TMI item. However, the staff stated that the COL applicant should test the RCIC bypass startup system during plant startup and designated this as DFSER [COL] Action Item 20.3.1-4. In the advance SER, the staff concluded that since GE had not yet included a COL action item in the SSAR addressing this test, DFSER COL Action Item 20.3.1-4 would remain open until GE had done so. In Amendment 34, GE provided revised SSAR Section 1A.2.23 and provided a new Section 1A.3.8 that establish a COL action item for the COL applicant to test the RCIC bypass startup system during plant startup. This approach adequately addresses DFSER COL Action Item 20.3.1-4.

20.4.65 Issue II.K.3(16): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Reduction of Challenges and Failures of Relief Valves; Feasibility Study and System Modification

Refer to the evaluation of 10 CFR 50.34(f)(1)(vi) in Section 20.5.6 of this report.

20.4.66 Issue II.K.3(17): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Report on Outage of ECC Systems; Licensee Report and Technical Specification Changes

This TMI item required all GE plants to review data on ECC system outages to determine if cumulative outage time limitations should be incorporated in TS. It also required submittal of a report detailing outage dates, lengths of outages, and causes of the outages for all ECCSs.

The DFSER reported that the STS permit several components of the ECCS to have substantial outage times (e.g., 72 hours for one diesel generator; 14 days for the HPCF system). The ABWR TS contain limits on allowable outage times for ECCSs and ECC components but do not specify cumulative outage time limitations for the ECCSs. This was identified in the DFSER as TS Item 20.3.1. The advance SER stated that cumulative outage times were not required to be in the ABWR TS, but would be implemented in plant administrative procedures as discussed in Chapter 16 of the SER. The DFSER, also reported that SSAR Section 1.9 established an action item for the COL applicant to report ECCS outages in annual summary reports to the NRC. The staff also reported in the DFSER that it would review compliance with this requirement during the COL review. This was identified in the DFSER as COL Action Item 20.3.1-5.

The staff verified that GE established a COL action item in SSAR Table 1.9-1 (Item 1.9) to prepare and submit annual reports on ECCS unavailability that also include information on outage dates, lengths, and causes; ECCSs or ECC components involved; and any corrective action taken. SSAR Section 1A.2.25 also states that operating license applicants will establish a plan to meet these reporting requirements. This is approach adequately addresses DFSER COL Action Item 20.3.1-5.

20.4.67 Issue II.K.3(18): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Modification of ADS Logic; Feasibility Study and Modification for Increased Diversity for Some Event Sequences

Refer to the evaluation of 10 CFR 50.34(f)(1)(vii) in Section 20.5.7 of this report.

20.4.68 Issue II.K.3(21): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Restart of Core Spray and LPCI Systems on Low Level; Design and Modification

Refer to the evaluation of 10 CFR 50.34(f)(1)(viii) in Section 20.5.8 of this report.

20.4.69 Issue II.K.3(22): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Automatic Switchover of RCIC System Suction; Verify Procedures and Modify Design

This TMI item required that until the automatic switchover of RCIC system suction from the condensate storage tank to the suppression pool when the condensate storage tank level is low was implemented in BWRs, licensees and applicants would need to verify that clear and cogent procedures existed for the manual switchover of the RCIC system suction.

The RCIC system in the ABWR design includes an automatic switchover feature to change the pump suction source from the RCIC condensate storage tank to the suppression pool. The safety-grade switchover will automatically occur when the RCIC system receives a low-level signal from the condensate storage tank or a high-level signal from the suppression pool. The staff concludes that since the ABWR design incorporates this automatic switchover, there is no need for verification of the manual switchover procedures and that GE's approach adequately addresses the requirements of this TMI item for the ABWR design.

20.4.70 Issue II.K.3(23): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Central Water Level Recording

Refer to the evaluation of 10 CFR 50.34(f)(2)(xxiv) in Section 20.5.36 of this report.

20.4.71 Issue II.K.3(24): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Confirm Adequacy of Space Cooling for HPCI and RCIC Systems

Refer to the evaluation of 10 CFR 50.34(f)(1)(ix) in Section 20.5.9 of this report.

Generic Issues

20.4.72 Issue II.K.3(25): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Effect of Loss of ac Power on Pump Seals

Refer to the evaluation of 10 CFR 50.34(f)(1)(iii) in Section 20.5.3 of this report.

20.4.73 Issue II.K.3(27): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Provide Common Reference Level for Vessel Level Instrumentation

This issue required all reactor vessel level instruments on GE plants to be referenced to the same point to avoid potential confusion of operators if different reference points were used for various reactor vessel water level instruments. It recommended the use of either the bottom of the vessel or the active fuel as reasonable common reference points.

The review criterion for this issue is to confirm that the ABWR design has a common zero reference for all water level indications. SSAR Section 19B.2.71 states a common reference for the reactor vessel water level has been set at the top of the active fuel level as described in SSAR Section 7.7. The staff confirmed that SSAR Section 7.7.1.1 (6)(c) indicates that the zero of the reactor vessel water level instruments has been set at the top of the active fuel and the instruments are calibrated to be accurate at the normal power operating point. Therefore, the staff concludes that the ABWR design adequately addresses the requirements of this TMI item.

20.4.74 Issue II.K.3(28): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Study and Verify Qualification of Accumulators on ADS Valves

Refer to the evaluation of 10 CFR 50.34(f)(1)(x) in Section 20.5.10 of this report.

20.4.75 Issue II.K.3(30): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Revised SBLOCA Methods to Show Compliance with 10 CFR Part 50, Appendix A

This issue requires all licensees and applicants to revise, document, and submit for NRC approval the analyses used by NSSS vendors and/or fuel suppliers for SBLOCA analysis in compliance with 10 CFR Part 50, Appendix K. The revised analyses were to account for comparisons with experimental data, including data from the loss-of-fluid test (LOFT) and Semiscale test facilities.

In response to this issue, the BWROG conducted a study that was later endorsed by GE as being applicable to the ABWR within this area. In a letter from R.H. Buchholz (GE) to D.G. Eisenhower (NRC) dated June 26, 1981, GE submitted information on the results of its study and NRC's concerns with the small-break model. The GE information consisted of modeling in SAFE, treatment of pressure variation, and overall model assessment. The staff reviewed this information and concluded that it was acceptable. The staff concluded that the SBLOCA model need not be changed because the test data comparisons and other information submitted by GE acceptably demonstrate that its small-break model complies with the analysis requirements in 10 CFR Part 50, Appendix K.

The staff also reviewed the applicability of the BWROG evaluation to the ABWR design. It concurs that no model changes are required for the ABWR, because it is similar in design to the current BWRs and thus will respond similarly to SBLOCAs. Therefore, the staff concludes that the requirements of this TMI item have been adequately addressed for the ABWR SBLOCA model.

20.4.76 Issue II.K.3(31): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Plant-Specific Calculations to Show Compliance with 10 CFR 50.46

This issue requires licensees and applicants to submit for NRC approval plant-specific calculations, using NRC-approved models for SBLOCAs, to show compliance with 10 CFR 50.46.

The ABWR-specific SBLOCA calculations in SSAR Section 6.3 show compliance with 10 CFR 50.46, as discussed in Section 6.3 of this report. Therefore, the staff concludes that GE has adequately addressed the requirements of this TMI item for the ABWR design.

20.4.77 Issue II.K.3(44): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Evaluation of Anticipated Transients with Single Failure to Verify no Significant Fuel Failure

This issue requires licensees and applicants of GE plants to demonstrate that the core remains covered or provide analysis to show that no significant fuel damage results from core uncover for anticipated transients combined with the worst single failure and proper operator actions. This category includes transients that result from a stuck-open relief valve.

GE has endorsed the results of the BWROG study (NEDO-24708, "Additional Information Required for NRC Staff Generic Report on Boiling Water Reactors," which had been accepted by the staff) as applicable to ABWR in this area. In a letter dated December 29, 1980, from D.B. Walters (BWROG) to NRC, the BWROG enclosed an evaluation (NEDO-24708) in which it stated that the worst-case transient with single failure combination for BWR/2-6 plants is the loss-of-feedwater event with a failure of the high-pressure ECCS. However, since the ABWR design includes three high-pressure core injection systems, the probability of a loss of all high-pressure ECCS is low. GE also considered an event with a stuck-open relief valve, and a high-pressure ECCS failure, and concluded that the core remained covered throughout the transient either because the RCIC system operated or because the RCS was depressurized by automatic or manual means, permitting low-pressure inventory makeup. GE also assumed the operator would manually depressurize the vessel to permit low-pressure injection.

The staff has reviewed the results of the BWROG's study and its applicability to ABWR and finds that GE has shown that the ABWR design can keep the core covered and have no fuel damage from core uncover for transients combined with the worst single failure. Therefore, GE adequately addressed the requirements of this TMI item for the ABWR design.

20.4.78 Issue II.K.3(45): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Evaluate Depressurization with Other Than Full ADS

Refer to the evaluation of 10 CFR 50.34(f)(1)(xi) in Section 20.5.11 of this report.

20.4.79 Issue II.K.3(46): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Response to List of Concerns from ACRS Consultant

This TMI item includes 16 questions, developed by Mr. C. Michaelson of the ACRS staff, most of which pertain to PWRs, to which all licensees and applicants were required to respond. GE responded to Mr. Michaelson's concerns as they related to BWRs in a letter dated February 21, 1980.

The staff required GE to review each of Mr. Michaelson's questions and verify that the responses given for BWRs in its February 21, 1980, letter were valid for the ABWR design. In SSAR Table 1A-1, GE responds to all 16 questions. One question pertains to the adequacy of the net positive suction head (NPSH) since the ABWR RCIC and HPCF systems share a common suction line from the condensate storage tank. It is an ABWR design requirement that adequate NPSH be available to the RCIC and HPCF pumps for simultaneous operating modes of these systems. Other questions pertain to the isolation of small breaks, the adequacy of auxiliary feedwater, the recirculation mode of HPCI pumps at high pressure, and the simultaneous operation of HPCI and RHR pumps.

The staff reviewed the responses in SSAR Table 1A-1 and found them to be similar to GE's responses to Mr. Michaelson's questions for operating BWRs that it previously accepted. This approach is acceptable for the ABWR design. Therefore, the staff concludes that GE adequately addressed the requirements of this TMI item for the ABWR design.

20.4.80 Issue III.A.1.1(1): Emergency Preparedness and Radiation Effects -- Improve Licensee Emergency Preparedness - Short Term; Upgrade Emergency Preparedness, Implement Action Plan Requirements for Promptly Improving Licensee Emergency Preparedness

This issue requires approval of the overall state of preparedness, primarily with respect to the capability of

Generic Issues

offsite agencies to take appropriate emergency actions in the event of nuclear power plant accidents.)

SSAR Section 19B.3.1 states that the COL applicant is responsible for providing resolutions of issues identified as "COL Applicant" in the "Safety Issues Index" of SSAR Appendix 19B. The SSAR states that issues are identified for COL action because they pertain to operating personnel issues, operating procedures, and other topics beyond the scope of the ABWR design certification review. SSAR Section 19B.1.1 lists specific documentation the COL applicant is to provide for resolution of such issues.

GE identified this issue for COL applicant action in the "Safety Issues Index." The staff verified that GE established a COL action item (Item 19-28) in SSAR Table 1.9-1 to address unresolved generic and TMI safety issues. This approach is acceptable to the staff. However, the COL applicant's responsibility for resolution of this issue extends only to providing support to the review and approval process and addressing the review findings to the satisfaction of the NRC.

20.4.81 Issue III.A.1.2: Emergency Preparedness and Radiation Effects -- Improve Licensee Emergency Preparedness - Short Term; Upgrade Emergency Preparedness

Refer to the evaluation of 10 CFR 50.34(f)(2)(xxv) in Section 20.5.37 of this report.

20.4.82 Issue III.A.2.1: Emergency Preparedness and Radiation Effects -- Improve Licensee Emergency Preparedness - Long Term; Amend 10 CFR Part 50 and 10 CFR Part 50, Appendix E

This issue required the NRC to revise 10 CFR Part 50, as appropriate, to upgrade the emergency preparedness of nuclear power plants and to revise the inspection program to cover the upgraded requirements.

SSAR Section 19B.3.1 states that the COL applicant is responsible for providing resolutions of issues identified as "COL Applicant" in the "Safety Issues Index" of SSAR Appendix 19B. The SSAR states that issues are identified for COL action because they pertain to operating personnel issues, operating procedures, and other topics beyond the scope of the ABWR design certification review. SSAR Section 19B.1.1 lists specific documentation the COL applicant is to provide for resolution of such issues.

GE identified this issue for COL applicant action in the "Safety Issues Index." The staff verified that GE established a COL action item (Item 19-28) in SSAR

Table 1.9-1 to address unresolved generic and TMI safety issues. This approach is acceptable to the staff. However, the COL applicant's responsibility for resolution of this issue extends only to complying with the current regulations. The staff will review the COL applicant's proposed resolution of this issue on a case-by-case basis, evaluating compliance with the version of the emergency preparedness requirements of 10 CFR Part 50, as well as 10 CFR Part 50, Appendix E, current at the time of the review.

20.4.83 Issue III.A.2.2: Emergency Preparedness and Radiation Effects -- Improve Licensee Emergency Preparedness - Long Term; Development of Guidance and Criteria

This issue requires the emergency plans to include information on meteorological criteria, means for promptly notifying the population, and emergency response facilities as detailed in Revision 1 to NUREG-0654 (FEMA-REP-1), "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants."

SSAR Section 19B.3.1 states that the COL applicant is responsible for providing resolutions of issues identified as "COL Applicant" in the "Safety Issues Index" of SSAR Appendix 19B. The SSAR states that issues are identified for COL action because they pertain to operating personnel issues, operating procedures, and other topics beyond the scope of the ABWR design certification review. SSAR Section 19B.1.1 lists specific documentation the COL applicant is to provide for resolution of such issues.

GE identified this issue for COL applicant action in the "Safety Issues Index." The staff verified that GE established a COL action item (Item 19-28) in SSAR Table 1.9-1 to address unresolved generic and TMI safety issues. This approach is acceptable to the staff. The staff will review the COL applicant's proposed resolution of this issue on a case-by-case basis.

20.4.84 Issue III.A.3.3: Emergency Preparedness and Radiation Effects -- Improving NRC Emergency Preparedness - Communications

This issue requires the availability of communication means that will enable the NRC, in the event of a nuclear accident, to (1) monitor and evaluate the situation and (2) potentially advise the plant operating staff, as needed, and (3) in extreme cases, be able to issue orders governing such operations.

SSAR Section 19B.3.1 states that the COL applicant is responsible for providing resolutions of issues identified as

"COL Applicant" in the "Safety Issues Index" of SSAR Appendix 19B. The SSAR states that issues are identified for COL action because they pertain to operating personnel issues, operating procedures, and other topics beyond the scope of the ABWR design certification review. SSAR Section 19B.1.1 lists specific documentation the COL applicant is to provide for resolution of such issues.

GE identified this issue for COL applicant action in the "Safety Issues Index." The staff verified that GE established a COL action item (Item 19-28) in SSAR Table 1.9-1 to address unresolved generic and TMI safety issues. This approach is acceptable to the staff. The staff will review the COL applicant's proposed resolution of this issue on a case-by-case basis.

20.4.85 Issue III.D.1.1: Radiation Protection -- Radiation Source Control - Primary Coolant Sources Outside the Containment Structure

Refer to the evaluation of 10 CFR 50.34(f)(2)(xxvi) in Section 20.5.38 of this report.

20.4.86 Issue III.D.3.3: Radiation Protection -- Worker Radiation Protection Improvement - Inplant Radiation Monitoring

Refer to the evaluation of 10 CFR 50.34(f)(2)(xxvii) in Section 20.5.39 of this report.

20.4.87 Issue III.D.3.4: Radiation Protection -- Worker Radiation Protection Improvement - Control Room Habitability

Refer to the evaluation of 10 CFR 50.34(f)(2)(xxviii) in Section 20.5.40 of this report.

20.5 10 CFR 50.34(f), Additional TMI Requirements

This section addresses staff evaluation of paragraphs (1)(i) through (3)(vii) of 10 CFR 50.34(f).

20.5.1 10 CFR 50.34(f)(1)(i): Consideration of Degraded or Melted Cores in Safety Review - Rulemaking Proceeding on Degraded Core Accidents (TMI Item II.B.8), "Design Alternatives from PRA"

Paragraph (1)(i) of 10 CFR 50.34(f) requires the applicant to "perform a plant/site specific probabilistic risk assessment, the aim of which is to seek such improvements in the reliability of core and containment heat removal systems as are significant and practical and do not impact excessively on the plant."

20.5.1.1 Introduction

GE has made extensive use of the results of the PRA to arrive at a final ABWR design. As a result, the estimated core damage frequency and risk calculated for the ABWR is very low both relative to operating plants and in absolute terms. The low core damage frequency and risk for the ABWR is a reflection of GE's efforts to systematically minimize the effect of initiators or sequences that have been important contributors to core damage frequency in previous BWR PRAs. This has been done largely through the incorporation of a number of hardware improvements in the ABWR design. These include the provision of: three separated divisions of ECCSs, a diverse and independent combustion gas turbine capable of providing ac power to any of the three divisions, an ac-independent water addition system, and an FMCRD system as a backup to the hydraulic drive system. Several improvements have also been incorporated in the ABWR design to mitigate the consequences of a core damage event, including inerting of the containment atmosphere, inclusion of a lower drywell flooder system and a containment overpressure protection (vent) system, the use of basaltic concrete in the lower drywell, and an increased ultimate pressure capacity. These and additional ABWR design features which contribute to low core damage frequency and risk for the ABWR are discussed further in FSER Section 19.1.

In response to 10 CFR 50.34(f)(1)(i), GE provided an initial evaluation of further ABWR design improvements in a February 25, 1992 submittal. This submittal was based on the original PRA results, and included consideration of risk from both internally and seismically initiated events. Based on this evaluation, GE concluded that none of the design improvements considered were cost beneficial.

The initial evaluation was subsequently revised to reflect the results of the updated Level 1 PRA and containment analyses, and was resubmitted on June 30, 1992. The revised assessment was based on the risk reduction potential for internal events only, in contrast to the original evaluation which considered both internally and seismically initiated events. This more limited scope was a consequence of GE's change in methodology from a quantitative treatment of seismic risk to a qualitative, margins-type analysis of seismic events. The net result of the new analysis was an order of magnitude reduction of estimated risk from severe accidents -- from 0.047 person-Sv to 0.0048 person-Sv (4.7 person-rem to 0.48 person-rem) over a 60-year plant life -- largely due to removal of seismic events from the risk profile. The reduced risk in the revised analysis further strengthened GE's original conclusion that none of the design improvements, beyond

Generic Issues

those already incorporated in the ABWR design, were cost beneficial.

In response to staff comments, GE further modified their evaluation of design improvements to include additional discussion of selected design alternatives, further clarification of the basis for risk-reduction estimates, and additional factors in estimating costs for the modifications. The risk estimates were also adjusted to reflect the results of the final Level 2 analysis and the updated offsite consequence calculations performed as part of the Level 3 portion of the PRA. The final evaluation was submitted as SSAR Appendix 19P.

GE's evaluation of potential design improvements was submitted in response to the requirements of 10 CFR 50.34(f)(1)(i). The staff's review of GE's final evaluation of potential ABWR design improvements is presented below.

20.5.1.2 Estimate of Risk for ABWR

20.5.1.2.1 GE Risk Estimates

GE estimated offsite consequences at five different sites, each representing a different geographic region of the U.S. Offsite consequences were calculated for each release class or case from the Level 2 analysis using the CRAC2 code. The meteorological and population data were obtained from previously developed information contained in NUREG/CR-2239, "Technical Guidance for Siting Criteria Development." The source terms were determined using the MAAP code for each of the release categories as discussed in Section 19.1 of the FSER. The results of the five sets of consequence calculations were averaged together, to represent a typical site in the U.S.

GE's estimate of the cumulative offsite risk to the population within 80 km (50 miles) of the site is provided in SSAR Table 19P-1. The total cumulative exposure calculated by GE is about 0.003 person-Sv (0.3 person-rem), assuming a 60-year plant life. The extremely small level of risk calculated by GE is primarily due to the low estimated core damage frequency for the ABWR ($1.6\text{E-}7$ per reactor year). As a case in point, even if all core damage accidents resulted in the worst release, based on GE's core damage frequency estimates for internal events, the total exposure would only be about 0.3 person-Sv (30 person-rem).

As a result of the low estimated core damage frequency and associated risk levels for the ABWR, any potential modifications which cost more than a few dollars would not be cost-effective, even if the design modification were

to totally eliminate the severe accidents or their consequences.

The staff notes that the frequencies of core damage accidents and release bins on which GE based its evaluation of design alternatives are slightly different than those reported in SSAR Section 19D.5. However, these differences are minor and would not alter the essential conclusions of the analysis.

20.5.1.2.2 Staff Review of GE's Risk Estimates

The staff independently estimated the risk associated with severe accidents in the ABWR. A comparison of GE and staff estimates of the person-Sv (person-rem) exposure for each of GE's release classes is provided in FSER Table 20.5.1-1 for internally initiated events. GE's estimates are based on the use of the MAAP and CRAC2 computer codes, and meteorology for five different sites, as described previously. The staff estimates of person-Sv (person-rem) are based on use of the 50th percentile source terms developed during the initial staff review of the ABWR (FSER Table 20.5.1-2), the MACCS offsite consequence code, and meteorology for the Zion site. The staff estimates of the frequency of occurrence of each release class are as reported in FSER Section 19.1.

The GE and staff estimates of person-Sv (person-rem) exposure per event are generally consistent for the large release classes (Cases 7, 8, and 9). The staff's dose estimate is significantly higher for vented scenarios (Case 1) due to the higher fission product release fractions used in the staff's assessment. Similarly, the staff's estimate is much lower for sequences with normal containment leakage (NCL) due to a significantly smaller staff source term for this case. The differences between staff and GE estimates for both these release classes are insignificant, however, since these release classes do not contribute appreciably to total risk.

The estimated total risk over a 60-year reactor operating lifetime is extremely small in both the GE and staff assessment. GE's analysis indicates a total dose of about 0.003 person-Sv (0.3 person-rem) over the 60-year period. The staff's estimate is about 0.01 person-Sv (1 person-rem). The difference is due largely to an increased frequency of early releases in the staff assessment to account for: (1) the contribution from unisolated LOCAs outside containment, and (2) an increased probability of early containment failure from direct containment heating. It can be noted that total risk is dominated by events which lead to early containment failure, and containment bypass. This is consistent with the results from PRAs for operating plants.

Table 20.5.1-1a Comparison of GE and staff adjusted offsite consequences (person-Sv)

Case*	GE Estimates ^{1/}				Staff-Adjusted Estimates			
	Frequency	Person-Sv	Person-Sv per 60 y	Fraction	Frequency ^{2/}	Person-Sv ^{3/}	Person-Sv per 60 y	Fraction
NCL	1.3E-7	96	.00075	.28	1.34E-7	.1	.000008	<.01
Case 1	2.1E-8	1.4E2	.00018	.07	2.08E-8	1.4E3	.0017	.13
Case 7	3.9E-10	2.7E4	.00063	.23	3.6E-10	1.6E4	.00035	.03
Case 8	4.1E-10	3.2E4	.00079	.29	3.6E-9 ^{4/}	4.5E4	.0097	.77
Case 9	1.7E-10	3.3E4	.00034	.13	3.3E-10	4.5E4 ^{5/}	.00089	.07
Total	1.6E-7		.00269	1.0	1.6E-7		.0126	1.0

Table 20.5.1-1b Comparison of GE and staff adjusted offsite consequences (person-rem)

Case*	GE Estimates ^{1/}				Staff-Adjusted Estimates			
	Frequency	Person-Rem	Person-Rem per 60 y	Fraction	Frequency ^{2/}	Person-Rem ^{3/}	Person-Rem per 60 y	Fraction
NCL	1.3E-7	9600	.075	.28	1.34E-7	100	.0008	<.01
Case 1	2.1E-8	1.4E4	.018	.07	2.08E-8	1.4E5	.17	.13
Case 7	3.9E-10	2.7E6	.063	.23	3.6E-10	1.6E6	.035	.03
Case 8	4.1E-10	3.2E6	.079	.29	3.6E-9 ^{4/}	4.5E6	.97	.77
Case 9	1.7E-10	3.3E6	.034	.13	3.3E-10	4.5E6 ^{5/}	.089	.07
Total	1.6E-7		0.269	1.0	1.6E-7		1.26	1.0

Notes:

* For case description, refer to Table 20.5.1-2.

^{1/} Based on information reported in SSAR Table 19P-1.

^{2/} Based on GE's containment event tree end state frequencies with staff corrections.

^{3/} Based on staff 50th percentile source terms (see Table 20.5.1-2) and use of MACCS consequence code with Zion site meteorology.

^{4/} Staff frequency estimate includes: (1) contribution from unisolated LOCAs outside containment, and (2) increased probability of early containment failure from direct containment heating.

^{5/} Based on the staff's source term estimate for Case 8.

Generic Issues

Table 20.5.1-2 Cesium and Iodine release fractions, as estimated by the staff and GE

GE Case		<u>Staff's Estimate</u>			<u>GE's Estimate</u>
	<u>Identifier</u>	<u>5th</u>	<u>50th</u>	<u>95th</u>	
Cesium	NCL	2.1×10^{-11}	3.2×10^{-9}	3.9×10^{-8}	5.1×10^{-5}
	Case 1	2.7×10^{-6}	2.7×10^{-4}	5.3×10^{-2}	1.3×10^{-5}
	Case 7	6.6×10^{-6}	2.4×10^{-3}	1.4×10^{-1}	9.9×10^{-2}
	Case 8	0.002	0.06	0.75	0.25
	Case 9	8.3×10^{-4}	7.8×10^{-3}	1.7×10^{-1}	0.36
Iodine	NCL	3.4×10^{-11}	2.3×10^{-6}	3.8×10^{-4}	3.8×10^{-6}
	Case 1	8.5×10^{-6}	9.2×10^{-4}	6.1×10^{-2}	1.5×10^{-7}
	Case 7	4.7×10^{-4}	9.7×10^{-2}	3.2×10^{-1}	8.9×10^{-2}
	Case 8	0.007	0.19	0.69	0.19
	Case 9	0.002	0.10	0.16	0.37

Case	Description
NCL	Normal containment leakage, no containment failure.
Case 1	Fission products scrubbed by suppression pool before release, includes the "venting" sequences.
Case 7	Late containment failure due to overpressurization, spray available, no suppression pool scrubbing.
Case 8	Early containment failure, no suppression pool scrubbing.
Case 9	Late containment failure due to overpressurization, no spray, no suppression pool scrubbing.

As discussed below, the staff based its assessment of risk-reduction potential for the various ABWR design improvements on the staff estimate of risk for internally initiated events, that is, 0.01 person-Sv (1 person-rem). However, the validity of the conclusions of this analysis were tested by considering the uncertainties in core damage frequency estimates, as well as the potential frequency of core damage due to external events.

20.5.1.3 Identification of Potential Design Improvements

20.5.1.3.1 List of Potential Design Improvements

GE identified a set of potential design improvements for the ABWR based on a survey of previous industry- and NRC-sponsored studies of preventative and mitigative features which address severe accidents. Through this effort GE developed a composite list of 68 potential design improvements, organized into 14 general categories. These categories and many of the design improvements are the same as considered for the General Electric Standard Safety Application Report (GESSAR) II design. The resulting list of potential design improvements for the ABWR is presented in SSAR Table 19P-3.

GE eliminated certain design improvements from further consideration on the basis that they are either already incorporated into the ABWR design, or not applicable to the ABWR design. Examples of design improvements already included in the design are: improved low-pressure injection system (fire pump), RWCU decay heat removal, low flow unfiltered vent, and combustible gas control (inerted containment). On the basis of this screening, 21 potential design improvements covering 12 of the 14 general categories were retained for further consideration. The set of design improvements selected for further evaluation is listed in Table 20.5.1-3, and summarized in Section 20.5.1.3.2 of this report.

The staff and its contractor, Brookhaven National Laboratory (BNL), have reviewed the set of potential design improvements identified by GE (SSAR Table 19P-3) and find it to be comprehensive. The list includes all improvements identified as part of the GESSAR II review, and the NRC Containment Performance Improvement (CPI) program. The staff notes that the set of design improvements is not all-inclusive, in that additional, perhaps less expensive design improvements can always be postulated. However, the staff concludes that the benefits offered by any additional modifications would not likely exceed those for the modifications evaluated. The staff also concludes that the costs of alternative improvements would not likely be less than that of the lowest cost improvements evaluated, when

the subsidiary costs associated with maintenance, procedures, and training are considered. On this basis, the staff concludes that the set of potential design improvements identified by GE is acceptable.

The set of design improvements selected for further evaluation also appears to be reasonable. The staff notes that the improvements considered include a filtered containment vent, and flooded rubble bed core retention device, which are two improvements specifically cited in NUREG-0660 for evaluation as part of TMI Item II.B.8. A modification intended to delay the time of reactor vessel failure through the use of alternative materials for the bottom head penetration piping was also considered by GE. This modification was instigated by the results of recent analyses of reactor vessel bottom head failure as documented in draft NUREG/CR-5642.

Finally, it should be noted that certain features of several of the improvements selected for further evaluation have been or will be incorporated as part of the ABWR design, independent of this evaluation of design improvements. For example, severe accident EPGs or accident management guidelines (AMGs) will be implemented by the COL applicant as part of its accident management program, as discussed in FSER Section 19.2, and much of the benefits of improved maintenance procedures or manuals will be achieved through the COL applicant's reliability assurance program, as discussed in Section 19.1.3.7 of this report.

20.5.1.3.2 Description of Design Improvements

A description of the design improvements selected by GE for cost-benefit evaluation is provided in SSAR Sections 19P.3 and 19P.4, and summarized below.

- Severe accident EPGs or AMGs - extend the EPGs and EOPs to address arrest of a core melt, emergency planning, radiological release assessment and other areas related to severe accidents. This modification would lead to increased reliability of manual actions in response to core-damage events.
- Computer-aided instrumentation - provide artificial intelligence-based improvements to plant status monitoring, including human-engineered displays of important variables in the EPGs and AMGs, and procedural options for the operator to evaluate during severe accidents. This modification would lead to increased reliability of manual actions to prevent core damage.

Generic Issues

Table 20.5.1-3 Potential Design Improvements and Associated Costs (Provided by GE)

Modification		Estimated Cost (\$M)	Person-Sv (Person-Rem) Averted		Cost(\$M)/Person-Sv (Person-Rem) Averted	
1.	Accident Management					
1a.	Severe accident EPGs	0.60	0.00015	(0.015)	4,000	(40)
1b.	Computer-aided instrumentation	0.60	0.00010	(0.01)	>4,000	(>40)
1c.	Improved maintenance procedures/manuals	0.30	0.00016	(0.016)	1,880	(18.8)
2.	Decay Heat Removal					
2a.	Passive high pressure system	1.75	0.00138	(0.138)	1,270	(12.7)
2b.	Improved depressurization	0.60	0.00042	(0.042)	1,430	(14.3)
2c.	Suppression pool jockey pump	0.12	0.00002	(0.002)	>4,000	(>40)
2d.	Safety-related condensate storage tank	1.0	0.00010	(0.01)	>4,000	(>40)
3.	Containment Capability					
3a.	Larger volume containment	8.0	0.00150	(0.15)	>4,000	(>40)
3b.	Increased containment pressure capacity	12.0	0.00020	(0.02)	>4,000	(>40)
3c.	Improved vacuum breakers	0.10	0.0000003	(0.00003)	>4,000	(>40)
3d.	Improved bottom head penetration design	0.75	0.00057	(0.057)	1,320	(13.2)
4.	Containment Heat Removal					
4a.	Larger volume suppression pool	8.0	0.000002	(0.0002)	>4,000	(>40)
5.	Containment Atmosphere Mass Removal					
5.a	Low-flow filtered vent	3.0	0.00014	(0.014)	>4,000	(>40)
7.	Containment Spray Systems					
7a.	Drywell head flooding	0.10	0.00060	(0.06)	1,700	(1.7)
8.	Prevention Concepts					
8a.	Additional service water loop	6.0	0.00016	(0.016)	>4,000	(>40)
9.	AC Power Supplies					
9a.	Steam driven turbine generator	6.0	0.00052	(0.052)	>4,000	(>40)
9b.	Alternate pump power source	1.2	0.00069	(0.069)	1,740	(17.4)
10.	DC Power Supplies					
10a.	Dedicated RHR dc Power Supply	3.0	0.00069	(0.069)	>4,000	(>40)
11.	ATWS Capability					
11a.	ATWS-sized vent	0.30	0.00030	(0.03)	1,000	(10)
13.	System Simplification					
13a.	Reactor building sprays	0.10	0.00017	(0.017)	5,900	(5.9)
14.	Core Retention Devices					
14a.	Flooded rubble bed	18.8	0.00001	(0.001)	>4,000	(>40)

- Improved maintenance procedures or manuals - provide improved maintenance manuals and additional information on the components important to the risk of the plant within the GE scope of supply. These manuals and information would lead to increased reliability of important equipment.
- Passive high pressure system - add an isolation condenser-type high pressure system for removing decay heat from both the core and containment. The benefit of this system would be equivalent to an additional RCIC system and containment heat-removal system.
- Improved depressurization - provide manually controlled, seismically protected air operators to permit manual depressurization in the event of loss of dc control power or control air events. Improved depressurization would reduce the threat of containment failure due to high pressure melt ejection, and allow more reliable access to low-pressure systems.
- Suppression pool jockey pump - add a small, ac-independent makeup pump to provide low-pressure decay heat removal from the reactor pressure vessel using suppression pool water as the source. This modification would have a benefit similar to that provided by the ac-independent water addition mode of RHR (fire water), but without the associated long-term containment inventory concerns.
- Safety-related condensate storage tank - upgrade the structure of the condensate storage tank such that it would be available to provide makeup to the reactor following a large seismic event. This would enhance core injection capabilities in seismic events, by providing an alternative to the suppression pool as a source of water for injection.
- Larger volume containment - increase the volume of containment by a factor of two. This would reduce the peak pressures associated with energetic events, thereby reducing the potential for drywell head failure, and would also reduce the rate of long-term containment pressurization, thereby delaying the time of fission product release.
- Increased containment pressure capacity - increase the ultimate pressure capacity of containment (including seals) to a level at which all release modes except normal containment leakage are eliminated.
- Improved vacuum breakers - add a second vacuum breaker valve in each of the eight drywell-to-wetwell vacuum breaker lines to make these valves redundant. This modification would reduce the potential for suppression pool bypass due to stuck-open or leaking vacuum breaker valves.
- Improved bottom head penetration design - change the transition piece (used to connect the stainless steel RPV drain line to the RPV) from carbon steel to a material with a higher melting point, such as inconel. Also, establish external welds or restraints on the CRDs external to the vessel so that the drives would not be ejected following failure of the internal welds. This modification would delay the time of reactor vessel failure by several hours, thereby increasing the potential to arrest core damage in the vessel, but may also increase the potential for gross failure of the lower head.
- Larger volume suppression pool - increase the size of the suppression pool to provide reduced pool heatup rates. This modification would reduce the frequency of core melt from Class II sequences (loss of containment heat removal), and ATWS sequences by providing additional time for operator actions and recovery of heat removal systems.
- Low-flow filtered vent - add a filter system external to the containment to further reduce the magnitude of radioactive releases via containment venting. The system would be similar to the multi-venturi scrubbing systems implemented in some plants in Europe. The system would provide fission product scrubbing beyond that presently offered by the suppression pool, but would not affect releases due to drywell head failure and containment bypass sequences.
- Drywell head flooding - provide an additional line to permit intentional flooding of the upper drywell head using the existing fire water additional system. Drywell head flooding would cool the drywell head seal and provide fission product scrubbing in the event of drywell head leakage. Instrumentation and controls to permit manual control from the control room were considered part of this modification.
- Additional service water loop - provide an additional service water cooling loop (pump and heat exchanger) to improve the overall reliability of the service water network. This cooling loop would be capable of removing heat from any one of the three divisions. This would reduce the frequency of sequences involving failure of injection due to loss of component cooling.
- Steam-driven turbine generator - add a steam-driven turbine generator that uses reactor steam and exhausts to the suppression pool. The benefits of this modifi-

Generic Issues

cation would be a further reduction in the frequency of station blackout sequences, similar to that which might be obtained by adding another gas turbine generator.

- Alternate pump power source - provide a separate diesel generator and supporting auxiliaries to power the feedwater or condensate pumps. This modification would remove the reliance of these pumps on offsite power, and permit them to be used as a backup to HPCF and low-pressure core flooders.
- Dedicated dc power supply - provide a separate, diverse dc power source (fuel cell or separate battery) to supply a dc motor-pump combination for RPV and containment cooling. This modification would further reduce the risk from loss of offsite power and station blackout.
- ATWS-sized vent - provide a wetwell vent line capable of passing the steam flow associated with ATWS. The system would be significantly larger than the existing containment overpressure protection system (COPS) design, and manually initiated from the control room. This system would prevent containment overpressure failure in ATWS events, and thereby prevent core damage.
- Reactor building sprays - modify the fire water spray system in the reactor building to spray in areas vulnerable to release. This modification would reduce the risk associated with releases into the reactor building, such as drywell head failures and containment bypass events, but would not impact releases via COPS.
- Flooded rubble bed - provide a bed of refractory pebbles that would be flooded with water. The rubble bed would impede the flow of molten corium to the concrete drywell structures, and increase the available heat transfer area, thereby enhancing debris coolability. This modification would further reduce the potential for core concrete interactions in the ABWR. A major drawback of the modification is that additional experimental testing would be necessary to validate the concept for the ABWR application.

20.5.1.4 Risk Reduction Potential of Design Improvements

20.5.1.4.1 GE Evaluation of Risk Reduction Potential

GE used the reduction in cumulative risk of accidents occurring during the life of the plant as the basis for estimating the benefit that could be derived from plant improvements. Estimates of risk reduction were developed

by determining the approximate effect of each modification on the frequency of the various release classes in the PRA. GE's basis for estimating the risk reduction for each design improvement is provided in SSAR Section 19P.4, and summarized in Table 20.5.1-4 of this report.

The staff reviewed GE's bases for estimating the risk reduction associated with the various design improvements. The staff notes that considerable judgement was exercised in estimating the risk reduction potential, but that in general, the rationale and assumptions on which the risk reduction estimates are based (center column of Table 20.5.1-4) are reasonable, and in many cases conservative. However, this is not to say that the estimates of person-Sv (person-rem) averted are conservative, since the staff is not in complete agreement with GE's characterization of baseline risk. For example, the risk-reduction potential for improved vacuum breakers appears to be underestimated in GE's analysis. GE estimates that improved vacuum breakers (addition of a second vacuum breaker valve in series with each of the existing valves) would reduce risk by about $3\text{E-}7$ person-Sv (0.00003 person-rem). This value is in large part due to significant credit for fission product removal by wetwell sprays (when available), and the failure to account for the impact of the design improvement on bypass scenarios in which sprays are not available. GE's risk-reduction estimate for this improvement would increase by at least three orders of magnitude if just the latter factor was accounted for. Nevertheless, the risk reduction would remain small since the probability of the events involved is on the order of $1\text{E-}10$ per reactor year.

20.5.1.4.2 Staff Evaluation of Risk-Reduction Potential

In view of the extremely small residual risk for the ABWR, rather than perform an independent assessment of the risk-reduction potential of each ABWR design improvement, the staff used a bounding assumption that each improvement would eliminate all of the risk for the ABWR (0.01 person-Sv (1 person-rem) for the 60-year plant life). This approach tends to overestimate the benefits because the ABWR risk profile reflects contributions from several unique types of sequences (e.g., station blackout, containment bypass, and LOCAs). An individual design improvement would generally reduce or eliminate some of these contributors but would not be effective on others. Moreover, there are numerous and diverse modes of containment failure which must be dealt with to ensure containment integrity in a severe accident. Thus, a carefully selected set of plant improvements would generally be needed, each one acting on particular components of risk, to effectively and significantly reduce total risk.

Table 20.5.1-4 Summary of GE's assessment of risk reduction for candidate design improvements

Potential ABWR design modification	GE's basis for estimating risk reduction	Person-SV (person-rem) averted
Accident Management		
Severe accident EPGs/AMGs	10% reduction in failure rates for manually initiated mitigative actions	0.00015 (0.015)
Computer-aided instrumentation	10% reduction in failure rates for manually initiated preventive actions	0.00010 (0.01)
Improved maintenance procedures/manuals	10% improvement in reliability of HPCF, RCIC, RHR, LPCF	0.00016 (0.016)
Decay Heat Removal		
Passive high pressure system	Equivalent to adding a diverse RCIC and RHR system with 10% unavailability	0.00138 (0.138)
Improved depressurization system	Factor of 2 reduction in depressurization failure rates	0.00042 (0.042)
Suppression pool jockey pump	10% improvement in reliability of low pressure makeup (resulting in 2% reduction in core damage frequency from low pressure sequences)	0.00002 (0.002)
Safety-related condensate storage tank	Engineering judgement	0.00010 (0.01)
Containment Capability		
Larger volume containment	Elimination of all containment release modes involving drywell head failure (Cases 3, 6, 7, 8, 9)	0.00150 (0.15)
Increased containment pressure capacity	Elimination of all containment release modes except normal containment leakage	0.00020 (0.02)
Improved vacuum breakers	Elimination of releases from Release Class 2	.0000003 (0.00003)
Improved bottom head penetration design	Factor of 2 increase in the probability of arresting core damage in-vessel	0.00057 (0.057)

Generic Issues

Table 20.5.1-4 Summary of GE's assessment of risk reduction for candidate design improvements (continued)

Potential ABWR design modification	GE's basis for estimating risk reduction	Person-SV (person-rem) averted
Containment Heat Removal Large volume suppression pool	Elimination of Class II Sequences	.000002 (0.0002)
Containment Mass Removal Low-flow filtered vent	Elimination of the risk associated with releases via COPS	0.00014 (0.014)
Containment Spray Systems Drywell head flooding	Elimination of drywell head over-temperature failures and reduction in releases from drywell head over-pressure failures	0.00060 (0.06)
Prevention Concepts Additional service water Loop	10% increase in reliability of HPCF, RCIC, RHR, LPCF	0.00016 (0.016)
AC Power Supplies Steam-driven turbine generator	80% reduction in the diesel generator common mode failure rate	0.00052 (0.052)
Alternate pump power source	Equivalent to adding a diverse RCIC system	0.00069 (0.069)
DC Power Supplies Dedicated dc power supply	Factor of 10 increase in RCIC availability in LOOP and SBO sequences	0.00069 (0.069)
ATWS Capability ATWS-sized vent	Elimination of risk from ATWS (Case 9)	0.00030 (0.03)
System Simplification Reactor building sprays	10% reduction in risk from releases through the reactor building	0.00017 (0.017)
Core Retention Devices Flooded rubble bed	Elimination of sequences with core concrete interactions, except those with failure of containment heat removal (1% of Cases 1, 6, and 7)	0.000010 (0.001)

20.5.1.5 Cost Impacts of Candidate Design Improvements

GE determined the approximate costs for each design improvement. The costing methodology and assumptions are described in SSAR Section 19P.1.3. The cost basis for each plant improvement is provided in SSAR Section 19P.5 on an item-by-item basis.

GE stated in the SSAR that the cost estimates represent the incremental costs that would be incurred in a new plant, rather than costs that would apply on a backfit basis. GE also stated that the costs were intentionally biased on the low side, but that all known or reasonably expected costs were accounted for so that a reasonable assessment of the minimum cost would be obtained.

For modifications which reduce core damage frequency, GE reduced the costs of the design improvements by an amount proportional to the reduction in the present worth of the risk of averted onsite costs. Onsite costs considered include replacement power at \$.013/kwh differential cost, direct accident costs including onsite cleanup at \$2 billion, and the economic loss of the facility at \$1.4 billion. The resulting costs for each of the design improvements are provided in Table 20.5.1-3.

The staff reviewed the bases for GE's cost estimates and finds them to be reasonable. For certain improvements, the staff also compared GE's cost estimates with estimates developed elsewhere for similar improvements, even though the bases for some of these cost estimates were different. The staff considered the cost estimates developed as part of: the evaluation of design improvements for GESSAR II (NUREG-0979, Supplement 4), and the review of SAMDAs for Limerick and Comanche Peak (NUREG-0974 and -0775, respectively).

The staff noted a number of inconsistencies in the cost estimates. For example, GE's cost estimates for certain improvements such as improved vacuum breakers (\$100,000), modified reactor building sprays (\$100,000), and ATWS-sized vent (\$300,000) were considerably less than expected. The costs for certain other improvements was much higher than expected, such as improved bottom head penetration design (\$750,000) and flooded rubble bed (approximately \$19 million).

It should be noted that only rough approximations of the costs of specific improvements are possible at this time. Large uncertainties exist because detailed designs are not available and because experience with construction and licensing problems that could surface with this type of work is limited. Nevertheless, the staff views GE's approximate cost estimates as adequate, given the

uncertainties surrounding the underlying cost estimates, and the level of precision necessary given the greater uncertainty inherent on the benefit side, with which these costs were compared.

20.5.1.6 Cost-Benefit Comparison

A cost-benefit comparison was performed to determine whether any of the potential severe accident design features could be justified. GE's estimates of the cost per person-Sv (person-rem) averted for the various design improvements are presented in Table 20.5.1-3. The GE values are based on the risk-reduction estimates reported in Tables 20.5.1-3, and 20.5.1-4 of this report. The staff analysis is based on the conservative assumption that each design improvement would eliminate all of the residual risk (0.01 person-Sv (1 person-rem) over the 60-year plant life).

Consistent with current NRC practice (NUREG-3568), GE used a screening criterion of \$100,000 per person-Sv (\$1000 per person-rem) averted to identify whether any of the design improvements could be cost-effective. As shown in Table 20.5.1-3, the potential cost per averted person-rem ranges from about \$1.7 million to far-in-excess of \$40 million for the various suggested modifications according to the GE evaluation. Thus, this far exceeds the \$100,000 per person-Sv (\$1000 per person-rem) criterion. On this basis, GE concluded that no additional modifications to the ABWR design are warranted.

The staff's assessment similarly indicates that none of the design improvements approach a level where they could be considered cost-effective, in spite of the significant conservatism in assessing risk-reduction potential in the staff's analysis. The staff notes that the lowest cost modifications were estimated to cost about \$100,000, and realistically would only partially reduce the residual risk for the ABWR. Even though the cost of implementing design improvements in the ABWR may be less than for an existing plant, given that the ABWR has not yet been constructed, relatively large costs are still to be anticipated for many of the design improvements because they would involve first-of-a-kind engineering, and would need to be integrated within the existing design. In addition, the introduction of a new system will trigger a series of related requirements such as incremental training, procedural changes, and possible licensing requirements. These are all legitimate costs that require consideration in a comprehensive cost estimate. The staff concludes that none of the modifications evaluated would be cost-effective given the low residual risk for the ABWR, and the \$1000 per person-rem criterion.

Generic Issues

The staff has considered the robustness of this conclusion relative to a number of critical assumptions in the analysis as described below. These involve: the effect of uncertainties in estimating core damage frequency, the use of alternative cost-benefit criteria, and the inclusion of external events within the scope of the analysis.

Based on uncertainty analyses performed by GE for the Level 1 portion of the PRA (see FSER Section 19.1.3.2.6), the 95th percentile core damage frequency is $4.5E-7$ per reactor year. This is a factor of three higher than the mean value on which the cost-benefit analysis is based, but still very low both compared to operating plants and in absolute terms. Even if the benefits of the various design improvements were re-quantified on the basis of this upper-bound value, none of the improvements would become cost beneficial. This would remain the case even if the cost-benefit criterion was also increased by a factor of 10 to \$1 million per person-Sv (\$10,000 per person-rem) averted.

If external events are included, the estimate of ABWR risk could be one or possibly two orders of magnitude higher than considered in this analysis. For example, based on the BNL review of GE's original seismic PRA, as documented in the DSER (SECY-91-309), the total risk from internal and seismic events for the 60-year plant life would range from about 0.4 to 2 person-Sv (40 to 200 person-rem) depending on the site population. However, the value for the final ABWR design would be somewhat less since these estimates do not account for plant improvements incorporated in the design subsequent to the original PRA analysis, including upgrading the seismic capability of the diesel-driven fire water pump.

Even assuming the higher of these two risk estimates and complete elimination of all risk, any design modifications or combinations which cost more than \$200,000 would not be cost-effective. This would eliminate most of the candidate design modifications from further consideration. Based on the GE analysis, those modifications which were estimated to cost less than \$200,000 have a relatively low risk-reduction potential, and would generally eliminate only about 10 percent of the residual risk from internal events. The improvements are also not expected to be effective in eliminating most of the added risk from seismic events. Since the minimum cost of these systems would be about \$100,000, none of these improvements are expected to be cost-effective when their actual effectiveness in reducing risk is taken into account.

The staff concludes that with the significant margins in the results of the cost-benefit analysis, the findings of the analysis would be unchanged even considering the above factors.

20.5.1.7 Conclusions

As discussed in Chapter 19.1.3.2.2 of this report, GE has made extensive use of the results of the PRA to arrive at a final ABWR design. As a result, the estimated core damage frequency and risk calculated for the ABWR is very low both relative to operating plants and in absolute terms. The low core damage frequency and risk for the ABWR is a reflection of GE's efforts to systematically minimize the effect of initiators or sequences that have been important contributors to core damage frequency in previous BWR PRAs. This has been done largely through the incorporation of a number of hardware improvements in the ABWR design. These include providing three separated divisions of ECCSs, a diverse and independent combustion gas turbine capable of providing ac power to any of the three divisions, an ac-independent water addition system, and an FMCRD system as a backup to the hydraulic drive system. Several improvements have also been incorporated in the ABWR design to mitigate the consequences of a core damage event, including inerting of the containment atmosphere, inclusion of a lower drywell floodler system and a containment overpressure protection (vent) system, the use of basaltic concrete in the lower drywell, and an increased ultimate pressure capacity. These and additional ABWR design features which contribute to low core damage frequency and risk for the ABWR are discussed further in Section 19.1.3.2.3 of this report.

Because the ABWR design already includes numerous plant features oriented towards reducing core damage frequency and risk, the benefits and risk-reduction potential of additional plant improvements is significantly reduced. This is true for both internally and externally initiated events. For example, the ABWR seismic design basis (0.3g safe shutdown earthquake) has been shown to result in significant ability to withstand earthquakes well beyond the design basis, as characterized by a high confidence with low probability of failure (HCLPF) value of 0.6g. Moreover, with the features already incorporated in the ABWR design, the ability to estimate core damage frequency and risk approaches the limitations of probabilistic techniques. Specifically, when core damage frequencies of one in a hundred thousand or a million years are estimated in a PRA, it is the areas of the PRA where modelling is least complete, or supporting data is sparse or even nonexistent that could actually be the more important contributors to risk. Areas not modelled or incompletely modelled include human reliability, sabotage, rare initiating events, construction or design errors, and systems interactions. Although improvements in the modelling of these areas may introduce additional contributors to core damage frequency and risk, the staff

does not expect that they would be significant in absolute terms.

10 CFR 50.34(f)(1)(i) requires the applicant to perform a plant- or site- specific probabilistic risk assessment, the aim of which is to seek such improvements in the reliability of core and containment heat removal systems as are significant and practical and do not impact excessively on the plant. The staff concludes that the ABWR PRA, and GE's use of the insights of this study to improve the design of the ABWR meets this requirement. The staff concurs with the GE conclusion that none of the potential design modifications evaluated are justified based on cost-benefit considerations. The staff also concludes that it is unlikely that any other design changes would be justified on the basis of person-Sv (person-rem) exposure considerations, because the estimated core damage frequencies would remain very low on an absolute scale.

20.5.2 10 CFR 50.34(f)(1)(ii): System Design -- Auxiliary Feedwater System - Auxiliary Feedwater System Evaluation (TMI Item II.E.1.1)

Paragraph (1)(ii) of 10 CFR 50.34(f) requires the performance of an evaluation of the proposed AFWS of PWR plants to include a simplified AFWS reliability analysis using event-tree and fault-tree logic techniques, design review of AFWS, and an evaluation of AFWS flow design bases and criteria.

This requirement is applicable to PWRs only, since a BWR design does not include an AFWS. Therefore, it is not technically relevant to the ABWR design and does not need to be addressed.

20.5.3 10 CFR 50.34(f)(1)(iii):

Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Commission Orders on B&W Plants - Impact of RCP Seal Damage Following SBLOCA With Loss of Offsite Power (TMI Item II.K.2(16)) and

Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Effect of Loss of ac Power on Pump Seals (TMI Item II.K.3(25))

TMI Item II.K.2(16) is applicable to PWRs only and does not need to be addressed for the ABWR, but II.K.3(25)

applies to BWRs. Paragraph (1)(iii) of 10 CFR 50.34(f) requires analyses or experiments to determine the consequences at each plant of a loss of cooling water to the reactor recirculation pump seal coolers. Pump seals should be designed to withstand a complete loss of ac power for at least two hours and the adequacy of the seal design should also be demonstrated. The design should prevent an excessive loss of reactor coolant inventory after an anticipated operational occurrence. It is assumed that the loss of ac power constitutes a loss of offsite power.

SSAR Sections 5.4.1 and 1A.2.30 state that the ABWR design features reactor internal pumps (RIPs) that do not require pump shaft seals. During a loss of ac power, the RIPs are shut down automatically, but there are no shaft seals that require restoration of cooling. During its review of this issue, the staff required GE to confirm that the failure of the following systems would not cause a LOCA

- recirculation motor cooling system
- recirculation motor seal purge system
- recirculation motor inflatable shaft seal subsystem.

GE confirmed in SSAR Section 1A.2.30 that an ac failure would temporarily disrupt the operation of these systems, but their failure would not generate a LOCA. Therefore, the staff concludes that GE adequately addressed the requirements of this TMI item for the ABWR design.

20.5.4 10 CFR 50.34(f)(1)(iv): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Report on Overall Safety Effect of PORV Isolation (TMI Item II.K.3(2))

Paragraph (1)(iv) of 10 CFR 50.34(f) requires performance of an analysis of the probability of a SBLOCA caused by a stuck-open PORV of PWR plants. If the probability is a significant contributor to the probability of SBLOCAs from all causes, it also requires a description and evaluation of the effect on SBLOCA probability of an automatic PORV isolation system that would operate when the RCS pressure falls after the PORV has opened.

This requirement is applicable to PWRs only, therefore, it is not technically relevant to the ABWR design and does not need to be addressed.

Generic Issues

20.5.5 10 CFR 50.34(f)(1)(v): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Separation of HPCI and RCIC System Initiation Levels (TMI Item II.K.3(13))

Paragraph (1)(v) of 10 CFR 50.34(f) requires an evaluation of the safety effectiveness of providing for separation of the HPCI and RCIC system initiation levels so that the RCIC system initiates at a higher water level than the HPCI system, and of providing that both systems restart on low water level. For plants with high pressure core spray (HPCS) systems in lieu of HPCI systems, HPCS system is to be evaluated in lieu of the HPCI system.

In NUREG-0737, the staff stated that the initiation levels of the HPCI and RCIC systems should be separated so that the RCIC system initiates at a higher water level than the HPCI system. Further, the initiation logic of the RCIC system should be modified so that the RCIC system will restart upon receiving a low water level signal. These changes could reduce the number of challenges to the HPCI system and could result in less stress on the reactor vessel from cold water injection. Applicants were required to submit the analyses of these changes to the NRC staff and implement the changes if justified by the analyses.

The ABWR design is consistent with this position. The ABWR incorporates an HPCF system that initiates at reactor vessel level 1.5 and an RCIC system which initiates at level 2. At level 8, the injection valves for the HPCF and the RCIC steam supply and injection valves automatically close to prevent water from entering the main steam lines.

If the RPV again reaches a low level, the RCIC steam supply and injection valves automatically reopen to allow the RCIC to continue flooding the vessel. The HPCF injection valves will also automatically reopen unless the operator previously closed them manually.

The ABWR has three high-pressure makeup systems, two HPCFs and one RCIC. Current BWRs have the HPCS or HPCI, and RCIC. Thus, the water level set points for ABWR must meet the following requirements:

- During anticipated abnormal transients, including the loss of all feedwater flow event, the RCIC system will prevent the minimum water level from dropping below vessel level 1.5. This requirement is to minimize the challenge to the HPCFs.
- The ADS will not likely be actuated during any abnormal transient with a failure of RCIC. Therefore,

the reactor isolation and two HPCFs initiated at vessel level 1.5 will prevent the water level from dropping below level 1 during a loss-of-all-feedwater-flow transient with RCIC failure.

The staff found that GE provided different initiation levels for HPCF and RCIC so that the RCIC system initiates at a higher water level than the HPCF system, as well as the automatic restart of RCIC on low level. GE's proposed response adequately addresses the requirements of this TMI item for the ABWR design. An in-depth discussion of the RCIC system is provided in Section 5.4.6 of this report and of RCIC and HPCF in Section 6.3.1 of this report.

20.5.6 10 CFR 50.34(f)(1)(vi): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Reduction of Challenges and Failures of Relief Valves; Feasibility Study and System Modification (TMI Item II.K.3(16))

Paragraph (1)(vi) of 10 CFR 50.34(f) requires the performance of a study to identify practicable system modifications that would reduce challenges and failures of relief valves in BWRs, without compromising the performance of the valves or other systems.

In 1980, the staff determined that in the prior three years of operation of all BWR plants, there were approximately 30 failures of relief valves to close in 73 reactor-years (0.41 failures per reactor-year). This demonstrated that the failure of a relief valve to close was the most likely cause of an SBLOCA. The high failure rate is the result of a high relief-valve challenge rate and a relatively high probability of failure for each challenge (0.16 failures per challenge). Typically, five valves are challenged per event. This results in an equivalent failure rate for each challenge of 0.03. In NUREG-0737, the staff stated that the challenge and failure rates can be reduced in many ways.

To resolve this issue, the staff required GE to investigate the feasibility of reducing challenges to relief valves and implement those changes shown to reduce challenges to the safety and relief valves without compromising the performance of the relief valves or other systems. Challenges to the relief valves were expected to be reduced substantially (by an order of magnitude).

GE evaluated the possible ways for reducing the challenges and failure rate of SRVs, and reduced the MSIV isolation set points from RPV level 2 to level 1.5. The staff concludes that this modification will reduce SRV challenge

rates and concludes that GE's response adequately addresses the requirements of this TMI item for the ABWR design. Further discussion of SRVs is provided in Section 5.2.2 of this report.

20.5.7 10 CFR 50.34(f)(1)(vii): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Modification of ADS Logic; Feasibility Study and Modification for Increased Diversity for Some Event Sequences (TMI Item II.K.3(18))

Paragraph (1)(vii) of 10 CFR 50.34(f) requires performance of a feasibility and risk assessment study to determine the optimum ADS design and modifications that would eliminate the need for manual activation to ensure adequate core cooling for BWRs.

In NUREG-0737, the staff stated that the ADS actuation logic should be modified to eliminate the need for manual actuation to ensure adequate core cooling. The applicant was required to conduct a feasibility and risk assessment study to determine the best approach. For example, the applicant was to consider actuating the ADS on a low reactor vessel water level signal if the instruments indicate no HPCI or HPCS flow and a low-pressure ECCS is running. This logic would complement, not replace, the existing ADS actuation logic. The ADS must be manually actuated to cool the core adequately for transient and accident events that do not directly produce a high drywell pressure signal (e.g., stuck-open relief valve or steam line break outside containment), and are degraded by the loss of high-pressure ECCSs. This TMI item requires that the ADS logic be modified to eliminate operator action.

In the DFSE, the staff reported that GE proposed to modify the ABWR design in a manner consistent with option 4 of the BWROG response to TMI Item II.K.3(18). GE discussed this proposal in a letter dated October 28, 1982, from T.J. Dente (GE) to D.G. Eisenhut (NRC), "NUREG-0737 Item II.K.3.18 Modification of Automatic Depressurization System Logic." This modification involves a manual inhibit switch and a timer that bypasses the high drywell pressure permissive if the reactor water level is low for a sustained period. GE added an 8-minute high drywell pressure bypass timer to the ABWR ADS initiation logic. This timer will initiate on a low water level-1 signal. When it times out, it bypasses the need for a high drywell pressure signal to initiate the standard ADS initiation logic.

The DFSE reported that the bypass timer would be tested periodically as required by the STS and stated that this was identified in Chapter 6 of the DFSE as TS Item 6.3.3-1.

It also indicated that GE documented the results of its evaluation of the adequacy of the 8-minute bypass timer in the SSAR (see GE letter to NRC, MFN No. 038-92, dated February 14, 1992). The staff agreed with the analysis and results and concluded that the modifications to the ADS design were acceptable. In response to TS Item 6.3.3-1, GE modified the ABWR TS to require periodic testing of the ADS bypass timer. The staff concludes that GE has adequately responded to this item as discussed further in Section 6.3.3 of this report.

On the basis of its review of the ADS design and resolution of the DFSE TS item, the staff concludes that GE's response adequately addresses this TMI requirement.

20.5.8 10 CFR 50.34(f)(1)(viii): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Restart of Core Spray and LPCI Systems on Low Level; Design and Modification (TMI Item II.K.3(21))

Paragraph (1)(viii) of 10 CFR 50.34(f) requires performance of a study of the effect on all core-cooling modes of BWRs under accident conditions of designing the core spray and low-pressure coolant injection systems to ensure that the systems will automatically restart on loss of water level, after having been manually stopped, if an initiation signal is still present.

In NUREG-0737, the staff identified a concern that the operator may stop the required flow of the core spray and LPCI system. These systems will not restart automatically on a loss-of-water level signal if an initiation signal is still present. The staff indicated that the vendor should modify the core spray and LPCI system logic so that these systems will restart, if required, to ensure adequate core cooling. Before modifying the design, the staff also stated that the vendor should submit a preliminary design to the staff for approval because it affects several core-cooling modes under accident conditions.

In the DFSE, the staff reported that for the ABWR, GE endorses the conclusions of the study performed by the BWROG which was forwarded to the staff in a December 29, 1980, letter from D.B. Waters (BWROG) to D.G. Eisenhut (NRC). The BWROG concluded that the current BWR ECCS design is adequate and that the proposed changes would decrease the overall safety of the plant. For example, the BWROG stated that the modification would significantly escalate the control system complexity, restrict the operator's flexibility when dealing with anticipated events, and reduce system reliability. The BWROG concluded that the current ECCS design is

Generic Issues

adequate because the BWR operator training is comprehensive and thoroughly addresses reactor-water-level control, the EOPs address this issue, the operator has sufficient time to correct errors, and the low reactor-water-level conditions are clearly displayed and alarmed in the control room.

The staff reported that it reviewed the results of the BWROG study and considered the emphasis placed on water-level control in BWR operator training. The staff found GE's response acceptable and agrees that the ABWR design need not be modified to provide an automatic restart of the low-pressure ECCS. Therefore, the staff concludes that the ABWR design meets the requirements of this TMI item.

20.5.9 10 CFR 50.34(f)(1)(ix): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Confirm Adequacy of Space Cooling for HPCI and RCIC Systems (TMI Item II.K.3(24))

Paragraph (1)(ix) of 10 CFR 50.34(f) requires a determination of the adequacy of space cooling for the high pressure coolant injection systems at BWRs. Long-term operation of these systems during a complete loss of offsite power, for up to two hours, may require space cooling to maintain the HPCI pump rooms within allowable temperature limits. The HPCI systems and their respective support systems should be designed to withstand the consequences of a complete loss of offsite ac power for 2 hours. For plants with HPCS systems in lieu of HPCI systems, the HPCS system is to be evaluated in lieu of the HPCI system.

The ABWR design contains two systems to provide room cooling to the HPCF and RCIC systems. During normal operation, the two HPCF pump rooms and the single RCIC pump room are cooled by the non-safety-related secondary containment HVAC system. During accident conditions, the room-cooling function is transferred to the safety-grade essential equipment HVAC system and the secondary containment HVAC system is isolated. The essential equipment HVAC system consists of a fan coil unit in each of the pump rooms. Cooling for each fan coil is provided by the safety-related portion of the applicable train of the RCW system. The fan coil unit in the applicable pump room, the HPCF subsystem or the RCIC system which the fan coil unit is serving, and the associated RCW train are all on the same essential electrical power division, which includes the divisional onsite ac power source, that is, emergency onsite diesel generator. The fan coil unit is necessary to keep the temperature of

the associated pump room within its design limits. It is automatically initiated upon startup of the respective HPCF or RCIC pump. Based on the above, the staff concludes that space cooling for the HPCF and the RCIC systems will be available as required by this TMI item following a complete loss of offsite ac power to the plant for at least 2 hours.

In the DFSER, the staff noted that the design characteristics for the fan coil units were not specifically identified in SSAR Section 9.4.5. The staff was not able to confirm the ability of the fan coil units to remove sufficient heat to maintain the pump room temperatures within design limits. Also, SSAR Tables 8.3-1 and 8.3-2, which list diesel generator loads, did not include the fan coil units. These concerns were identified in the DFSER as Open Item 20.3-1. Subsequently, in Amendment 32 of the SSAR, GE stated in Section 9.4.5.2.2.1 that the fan coil units are sized to maintain the rooms within 40 °C (104 °F) operational temperature, which is well below the temperature limits specified in SSAR Appendix 3I regarding environmental qualification. Based on this information, the staff determined that there is sufficient margin to provide adequate space cooling during a LOCA. Further, GE added the fan coil units and their capacities to SSAR Table 9.4-4e, and their associated loads in SSAR Table 8.3-1 (the operational loads are one-half of the connected loads). Based on this additional information, the staff concludes that the fan coil units receive emergency power from the diesel generators and are sized to remove the worst-case heat load from the rooms. This additional information adequately resolved the concerns of DFSER Open Item 20.3-1.

The staff concludes that the HPCF and RCIC room cooling units are adequately sized and powered to maintain the rooms within design environmental conditions following a complete loss of offsite ac power to the plant for at least 2 hours, and therefore, the ABWR design adequately addresses the requirements of this TMI item.

20.5.10 10 CFR 50.34(f)(1)(x): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Study and Verify Qualification of Accumulators on ADS Valves (TMI Item II.K.3(28))

Paragraph (1)(x) of 10 CFR 50.34(f) requires the performance of a study to ensure that the ADS valves, accumulators, and associated equipment and instrumentation of BWRs will be capable of performing their intended functions during and following an accident situation, taking no credit for non-safety-related equipment

or instrumentation, and accounting for normal expected air (or nitrogen) leakage through valves.

The ABWR has 18 quality Group A, seismic Category I SRVs, eight of which can also perform the ADS function. One 189-liter (50-gallon) capacity nitrogen accumulator with a design pressure of 1379 kPa (200 psig) is provided for each ADS SRV to support its ADS function during and following an accident situation. The accumulator supplies compressed nitrogen gas to the valve for its actuations. The staff's evaluation is based on SSAR Sections 1A.2.31 and 6.7 and is limited to the adequacy of the accumulators and associated equipment to perform their intended functions during and following an accident situation. The staff's evaluation regarding the system's compliance with other regulatory requirements, such as the adequacy of the ADS SRV valves and their associated instrumentation and controls to perform their intended functions during and following an accident situation, is provided in Sections 5.2.2, "Overpressure Protection," 6.3, "Emergency Core Cooling Systems," and 7.3, "Engineered Safety Features Systems," of this report.

Each accumulator is sized to provide sufficient nitrogen gas for one associated ADS valve actuation at drywell design pressure or five actuations at normal drywell pressure with nominal pneumatic supply. Makeup supply for the accumulators is provided by the safety-related portions of the nitrogen gas supply system, which includes two redundant safety-related nitrogen gas supply trains. Each train consists of 10 high-pressure nitrogen gas bottles and associated piping, valves, and instrumentation. The nitrogen bottles in each train have sufficient stored compressed nitrogen gas to supply makeup nitrogen gas to the accumulators associated with four ADS valves to compensate for an expected nitrogen leakage of 28 liters per hour (L/hr) (1 standard cubic foot per hour (scfh)) for each valve for a 7-day period. Additionally, the accumulators can be refilled after the system is operating using the non-safety-related portion of the nitrogen gas supply system in conjunction with the non-safety-related atmospheric control system (ACS) described in SSAR Section 6.2.5. GE determined that a maximum of three ADS valves will be needed to meet short-term demands and one ADS valve will be needed to meet long-term needs and so, one safety-related train of the nitrogen gas supply system will be sufficient to provide the needed nitrogen makeup to the associated ADS accumulators. The system has alarm provisions to indicate failure of any redundant nitrogen supply train due to a loss of nitrogen supply pressure to the ADS valve accumulator.

The accumulators, including the associated equipment to the ADS valves, and the safety-related portions of the nitrogen gas supply system are designed to seismic

Category I, quality Group B or C, as appropriate, and quality assurance B requirements. The accumulators, including the associated equipment to the ADS valves, are designed to operate in the environmental conditions to be found in the drywell after a design-basis accident (see SSAR Section 7.3.1.1.2 for specific additional information and Section 3.11 of this report for general information on environmental qualification design criteria for equipment important to safety). SSAR Section 7.3.1.1.10 states that the safety-related equipment in the nitrogen gas supply system are selected to accommodate the hostile environment to which they may be exposed during an accident situation. The nitrogen supply system is designed to 1379 kPa (200 psig) and 66 °C (151 °F). In the DFSE, the staff expressed concern regarding the design temperature. The 66 °C (151 °F) design temperature is significantly lower than the temperature to which some portions of the system (e.g., inside the drywell and the reactor building) may be exposed during an accident situation. The staff also expressed concern that SSAR Section 6.7.4 did not explicitly specify a requirement for periodically testing the leakage through each valve to verify that such leakage is within the assumed value of 28 L/hr (1 scfh). These concerns were identified in the DFSE as Open Item 20.3-2.

Subsequently, in Amendment 31, GE modified SSAR Sections 6.7.4 and Figure 6.7-1. The figure now indicates that the high pressure nitrogen (HPIN) system piping from inside primary containment to the outboard containment isolation valve is designed to 171 °C (340 °F), while the remainder of the system outside primary containment is designed to 66 °C (151 °F). These specifications ensure that the system can withstand the worst postulated environmental conditions. SSAR Section 6.7.4 now states that periodic testing will be performed on system components to ensure that leakage will not exceed 28 L/hr (1 scfh). Based on this additional information, the staff concludes that the system is designed to ensure proper ADS SRV valve operation. This additional information adequately resolved the concerns of DFSE Open Item 20.3-2.

Based on its evaluation, the staff concludes that (1) GE's criteria for sizing each nitrogen gas accumulator and the provision for supply of makeup nitrogen gas to the accumulator is appropriate, and (2) the nitrogen accumulators and the associated equipment, including the safety-related portions of the nitrogen gas supply system, will perform their intended functions during and following an accident situation. Therefore, the ABWR design adequately addresses the requirements of this TMI item.

Generic Issues

20.5.11 10 CFR 50.34(f)(1)(xi): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Evaluate Depressurization with Other Than Full ADS (TMI Item II.K.3(45))

Paragraph (1)(xi) of 10 CFR 50.34(f) requires provision of an evaluation of depressurization modes, other than full actuation of the ADS, such as early depressurization with one or two SRVs, that would reduce the possibility of exceeding vessel integrity limits during rapid cooldown of BWRs. Slower depressurization would reduce the possibility of exceeding vessel integrity limits by rapidly cooling down.

Since the ABWR design provides three high pressure core injection systems, the probability of ADS actuation is lower than for current BWRs. In the report dated September 27, 1982 (memorandum from T. Speis to G. Lainas), the staff evaluation which addressed this subject concurred with the BWROG study in this area. In the analyses conducted for the study, it was assumed that all high-pressure injection systems would fail, but that all low-pressure systems would operate. The time at which the operator is assumed to actuate the ADS varied. The effects of depressurization over a 10-minute interval and a 20-minute interval were compared to the case in which the RCS is completely depressurized in 3.3 minutes. Vessel fatigue usage was the key parameter studied regarding vessel integrity. The BWROG analyzed the extent to which a longer depressurization period would reduce fatigue usage and then considered the effect of this reduced fatigue usage on the core cooling capability. The BWROG concluded that

- vessel integrity limits are not exceeded for full depressurization using the ADS
- for slower depressurization rates (longer than the approximate 3.3 minute interval for the normal depressurization rate), the usage assignable to the full depressurization using the ADS would not significantly affect vessel fatigue usage relative to that usage assignable to the full ADS blowdown
- slower depressurization rates reduce the core cooling capability, except when the operator begins to depressurize the RCS very early in the accident.

The results also stated that core cooling capability could be improved using a 10-minute depressurization period if the operator actuated the ADS within 1 to 6 minutes after the accident begins. However, it was considered more prudent to activate the high-pressure injection systems during this initial period to avoid using the ADS.

GE performed the analysis for a standard BWR, which has an RPV and ADS similar to the ABWR. Thus, the RPV cooldown rate and material design of the RPV vessel for the ABWR are similar to those features in current designs. Hence, the results should be the same for the ABWR. Based on its review, the staff concludes that the current mode of depressurization using the ADS in the same manner as current operating BWRs for the ABWR is satisfactory and adequately addresses the requirements of this TMI item.

20.5.12 10 CFR 50.34(f)(1)(xii): Evaluation of Alternative Hydrogen Control Systems

Paragraph (1)(xii) of 10 CFR 50.34(f) requires performance of an evaluation of alternative hydrogen control systems that would satisfy the requirement of paragraph (2)(ix) of 50.34(f) (see Section 20.5.21 below). As a minimum, the evaluation must consider a hydrogen ignition and post-accident inerting system and include:

- A comparison of costs and benefits of the alternative systems considered.
- For the selected system, analyses and test data to verify compliance with the requirements of paragraph (2)(ix) of 50.34(f).
- For the selected system, preliminary design descriptions of equipment, function, and layout.

In the DFSER, the staff required GE to provide this information since it had not yet done so. This was identified in the DFSER as Open Item 20.3-3. Subsequently, GE modified SSAR Section 19A.2.12 to state that Section 6.2.7.1 contained COL license information requiring the applicant referencing the ABWR design to provide this information, if appropriate. The staff agrees that evaluating an alternative hydrogen control system is beyond the scope of the ABWR design certification review, and should a COL applicant wish to provide an alternative system, it should provide the supporting information for staff review. Based on the clarifying information provided in the SSAR, the staff concludes that GE has adequately addressed the requirements of 10 CFR 50.34 (f)(1)(xii). Therefore, DFSER Open Item 20.3-3 is resolved.

20.5.13 10 CFR 50.34(f)(2)(i): Operating Personnel -- Simulator Use and Development - Long-Term Training Upgrade (TMI Item I.A.4.2)

Paragraph (2)(i) of 10 CFR 50.34(f) requires the provision of simulator capability that correctly models the control room and includes the capability to simulate SBLOCAs.

GE stated that simulator facilities for use in performing operator training are beyond the scope of the ABWR design certification review and the COL applicant will be responsible for addressing this requirement. This is consistent with the treatment of training in SSAR Chapter 13 and is acceptable because training will be addressed by the COL applicant. This was identified in the DFSER as COL Action Item 18.7.2.2-1. The staff verified that GE established a COL action item (Item 18.8.8 in SSAR Section 18.8) for an operator training program to meet 10 CFR Part 50. This approach adequately addresses DFSER COL Action Item 18.7.2.2-1 as discussed in Sections 13.2 and 18.7.2.2 of this report.

20.5.14 10 CFR 50.34(f)(2)(ii): Operating Procedures - Long-Term Program Plan Procedures for Upgrading of Procedures (TMI Item I.C.9)

Paragraph (2)(ii) of 10 CFR 50.34(f) requires establishment of a program, to begin during construction and follow into operation, for integrating and expanding current efforts to improve plant procedures. The scope of the program is required to include emergency procedures, reliability analyses, HFE, crisis management, operator training, and coordination with INPO and other industry efforts.

The development of detailed procedures is beyond the scope of the ABWR design certification review and the COL applicant will be responsible for addressing this TMI item in the detailed design implementation. This was identified in the DFSER as COL Action Item 18.7.2.2-2. The staff verified that GE established a COL action item in SSAR Section 13.5.3 for procedure development. Additionally, SSAR Section 13.5.3.1 states that the methods and criteria for the development, V&V, implementation, maintenance, and revision of procedures will include considerations of I.C.9. This approach is adequately addresses DFSER COL Action Item 18.7.2.2-2 as discussed in Section 13.5 of this report.

20.5.15 10 CFR 50.34(f)(2)(iii): Control Room Design - Control Room Design Reviews (TMI Item I.D.1)

Paragraph (2)(iii) of 10 CFR 50.34(f) requires the provision, for NRC review, of a control room design that

reflects state-of-the-art human factor principles prior to committing to fabrication or revision of fabricated control room panels and layouts.

GE stated that this requirement is beyond the scope of the ABWR design certification review and the COL applicant will be responsible for addressing it in the detailed design implementation. This was identified in the DFSER as COL Action Item 18.7.2.2-3. The staff verified that GE addressed this requirement further in the HFE design (SSAR Section 18.8), included detailed CR development in ABWR CDM Table 3.1 ITAAC and in SSAR Section 18E, and established a COL action item (Item 18.8.1) in SSAR Section 18.8. This approach adequately addresses DFSER COL Action Item 18.7.2.2-3 as discussed in Section 18.7.2.2 of this report.

20.5.16 10 CFR 50.34(f)(2)(iv): Control Room Design - Plant Safety Parameter Display Console (TMI Item I.D.2)

Paragraph (2)(iv) of 10 CFR 50.34(f) requires provision of a plant safety parameter display console that will display to operators a minimum set of parameters defining the safety status of the plant, capable of displaying a full range of important plant parameters and data trends on demand, and capable of indicating when process limits are being approached or exceeded.

GE addresses this requirement in SSAR Section 18.4.2.11 (evaluated in Section 18.3 of this report). This requirement is beyond the scope of the ABWR design certification review and the COL applicant will be responsible for addressing it in the SPDS design. This was identified in the DFSER as COL Action Item 18.7.2.2-4. The staff verified that GE established a COL action item in SSAR Section 18.8 (Item 18.8.4) to evaluate the SPDS design against the applicable provisions of NUREG-0737, Supplement 1. This approach adequately addresses DFSER COL Action Item 18.7.2.2-4 as discussed in Section 18.7.2.2 of this report.

It should be noted that Section 18.7.2.2 of this report states that the ABWR design will not provide a separate SPDS console, but rather, the functions of the SPDS will be integrated into the overall control room display capabilities. The staff concludes that an exemption from the requirements of 10 CFR 50.34(f)(2)(iv) for a separate SPDS console are appropriate for the ABWR for the reasons set forth in Sections 18.3 and 18.7.2.2 of this report.

Generic Issues

20.5.17 10 CFR 50.34(f)(2)(v): Control Room Design - Safety System Status Monitoring (TMI Item I.D.3)

Paragraph (2)(v) of 10 CFR 50.34(f) requires provisions for automatic indication of the bypassed and inoperable status of safety systems.

SSAR Section 7.1.2.10.2 states that the ABWR standard plant design complies with the requirements of IEEE 279, paragraph 4.13; RG 1.47; and BTP ICSB 21. IEEE 279, paragraph 4.13 and RG 1.47 state that if the protective action of some part of the I&C system has been bypassed or deliberately rendered inoperative for any purpose, this fact will be continuously indicated in the control room. The ABWR design also provides for automatic indication at the system level when the system loses power or when it is out of service. In addition, a switch will be provided for manual initiation of bypass indication for out-of-service conditions under limited circumstances when bypass is not automatically annunciated. On this basis, the staff concludes that the ABWR standard design meets the stated guidance of RG 1.47, and therefore, meets the requirements of 50.34(f)(2)(v) with respect to the I&C design of safety system status monitoring. See also Section 7.2.3 of this report.

The human factors details of this requirement are beyond the scope of the ABWR design certification review and GE stated that they will be addressed by the COL applicant in the detailed design implementation. This was identified in the DFSER as COL Action Item 18.7.2.2-5. The staff verified that GE established a COL action item in SSAR Section 18.8 (Item 18.8.1) to conduct the detailed HFE design according to design and implementation as defined by the ABWR CDM Table 3.1 ITAAC and SSAR Appendix 18E. The staff considers this to include the resolution of the human factors aspects of this requirement. This approach adequately addresses DFSER COL Action Item 18.7.2.2-5 as discussed in Section 18.7.2.2 of this report.

20.5.18 10 CFR 50.34(f)(2)(vi): Consideration of Degraded or Melted Cores in Safety Review - Reactor Coolant System Vents (TMI Item II.B.1)

Paragraph (2)(vi) of 10 CFR 50.34(f) requires that each applicant and licensee install high-point vents for the RCS and the reactor vessel head. These vents will be remotely operated from the control room. Although the purpose of the system is to vent noncondensable gases from the RCS which may inhibit core cooling during natural circulation, the vents must not lead to an unacceptable increase in the probability of a LOCA or a challenge to containment

integrity. These vents are part of the reactor coolant pressure boundary and thus, shall conform to the regulatory requirements of 10 CFR Part 50. The vent system shall be designed with sufficient redundancy to ensure a low probability of inadvertent or irreversible actuation.

This TMI item also requires the submittal of the following information on the design and operation of the high-point vent system:

- A description of the design, location, size, and power supply for the vent system, along with results of analyses for LOCAs initiated by a break in the vent pipe. The results of the analyses should demonstrate compliance with the acceptance criteria of 10 CFR 50.46.
- Procedures and supporting analysis for the operator to use in operating the vents. This information should include the information available to the operator for beginning or ending the use of the vents.

In the ABWR design, the 18 power-operated safety and relief valves are the primary venting capability. Each SRV is seismically qualified. The high-pressure nitrogen gas supply to the eight SRVs which comprise the ADS is also seismically qualified. To vent the RCS, each SRV can be operated in the power-actuated mode by remote manual controls in the main control room. The discharge line for each SRV will include linear variable differential transformers (LVDTs) and a thermocouple to monitor the position and leakage of the SRV. Each SRV discharges to the suppression pool.

The RCS can also be vented through the RCIC system, which directs steam from one of the main steam lines to a turbine-driven pump, from which the steam is exhausted to the suppression pool. The RCIC system can vent the RCS during hot standby mode or during reactor isolation.

The top head vent line of the RPV can also direct steam and noncondensable gases from the reactor's upper dome. This line is used principally to vent the reactor during the final stages of normal shutdown from power operation. A reactor head vent line is a continuous vent which is normally open to discharge to a main steam line.

The COL applicant will develop plant-specific procedures to govern the operator's use of the relief mode for venting the reactor. This was identified in the DFSER as COL Action Item 20.3-1. In the advance SER, the staff concluded that since GE had not yet included a COL action item in the SSAR addressing the development of these procedures, DFSER COL Action Item 20.3-1 would

remain open until GE had done so. In Amendment 33, GE revised SSAR Sections 1A.2.5 and 1A.3.6 to establish a COL action item for the COL applicant to develop the indicated plant procedures. This approach adequately addresses DFSEER COL Action Item 20.3-1.

GE has submitted no additional accident analyses to address a break in any of the vent lines because the plant's design basis includes a complete steam line break, which is more bounding.

The staff concurs with the applicant's assessment because it includes adequate capacity, operation, and procedural provisions of the ABWR vent system. The staff concludes that GE has adequately addressed the requirements of this TMI item for the ABWR design.

20.5.19 10 CFR 50.34(f)(2)(vii): Consideration of Degraded or Melted Cores in Safety Review - Plant Shielding to Provide Access to Vital Areas and Protect Safety Equipment for Post-Accident Operation (TMI Item II.B.2)

Paragraph (2)(vii) of 10 CFR 50.34(f) requires radiation and shielding design reviews of spaces around systems that may, as a result of an accident, contain source term radioactive materials, and design as necessary to permit adequate access to important areas and to protect safety equipment from the radiation environment. GE's response adequately addresses the requirements of this TMI item for the ABWR design as discussed in Sections 12.3.6 and 13.6.3.5 of this report.

20.5.20 10 CFR 50.34(f)(2)(viii): Consideration of Degraded or Melted Cores in Safety Review - Post-Accident Sampling (TMI Item II.B.3)

Paragraph (2)(viii) of 10 CFR 50.34(f) requires the capability to promptly obtain and analyze samples from the reactor coolant system and containment that may contain TID-14844 source term radioactive materials without radiation exposure to any individual exceeding 0.05 Sv (5 rem) to the whole-body or 0.50 Sv (50 rem) to the extremities.

GE's response adequately addresses the requirements of this TMI item for the ABWR design as discussed in Section 9.3.2.2 of this report.

20.5.21 10 CFR 50.34(f)(2)(ix): Consideration of Degraded or Melted Cores in Safety Review - Rulemaking Proceeding on Degraded Core Accidents (TMI Item II.B.8), "Hydrogen Control System"

Paragraph (2)(ix) of 10 CFR 50.34(f) requires a system for hydrogen control that can safely accommodate hydrogen generated by the equivalent of a 100-percent, fuel clad metal-water reaction. The hydrogen control system and associated systems shall provide with reasonable assurance that:

- (A) Uniformly distributed hydrogen concentrations in the containment do not exceed 10 percent during and following an accident that releases an equivalent amount of hydrogen as would be generated from a 100-percent, fuel clad metal-water reaction, or that the post-accident atmosphere will not support hydrogen combustion.
- (B) Combustible concentrations of hydrogen will not collect in areas where unintended combustion or detonation could cause loss of containment integrity or loss of appropriate mitigating features.
- (C) Equipment necessary for achieving and maintaining safe shutdown of the plant and maintaining containment integrity will perform its safety function during and after being exposed to the environmental conditions attendant with the release of hydrogen generated by the equivalent of a 100-percent, fuel clad metal-water reaction including the environmental conditions created by activation of the hydrogen control system.
- (D) If the method chosen for hydrogen control is a post-accident inerting system, inadvertent actuation of the system can be safely accommodated during plant operation.

SSAR Section 6.2.5 discusses the provisions for combustible gas control within the ABWR containment, which include an inerted containment and inclusion of thermal hydrogen recombiners. In Sections 6.2.5 and 19.2.3.3.1 of this report, the staff concludes that these provisions for hydrogen gas control within the ABWR containment, for both design-basis accidents and severe accidents, are acceptable.

Generic Issues

Criteria (A) and (B) are met through an inerted containment which ensures that the post-accident containment atmosphere will not support hydrogen combustion. Criterion (C) is met as discussed in Section 19.2.3.3.7 of this report. Criterion (D) is not applicable to the ABWR design, as inerting is accomplished prior to the onset of the accident. The staff concludes that the ABWR design meets the requirements of 10 CFR 50.34(f)(2)(ix).

20.5.22 10 CFR 50.34(f)(2)(x): Reactor Coolant System Relief and Safety Valves - Testing Requirements (TMI Item II.D.1)

Paragraph (2)(x) of 10 CFR 50.34(f) requires that the licensee provide a test program and associated model development and conduct tests to qualify reactor coolant system relief and safety valves for all fluid conditions expected under operating conditions, transients, and accidents. The staff concludes that GE's approach adequately addresses the requirements of this TMI item for the ABWR design as discussed in Section 3.9.3.2 of this report.

20.5.23 10 CFR 50.34(f)(2)(xi): Reactor Coolant System Relief and Safety Valves - Relief and Safety Valve Position Indication (TMI Item II.D.3)

Paragraph (2)(xi) of 10 CFR 50.34(f) requires that the relief and safety valves for the RCS shall include a positive indication in the control room derived from a reliable valve-position detection device or a reliable indication of flow in the discharge pipe.

The ABWR SRVs are equipped with position sensors that will be qualified as Class 1E components. All SRV positions will be indicated in control room. The ABWR design also includes a backup method using thermocouples. The pipe downstream of the SRV is equipped with thermocouples which signal the annunciator and the plant process computer when the temperature in the tail pipe exceeds the predetermined set point.

The staff has reviewed the design, compared it with the position and clarification contained in NUREG-0737, "Clarification of TMI Action Plan Requirements," and determined that GE's response adequately addresses the requirements of this TMI item for the ABWR design as discussed in Section 5.2.2 of this report.

20.5.24 10 CFR 50.34(f)(2)(xii): System Design -- Auxiliary Feedwater System - Auxiliary Feedwater System Automatic Initiation and Flow Indication (TMI Item II.E.1.2)

Paragraph (2)(xii) of 10 CFR 50.34(f) requires the provision of automatic and manual AFWS initiation and the provision of AFWS flow indication in the control room of PWR plants.

This requirement is applicable to PWRs only, since a BWR design does not include an AFWS. Therefore, it is not technically relevant to the ABWR design and does not need to be addressed.

20.5.25 10 CFR 50.34(f)(2)(xiii): System Design -- Decay Heat Removal - Reliability of Power Supplies for Natural Circulation (TMI Item II.E.3.1)

Paragraph (2)(xiii) of 10 CFR 50.34(f) requires provision of a pressurizer heater power supply and associated motive and control power interfaces for PWR plants that is sufficient to establish and maintain natural circulation in hot standby conditions with only onsite power available.

This requirement is applicable to PWRs only, since a BWR design does not include a pressurizer. Therefore, it is not technically relevant to the ABWR design and does not need to be addressed.

20.5.26 10 CFR 50.34(f)(2)(xiv): System Design -- Containment Design - Isolation Dependability (TMI Item II.E.4.2)

Paragraph (2)(xiv) of 10 CFR 50.34(f) requires the provision of containment isolation systems that do the following:

- Ensure all non-essential systems are isolated automatically by the containment isolation signal
- Have two isolation barriers in series for each non-essential penetration, except instrument lines
- Do not result in reopening of the containment isolation valves on resetting of the isolation signal
- Utilize a containment set point pressure for initiating containment isolation as low as is compatible with normal operation

- Include automatic closing on a high radiation signal for all systems that provide a path to the environs.

The staff reviewed GE's containment isolation system for compliance with these requirements. The evaluation results and findings regarding each requirement are as follows:

- All non-essential systems are isolated automatically by the containment isolation system in accordance with the TMI Item II.E.4.2 requirements.
- As discussed in DFSER Sections 6.2.4 and 6.2.4.1, some penetrations do not have two isolation barriers in series that conform to the containment isolation requirements of GDC 56. This was identified in the DFSER as Open Item 20.3-4. (This issue was also identified in the DFSER as Open Item 6.2.4.1-1.) As discussed in Section 6.2.4.1 of this report, GE subsequently provided adequate justification for the alternative containment isolation design that resolved these open items. Therefore, DFSER Open Item 20.3-4 and DFSER Open Item 6.2.4.1-1 are resolved.
- Resetting the containment isolation signal will not result in the automatic reopening of containment isolation valves. The reopening of any containment isolation valve is on a valve-by-valve basis once the isolation signal has cleared and following subsequent logic reset.
- GE has committed to using a high drywell set point pressure of 14 kPag (2 psig) to isolate non-essential penetrations. This is a minimum value and compatible with normal operating conditions. The staff finds a set point value of 14 kPag (2 psig) acceptable.
- The containment purge and vent isolation valves isolate on high radiation levels in the reactor building HVAC air exhaust or in the fuel handling area HVAC air exhaust. As discussed in SSAR Section 6.2.4.1, the containment purge provision has not been found acceptable for ABWR.

Based on the above evaluation, the staff concludes that the ABWR design meets this TMI requirement.

20.5.27 10 CFR 50.34(f)(2)(xv): System Design -- Containment Design - Purging (TMI Item II.E.4.4)

Paragraph (2)(xv) of 10 CFR 50.34(f) requires provision of containment venting/purging capability to minimize purging time, consistent with the as-low-as-is-reasonably-achievable (ALARA) principles for occupational exposure.

It also requires assurance that the purge system will reliably isolate under accident conditions.

SSAR Section 19A.2.27 stated that during normal power operation, all large valves in the containment ventilation lines are closed and that only the 5-cm (2-in.) nitrogen makeup valves are open. These valves are characterized as air-operated valves with fast closure times which prevent substantial releases from containment should containment isolation be required. GE also stated that the 5-cm (2-in.) nitrogen bleed lines are sufficient to maintain normal containment pressure during normal operation when used in conjunction with containment spray and the drywell cooling system. In the DFSER the staff found that the use of the 5-cm (2-in.) nitrogen bleed lines for normal pressure control is consistent with ALARA considerations and there is high assurance that the purge system will reliably isolate under accident conditions.

SSAR Section 19A.2.27 also stated that the large ventilation valves will be tested not only on a regular basis, but also after any valve maintenance in order to ensure that closing times are within allowable limits. In the DFSER, the staff noted that these tests should include valve T31-F007, that the details of these tests should be submitted by the applicant referencing the ABWR design, and that this should be identified in the SSAR as a COL action item. This was identified in the DFSER as COL Action Item 20.3-2. Subsequently, GE modified SSAR Table 3.9-8, "Inservice Testing of Safety-Related Pumps and Valves", to include both T31-F007 and T31-F010, the isolation valves for the COPS rupture disk. The staff verified that GE established a COL action item to develop the inservice testing program for the components in Table 3.9-8. The test requirements in this table ensure that these valves will be adequately tested. This approach adequately addresses DFSER COL Action Item 20.3-2.

In response to the staff's request for additional information (RAI) dated June 5, 1990, GE stated that the isolation signal to valve T31-F007 would be deleted from SSAR Figure 6.2-39a. As of SSAR Amendment 11, this figure still contained an isolation signal to this valve, the DFSER identified this as Open Item 20.3-5. Subsequently, GE provided SSAR Figure 6.2-39, Sheet 1, which deleted the isolation signal. Therefore, DFSER Open Item 20.3-5 is resolved.

In another response to this RAI, GE agreed to amend the ABWR TS to allow a 24-hour (rather than a 72-hour) window at the beginning and end of a fuel cycle, during which the large diameter 56-cm (22-in.) purge lines can be open in accordance with the STS. This was identified in the DFSER as TS Item 6.2.5-1 and was discussed in Chapter 6 of the DFSER. In addition to proper operation

Generic Issues

during normal conditions, the staff stated in the DFSER that GE should provide justification that these valves will close during accident conditions. This was identified in the DFSER as Open Item 20.3-6. The staff has confirmed that SSAR Section 6.2.5 has been modified to clarify that these valves receive close signals during accident conditions. Therefore, DFSER Open Item 20.3-6 is resolved.

Based on the additional clarifying information provided in the SSAR, the staff concludes that the ABWR design meets this TMI requirement.

20.5.28 10 CFR 50.34(f)(2)(xvi): System Design -- Design Sensitivity of B&W Reactors - Design Evaluation (TMI Item II.E.5.1)

Paragraph (2)(xiv) of 10 CFR 50.34(f) requires establishment of a design criterion for the allowable number of actuation cycles of the ECCS and RPS of B&W plant designs consistent with the expected occurrence rates of severe overcooling events, considering both anticipated transients and accidents.

This requirement is applicable to B&W-designed plants only, therefore, it is not technically relevant to the ABWR design and does not need to be addressed.

20.5.29 10 CFR 50.34(f)(2)(xvii): Instrumentation and Controls - Additional Accident Monitoring Instrumentation (TMI Item II.F.1)

Paragraph (2)(xvii) of 10 CFR 50.34(f) requires provisions for instrumentation to measure, record, and read out in the control room containment pressure, water level, hydrogen concentration, radiation intensity (high level), and noble gas effluent at all potential accident release points. In addition, it requires continuous sampling of radioactive iodines and particulates in gaseous effluent from all potential accident release points, and an onsite capability to analyze and measure samples. Under TMI Item II.F.1, NUREG-0660 restates these requirements with additional guidance and clarification. NUREG-0660 calls for a human factors analysis, which is to include the use of the indicators listed above by the operator during both normal and abnormal plant conditions, integration of these indicators into plant emergency procedures and operator training, the use of other alarms during an emergency, and the need for prioritization of alarms.

In the DFSER, the staff stated that GE's responses to its RAI dated June 5, 1990, were still under review and that completion of this review was DFSER Open Item 20.3-7. SSAR Section 7.5 compares the ABWR design against the criteria of RG 1.97 (Rev. 3), addressing accident

monitoring instrumentation. Section 7.5 lists the variables that are considered essential safety-related information for the operators, and identifies specific exceptions to the guidance of RG 1.97. The capability to monitor the parameters required by the regulation is provided in the control room. Based on its review of SSAR Section 7.5, the staff concludes that the ABWR I&C design meets RG 1.97 as discussed further in Section 7.5.2 of this report and, therefore, also meets this TMI requirement. Therefore, DFSER Open Item 20.3-7 is resolved.

In the DFSER, the staff also reported that GE stated that the human factors aspects of this requirement were beyond the scope of the ABWR design certification review and the COL applicant will be responsible for addressing them in the detailed design implementation. This was identified in the DFSER as COL Action Item 18.7.2.2-3. The staff verified that GE established a COL action item (Item 18.8.1) in SSAR Section 18.8 for the detailed control room development as defined in ABWR CDM Table 3.1 ITAAC and in SSAR Section 18E. Further, GE established a COL action item (Item 18.8.13) in SSAR Section 18.8 to address II.F.1. This approach adequately addresses DFSER COL Action Item 18.7.2.2-3 as discussed in Section 18.7.2.2 of this report.

20.5.30 10 CFR 50.34(f)(2)(xviii): Instrumentation and Controls - Identification of and Recovery from Conditions Leading to Inadequate Core Cooling (TMI Item II.F.2)

Paragraph (2)(xviii) requires provision of instrumentation or controls (primary or backup) proposed for the plant to supplement existing instrumentation (including primary coolant saturation monitors) in order to provide an unambiguous, easy-to-interpret indication of inadequate core cooling (ICC). It also requires a description of the functional design requirements for the system, a description of the procedures to be used with the proposed equipment, the analysis used in developing these procedures, and a schedule for installing the equipment.

Reactor pressure vessel level is the only issue to be considered with respect to this issue since BWRs operate at saturation pressure and saturation monitors are not required for the ABWR. As in operating BWRs, the level instruments for the ABWR RPV are all delta p (dp) instruments. Each instrument uses a reference leg, which is maintained full by a condensing chamber connected directly to the steam space in the RPV, and uses a variable leg which is connected to the RPV water space. All the dp level instruments operate on the same physical principle. Therefore, common-cause failures caused by a design deficiency or maintenance error could result in inaccurate indication of reactor vessel water level.

The staff's concern about common-cause failures is based on experience with potential common-mode failure mechanisms in the reactor water level instruments. For example, during the past two years, anomalies have been observed in reactor vessel water level indication at several BWRs (Millstone-1, Pilgrim, LaSalle, and Washington Nuclear (WNP-2) during controlled depressurization to commence plant outages. These anomalies consisted of "spiking" or "notching" of level indication, and in one instance, a sustained error in level indication. The effect of noncondensable gas in the condensate chamber and reference leg of the cold-leg type of water level instruments has been determined to be the root cause of these level indication anomalies. Testing has shown that under depressurization conditions, noncondensable gases can cause significant errors in the level indication.

The ABWR design for the reactor vessel water level measurement system has the following features:

- (1) The ABWR has temperature-compensated RPV level indication for post-accident monitoring of the RPV.
- (2) The vertical drop in the drywell for the ABWR RPV water level reference leg instrument lines from the condensing chamber to the drywell wall has been limited to three feet.
- (3) The ABWR uses analog level transmitters.
- (4) The ABWR safety system utilizes two-out-of-four logic for the automatic safety systems initiated on RPV water level.

The staff reviewed the ABWR design for the reactor vessel water level measurement system and found that it meets the requirements specified in GL 84-23, "Reactor Vessel Water Level Instrumentation in BWRs," dated October 26, 1984. To ensure high functional reliability of the instrumentation, the staff issued GL 92-04, "Resolution of the Issues Related to Reactor Vessel Water Level Instrumentation in BWRs Pursuant to 10 CFR 50.54(f)," dated August 19, 1992, and Bulletin 93-03, "Resolution of Issues Related to Reactor Vessel Water Level Instrumentation in BWRs," dated May 28, 1993, requesting hardware modifications for operating reactors. In response to these generic communications, all BWR licensees have committed to implement hardware modifications to their level instrumentation systems. Similar cold-leg instruments are used in the ABWR design.

Before the noncondensable gas level inaccuracies, there were other problems with dp level instrumentation used in BWRs. The staff issued GL 84-23 to address concerns

related to high containment temperature during a depressurization event. High containment temperature combined with reactor depressurization can lead to false water level readings as a result of flashing or boiling in the reference leg within the containment.

In the DFSER, the staff stated that the design of the ABWR reactor vessel water level measurement system met the requirements specified in GL 84-23. However, the staff required GE to discuss GL 92-04 in the SSAR and include any design changes necessary to preclude the potential for false reactor coolant level readings. The staff also required GE to determine if compliance with this TMI requirement was affected by any design changes. This was identified in the DFSER as Open Item 20.3-8. In response to these generic communications, GE changed the design of the ABWR RPV level instrumentation system. GE incorporated a backfill modification system that will constantly purge the reference leg with a very low flow rate of water supplied by the CRD system. The constant flow of water up the reference leg will prevent dissolved gases from migrating down the reference leg.

The known common-mode deficiencies in BWR level instrumentation systems have been addressed at operating BWRs and by GE in the ABWR design. It should also be noted that these particular deficiencies would not have compromised the automatic protective functions of the level instrumentation for accident scenarios initiated while at power, and that no previous incidents at BWRs of inaccurate level indication have been misinterpreted by plant operators so as to lead to unsafe actions. The staff concluded in the advance SER that the ABWR level instrumentation system without the proposed level diversity meets the requirements of 10 CFR Part 50. However, in view of the importance of level instrumentation for safety in BWRs, and the experience discussed above where the potential existed to fail redundant level instruments due to a common cause, the staff believes that the addition of level instrumentation which operates on a diverse physical principle is desirable and prudent for the purpose of guiding operator emergency actions.

GE did not agree with the staff recommendation for diverse water level instrumentation and presented its position in a letter dated October 26, 1993. As part of the letter, GE presented the following summary:

ABWR water level instrumentation is rugged, simple and highly redundant for failure tolerance. All known operating problems have been addressed in this design and it is incredible to postulate simultaneous common-mode failures which would yield identical errors in all the dp instrumentation.

Generic Issues

Alternate technologies are unqualified for this application; further, there is no need to add this complexity, since the plant operating staff has ample additional indications of an impending problem without relying solely on water level. The EPGs direct the operator to use all information available to him and make conservative (safe) decisions.

In the attachment to the letter, GE also provided a list of indications of inadequate RPV water level which are independent of the dp RPV water level instrumentation. The staff recognizes that other parameters could aid the operator in assessing the adequacy of core cooling under accident conditions. These include instrumentation for indication of reactor power, core neutron flux, the recirculation flow control system response, and feedwater flow and steam flow mismatch. However, the staff believes that these indications could be easily misinterpreted or could be insufficient because they are only indirect methods of inferring reactor water level or core cooling.

Other evolutionary designs, such as the ABB-Combustion Engineering, Inc. (ABB-CE) System 80+, provide diverse methods of RPV level measurement. The inadequate core cooling instrumentation package in the CE System 80+ plant includes reactor vessel level monitoring system probes employing both dp sensors and the heated junction thermocouple concept. The staff is aware of a diverse method of level monitoring that is currently in use in at least one nuclear power plant in Germany employing ultrasonic measurement techniques. In addition, a diverse level measurement system which uses heated junction thermocouples has been in use for the past 5 years at a Swedish BWR, and another Swedish BWR uses float switches for diverse level indication and automatic systems actuation. Other Swedish BWRs have decided in principle to install diverse level measurement systems.

The staff indicated in the advance SER that the diverse method of level measurement is recommended for indication in the control room only (there is diverse instrumentation, namely high drywell pressure, in both the operating BWRs and the ABWR design which provides diverse signals for automatic safety systems actuation for many event scenarios). This would provide a direct and back-up means for the operator to identify inadequate core cooling and to take appropriate manual actions to initiate and control safety systems as identified in the plant EOPs. The staff also recommended that the diverse level measurement device be reliable, redundant, and capable of

being powered by onsite power sources. This was Open Item F20.5.30-1 in the advance SER.

The staff issued the draft Commission paper, "Diversity in the Method of Measuring Reactor Pressure Vessel Level in the Advanced Boiling Water Reactor and Simplified Boiling Water Reactor," on November 15, 1993, for public and industry comments. The ACRS discussed the issue in its 430th meeting on December 9 through 11, 1993, and sent its recommendation to the Commission in a letter dated December 16, 1993. The ACRS did not support the staff recommendation on diversity. Based on ACRS deliberations and GE's position, the staff reconsidered the need for the requirement for instrumentation diversity.

All the known common-mode deficiencies in BWR level instrumentation systems have been addressed by GE in the ABWR design. It should also be noted that these deficiencies would not have compromised the automatic functions of the level instrumentation for accident scenarios initiated while at power, and that no previous incidents at BWRs of inaccurate level indication have been misinterpreted by plant operators so as to lead to unsafe actions. In addition, for many events, the ECCS is started in the ABWR on high drywell pressure, as well as low reactor water level, thus providing some diversity. The ABWR EPGs will be used to develop the EOP that will be used with the reactor water level instrumentation.

Even though it may be desirable to provide instrumentation diversity in the ABWR design, there is not sufficient basis to postulate an unidentified potential common-mode failure. Further, diverse level measurement devices have not been demonstrated to be adequate. In light of the enhanced LOCA response in the ABWR and the guidance provided in the ABWR EPGs to address the use of the RPV instrumentation, the staff concludes that reactor vessel level instrumentation diversity is not required for the ABWR. On the basis of the above discussion, Open Item F20.5.30-1 (DFSER Open Item 20.3-8) is resolved. Based on the above discussion, the staff also concludes that the ABWR RPV level instrumentation will provide adequate indication of ICC as required by this TMI item.

In the DFSER the staff reported that GE stated that the human factors aspects of this requirement were beyond the scope of the ABWR design certification review and the COL applicant will be responsible for addressing them in the detailed design implementation. This was identified in the DFSER as COL Action Item 18.7.2.2-3. The staff verified that GE established a COL action item (Item 18.8.1) in SSAR Section 18.8 for the detailed control room development as defined in ABWR CDM Table 3.1 ITAAC and in SSAR Section 18E. Further, GE established a COL action item (Item 18.8.14) in SSAR

Section 18.8 to address II.F.2. This approach adequately addresses DFSEER COL Action Item 18.7.2.2-3 as discussed in Section 18.7.2.2 of this report.

20.5.31 10 CFR 50.34(f)(2)(xix): Instrumentation and Controls - Instruments for Monitoring Accident Conditions (TMI Item II.F.3)

Paragraph (2)(xix) of 10 CFR 50.34(f) requires instrumentation adequate for monitoring plant conditions following an accident that includes core damage.

SSAR Section 7.5 compares the ABWR design against the criteria of RG 1.97 (Rev. 3), addressing accident monitoring instrumentation. Section 7.5 lists the variables that are considered essential safety-related information for the operators, and identifies specific exceptions to the guidance of RG 1.97. The list incorporates adequate monitoring capability for post-accident plant conditions that include core damage, including reactor pressure, water level and temperature, containment pressure, temperature and radiation level, and shutdown operation status. Based on its review of SSAR Section 7.5, the staff concludes that the ABWR I&C design meets RG 1.97 as discussed further in Section 7.5.2 of this report and, therefore, also meets this TMI requirement.

20.5.32 10 CFR 50.34(f)(2)(xx): Electrical Power - Power Supplies for Pressurizer Relief Valves, Block Valves, and Level Indicators (TMI Item II.G.1)

Paragraph (2)(xx) of 10 CFR 50.34(f) requires power supplies for pressurizer relief valves, block valves, and level indicators for PWRs such that the level indicators are powered from vital buses, motive and control power connections to the emergency power sources are thorough devices qualified in accordance with requirements applicable to systems important to safety, and electric power is provided from emergency power sources.

This requirement is applicable to PWRs only, therefore, it is not technically relevant to the ABWR design and does not need to be addressed.

20.5.33 10 CFR 50.34(f)(2)(xxi): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- IE Bulletins - Describe Automatic and Manual Actions for Proper Functioning of Auxiliary Heat Removal Systems When Feedwater System Not Operable (TMI Item Issue II.K.1.(22))

Paragraph (2)(xxi) of 10 CFR 50.34(f) requires design of auxiliary heat removal systems such that necessary

automatic and manual actions can be taken to ensure proper functioning when the main feedwater system is not operable.

SSAR Section 7.7.1.4(9) states that if the ABWR main feedwater system is not operating, the reactor will be tripped automatically when reactor water level falls to reactor vessel level 3. The operator can manually initiate the RCIC system from the main control room or, if the operator takes no action, reactor water level will continue to decrease as the steam boils off until the low-low level set point (level 2) is reached. At low-low level, the RCIC system is automatically initiated to supply makeup water to the reactor pressure vessel. If level 1.5 is reached, both HPCF pumps will start automatically. These systems will continue automatic injection until the reactor water level reaches level 8, at which time the HPCF and RCIC systems are tripped. The HPCF will restart automatically once the high-level trip signal clears and the level 1.5 signal is received. The RCIC will automatically start after a level 8 trip and a level 2 signal.

If the vessel is isolated, reactor vessel pressure is regulated by automatic or remote manual operation of the main steam relief valves, which exhaust to the suppression pool. In this case, the automatic suppression pool cooling mode of the RHR system will transfer heat to the ultimate heat sink.

The HPCF system will automatically provide the required makeup flow for the accident situations with the reactor vessel at high pressure. No manual operations are required since the HPCF system will cycle on and off automatically as water level reaches vessel level 1.5 and level 8, respectively. If the HPCF system fails under these conditions, the operator can manually depressurize the reactor vessel using the ADS to permit the low-pressure emergency core cooling systems to provide makeup coolant. The RCS will automatically be depressurized if all of the following signals are present: high drywell pressure, level 1 water level, and pressure in at least one low-pressure injection system or one HPCF system. If the low level persists in the RPV, the ADS will also activate without drywell high pressure after a delay of about 8 minutes.

The ABWR design incorporates appropriate automatic and manual action capability to ensure proper heat removal when the main feedwater system is not operable. Therefore, the staff concludes that GE has adequately addressed the requirements of this TMI item for the ABWR design.

Generic Issues

20.5.34 10 CFR 50.34(f)(2)(xxii): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Commission Orders on B&W Plants - Analysis and Upgrading of Integrated Control System (TMI Item II.K.2(9))

Paragraph (2)(xxii) of 10 CFR 50.34(f) requires performance of a failure modes and effects analysis of the integrated control system (ICS) of B&W-designed plants to include consideration of failures and effects of input and output signals to the ICS.

This requirement is applicable to B&W-designed plants only, therefore, it is not technically relevant to the ABWR design and does not need to be addressed.

20.5.35 10 CFR 50.34(f)(2)(xxiii): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Commission Orders on B&W Plants - Hard-Wired Safety-Grade Anticipatory Reactor Trips (TMI Item II.K.2(10))

Paragraph (2)(xxiii) of 10 CFR 50.34(f) requires provision, as part of the RPS of B&W-designed plants, an anticipatory reactor trip that would be actuated on loss of main feedwater and on turbine trip.

This requirement is applicable to B&W-designed plants only, therefore, it is not technically relevant to the ABWR design and does not need to be addressed.

20.5.36 10 CFR 50.34(f)(2)(xxiv): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Central Water Level Recording (TMI Item II.K.3(23))

Paragraph (2)(xxiv) of 10 CFR 50.34(f) requires the capability to record reactor vessel water level in one location on recorders that meet normal post-accident recording requirements for BWRs.

SSAR Section 7.5.8 states that the reactor vessel water level wide range instruments and fuel zone instruments are utilized to provide post-accident monitoring indication. The four divisions of wide range level instruments cover the zone from above the core to the main steam lines. The two channels of fuel zone instruments cover the range from below the core to the top of the steam separator shroud.

The SSAR also states that in the event that the vessel water level is below the range of the wide range level sensor and the two channels of fuel zone level instrumentation disagree, the EOPs instruct the operator to use the lower level indicated in the two channels and to return the water level to the range of the wide range level instrumentation. GE adds that by using the four divisions of wide range level instruments, an unambiguous indication of vessel water level can be determined despite a postulated failure of a single instrument channel or division, thereby permitting the operator to implement the EOPs.

The staff concludes that GE's approach acceptably meets the requirements of this TMI item regarding post-accident reactor vessel water level recording for the ABWR design.

20.5.37 10 CFR 50.34(f)(2)(xxv): Emergency Preparedness and Radiation Effects -- Improve Licensee Emergency Preparedness - Short Term; Upgrade Emergency Preparedness (TMI Item III.A.1.2)

Paragraph (2)(xxv) of 10 CFR 50.34(f) requires the provision of an onsite Technical Support Center (TSC), an Operational Support Center (OSC), and a nearsite Emergency Operations Facility (EOF).

GE's proposal is discussed and evaluated in Section 13.3 of this report. The staff verified that GE established a COL action item in SSAR Section 19A.3.4 to detail a TSC and an OSC in the ABWR service building. This SSAR section also states that the COL applicant has an interface requirement to provide a near site emergency operational facility. On the basis of the staff's evaluation in Section 13.3 of this report and GE's establishment of the COL action item, the staff concludes that GE has adequately addressed the requirements of this TMI item for the ABWR design.

20.5.38 10 CFR 50.34(f)(2)(xxvi): Radiation Protection -- Radiation Source Control - Primary Coolant Sources Outside the Containment Structure (TMI Item III.D.1.1)

Paragraph (2)(xxvi) of 10 CFR 50.34(f) requires provision of leakage control and detection in the design of systems outside containment that contain (or might contain) TID-14844 source term radioactive materials following an accident. Applicants are required to submit a leakage control program, including an initial test program, a schedule for retesting these systems, and the actions to be taken for minimizing leakage from such systems. The goal is to minimize potential exposures to workers and the public and to provide reasonable assurance that excessive

leakage will not prevent the use of systems needed in an emergency.

The advance SER stated that the staff reviewed SSAR Section 1A.2.34, as well as GE's response (see page 20.3.15-22 of the SSAR) to staff Question 430.227 relating to this TMI item. It also stated that these references identify the applicable systems and the leak reduction measures for the ABWR design. Based on its review, the staff found in the advance SER that the leakage control program for the systems outside the containment for the ABWR design would include periodic leak testing and leak reduction measures for eight listed systems. GE revised and expanded the list of systems in Section 1A.2.34 to include the following: (1) RHR, (2) HPCF, (3) low pressure core flooders, (4) RCIC, (5) suppression pool cleanup, (6) reactor water cleanup, (7) fuel pool cooling and cleanup, (8) post-accident sampling, (9) process sampling, (10) containment atmospheric monitoring, (11) fission product monitor (part of leakage detection system), (12) hydrogen recombiner, and (13) standby gas treatment.

In the advance SER, the staff also found that the COL applicant will be required to develop plant procedures that will prescribe the leak testing methods for the above systems and schedule maintenance programs to monitor leakages and reduce detected leakages to lowest practical levels. GE stated that the leak tests will be conducted periodically at each refueling outage. Furthermore, the staff found that GE identified Item 5.5.2.2 in the administrative controls section of the ABWR TS, which calls for the COL applicant to establish a leakage control program for the subject systems. The TS item further stipulates that the program include preventive maintenance and periodic visual inspection requirements and integrated leak test requirements for each system at refueling cycle intervals or less.

Based on the above, the staff concludes that the ABWR design adequately addresses the requirements of this TMI item for the ABWR design.

The staff will review and evaluate the individual leakage control programs, including the initial test programs, of the COL applicant on a plant-specific basis.

20.5.39 10 CFR 50.34(f)(2)(xxvii): Radiation Protection -- Worker Radiation Protection Improvement - Inplant Radiation Monitoring (TMI Item III.D.3.3)

Paragraph (2)(xxvii) of 10 CFR 50.34(f) requires provisions for monitoring of inplant radiation and airborne

radioactivity as appropriate for a broad range of routine and accident conditions.

GE's response adequately addresses the requirements of this TMI item for the ABWR design as discussed further in Section 12.5.1 of this report.

20.5.40 10 CFR 50.34(f)(2)(xxviii): Radiation Protection -- Worker Radiation Protection Improvement - Control Room Habitability (TMI Item III.D.3.4)

Paragraph (2)(xxviii) of 10 CFR 50.34(f) requires an evaluation of the potential pathways for radioactivity and radiation that may lead to control room habitability problems under accident conditions resulting in a TID-14844 source term release, and to make necessary design provisions to preclude such problems.

The staff's review of control room habitability is provided in Section 6.4 of this report. The staff concludes that the ABWR control room design provides an acceptable means of maintaining the control room in a safe and habitable condition by providing adequate protection under accident conditions in accordance with the requirements of this TMI item.

20.5.41 10 CFR 50.34(f)(3)(i): Operating Procedures - Procedures for Feedback of Operating Experience to Plant Staff (TMI Item I.C.5)

Paragraph (3)(i) of 10 CFR 50.34(f) requires the provisions for administrative procedures for evaluating operating, design, and construction experience and for ensuring that applicable important industry experiences will be provided in a timely manner to those designing and constructing the plant.

The development of detailed procedures is beyond the scope of the ABWR design certification review and the COL applicant will be responsible for addressing this TMI item. The staff verified that GE established a COL action item in SSAR Section 13.5.3 for procedure development and a COL action item in SSAR Section 13.2 regarding the incorporation of operating experience into training programs. Additionally, SSAR Section 13.5.3.1 states that the methods and criteria for the development, V&V, implementation, maintenance, and revision of procedures will include considerations of I.C.5. This approach is acceptable to the staff as discussed in Sections 13.5 and 13.2 of this report.

Generic Issues

20.5.42 10 CFR 50.34(f)(3)(ii): Quality Assurance (QA) - Expand QA List (TMI Item I.F.1)

NUREG-0660 states that several systems important to safety of the TMI plant were not designed, fabricated, and maintained at a level equivalent to their safety importance. In accordance with the requirements of Criterion 1 of Appendix A to 10 CFR Part 50 and the guidance provided in SRP Section 17.3, "Quality Assurance Program Description," applicants are to include these types of non-safety-related items within the scope of their QA programs. This requirement was incorporated into Paragraph (3)(ii) of 10 CFR 50.34(f).

SSAR Table 3.2-1, "Classification Summary," identifies safety-related and non-safety-related items of the ABWR. Note "e" in the table and SSAR Section 3.1.2.1.1.2 state

- (1) the total QA program described in SSAR Chapter 17 is applied to the safety-related items
- (2) the non-safety-related items are controlled by the QA program described in SSAR Chapter 17 in accordance with the functional importance of the item

The staff concludes that GE's approach adequately addresses the requirements of this TMI item for the ABWR design.

20.5.43 10 CFR 50.34(f)(3)(iii): Quality Assurance - Develop More Detailed QA Criteria (TMI Item I.F.2)

Paragraph (3)(iii) of 10 CFR 50.34(f) requires the establishment of a quality assurance program. NUREG-0660, Item I.F.2 lists 11 considerations that the NRC should use to develop additional guidance to clarify requirements for the QA function. These considerations resulted in Revision 2 of SRP Section 17.1, "Quality Assurance During the Design and Construction Phases," in July 1981.

SSAR Chapter 17 references NEDO-11209, "Nuclear Energy Business Operation Quality Assurance Program Description," Revision 7 dated May 1987. The staff reviewed this report against the acceptance criteria of Revision 2 of SRP Section 17.1 and found it to be acceptable. The staff concludes that GE's approach adequately addresses the requirements of this TMI item for the ABWR design.

20.5.44 10 CFR 50.34(f)(3)(iv): Consideration of Degraded or Melted Cores in Safety Review - Rulemaking Proceeding on Degraded Core Accidents (TMI Item II.B.8), ".91-Meter (3-Footer) Diameter Equivalent Dedicated Containment Penetration"

Paragraph (3)(iv) of 10 CFR 50.34(f) requires one or more dedicated containment penetrations, equivalent in size to a single .91-m (3-ft) diameter opening, in order not to preclude future installation of systems to prevent containment failure, such as a filtered vented containment system. This requirement is intended to ensure provision of a containment vent design feature with sufficient safety margin well ahead of a need that may be perceived in the future to mitigate the consequences of a severe accident situation. The staff's evaluation of ABWR compliance with the requirement is limited to the effective penetration size for venting provided in the ABWR primary containment design.

In the DFSER, the staff found that the size of the primary containment penetration that could be used during a severe accident situation for venting the containment was smaller than the specific size identified in the TMI requirement. The staff required GE to submit a request for exemption from the requirement and supporting justification. The justification was expected to demonstrate that the penetration size was adequate to permit a vent relief path that is capable of providing the needed overpressure relief for the primary containment to prevent its uncontrolled failure during any credible severe accident situation. This was identified in the DFSER as Open Item 20.3-9.

SSAR Section 19A.2.44 states that the COPS precludes the need for a dedicated penetration equivalent in size to a single .91-m (3-ft) diameter opening. The COPS is part of the atmospheric control system and is discussed in SSAR Section 6.2.5.6. The COPS consists of two 200-mm (8-in.) diameter rupture disks mounted in series in a 250-mm (10-in.) line and is sized to allow 35 kg/sec (15.86 lbm/sec) of steam flow at the opening pressure of 6.3 kg/cm²g (90 psig), which corresponds to an energy flow of about 2.4 percent of rated power. The SSAR states that the COPS is capable of keeping containment pressures below ASME Service Level C limits for an ATWS event with failure of the SLCS and containment heat removal systems.

In Section 19.2.3.3.4 of this report, the staff concludes that the COPS design is acceptable. Although the diameter of the COPS pathway is only 200 mm (8 in.), the staff determined that this exception from the requirement of a .91-m (3-ft) diameter opening is acceptable since: (1) the limiting diameter of the COPS pathway is adequate to permit the needed vent relief path, and (2) a need for venting capability beyond that provided by the COPS has not been identified. This resolved DFSER Open Item 20.3-9. The staff concludes that GE's approach adequately addresses the requirements of this TMI item for the ABWR design. An exemption in accordance with 10 CFR 50.12(a)(2)(ii) is justified since the COPS provides sufficient venting capability to preclude the need for a .91 m (3-ft) diameter equivalent dedicated containment penetration. This exemption is in order based on the staff's conclusion that the underlying purpose of the regulation has been met.

20.5.45 10 CFR 50.34(f)(3)(v): Consideration of Degraded or Melted Cores in Safety Review - Rulemaking Proceeding on Degraded Core Accidents (TMI Item II.B.8), "Containment Integrity During an Accident Involving 100-Percent, Fuel Clad Metal-Water Reaction"

Paragraph (3)(v) of 10 CFR 50.34(f) requires design information to demonstrate that:

- (1) Containment integrity will be maintained during an accident that releases hydrogen generated from a 100-percent, fuel clad metal-water reaction accompanied by either hydrogen burning or the added pressure from post-accident inerting, assuming carbon dioxide is the inerting agent. Systems necessary to ensure containment integrity shall also be demonstrated to perform their function under these conditions.
- (2) Containment structure loadings produced by an inadvertent full actuation of a post-accident inerting hydrogen control system, but not including seismic or design-basis accident loadings will not produce stresses in steel containments in excess of the limits set forth in the ASME Code.

SSAR Section 19E.2.3.2 provides an evaluation of the capability of the containment to withstand pressurization from a 100-percent, fuel clad metal-water reaction. The staff finds the evaluation acceptable in Section 19.2.3.3.1 of this report. The staff evaluation of GE's assessment of systems necessary to ensure containment integrity under these conditions is provided in Section 19.2.3.3.7 of this report. Criterion (2) is not applicable to the ABWR design, as inerting is accomplished prior to the onset of the

accident. On this basis, the staff concludes that GE's approach adequately addresses the requirements of this TMI item for the ABWR design.

20.5.46 10 CFR 50.34(f)(3)(vi): System Design -- Containment Design - Dedicated Penetrations (TMI Item II.E.4.1)

Paragraph (3)(vi) of 10 CFR 50.34(f) requires redundant, dedicated containment penetrations for plant designs with external hydrogen recombiners so that, assuming a single active failure, the recombiner systems can be connected to the containment atmosphere.

SSAR Section 1A.2.13 states that the flammability control system (FCS) uses two permanently installed recombiners located in secondary containment which ensure that the FCS remains operable assuming a single active failure. In the DFSER, the staff noted that the FCS was not described in SSAR Section 6.2.5 and that GE had not demonstrated that redundant dedicated containment penetrations existed for the hydrogen recombiners. The staff required GE to provide

- information to clearly demonstrate that the permanently installed hydrogen recombiners have redundant, dedicated containment penetrations and that the penetrations meet all applicable design requirements. This information was to include
 - how long after a LOCA and at what hydrogen concentration the recombiners are to be utilized
 - line sizes as related to flow requirements
 - duration of recombiner operation
 - interface requirements for referencing applicants with regard to the recombiners
- a clearer copy of SSAR Figure 6.2-40, "Flammability Control System."

This was identified in the DFSER as Open Item 20.3-10. GE provided a new, clearer copy of Figure 6.2-40 and modified SSAR Section 6.2.5 to clarify FCS operation. Based on this additional information, the staff concludes that the FCS can adequately accommodate the effects of hydrogen and mitigate the consequences of hydrogen generated as a result of a LOCA, assuming a single active component failure. Therefore, Open Item 20.3-10 is resolved. The staff concludes that the ABWR design meets the requirements of this TMI item.

Generic Issues

20.5.47 10 CFR 50.34(f)(3)(vii): General Implications of TMI for Design and Construction Activities — Management for Design and Construction - Organization and Staffing to Oversee Design and Construction (TMI Item II.J.3.1)

Paragraph (3)(vii) of 10 CFR 50.34(f) requires provision of a description of the management plan for design and construction activities that includes:

- the organizational and management structure singularly responsible for direction of design and construction of the proposed plant
- technical resources directed by the applicant
- details of the interaction of design and construction within the applicant's organization and the manner by which the applicant will ensure close integration of the architect engineer and the nuclear steam supply vendor
- proposed procedures for handling the transition to operation
- the degree of top-level management oversight and technical control to be exercised by the applicant during design and construction, including the preparation and implementation of procedures necessary to guide the effort.

The development of the management plan for organization and staffing to oversee design and construction is beyond the scope of the ABWR design certification review and the COL applicant will be responsible for addressing it. The staff verified that GE established a COL action item in SSAR Section 19A.3.7 to develop the necessary management plan. This approach is acceptable to the staff.

20.6 Generic Communications

As part of its program to disseminate information on operating experience to the industry, the NRC issues generic communications when a significant safety-related event or condition at one facility is believed to potentially apply to other facilities. Using the basic criteria of safety significance and generic implications, many safety issues have been highlighted in generic communications. These generic communications encompass and address both staff positions (in the form of bulletins and most generic letters) and information alerts (in the form of information notices, circulars, and some generic letters). Potential concerns addressed initially by these generic communications may be subsequently revised or amplified. The resolution of these concerns may be incorporated into formal regulatory

requirements, such as rules in 10 CFR, or an analysis of such a concern may result in it becoming a USI or GSI.

The staff reviewed the ABWR design for incorporation of important lessons learned from operating experience using NRC bulletins and generic letters. These two classes of documents are used to communicate staff positions on issues potentially affecting operating facilities, thereby ensuring consideration of those issues judged to have significant public health and safety implications. The issues covered by bulletins and generic letters originate in a number of ways, including the staff's systematic review of operating experience. In the context of the NRC program to review and incorporate operating experience, bulletins and generic letters convey the most safety-significant lessons distilled from numerous sources of information. As a contrast, another product of that program, information notices, do not contain any requests for action on the part of licensees. Thus, bulletins and generic letters comprise a sufficient basis for reviewing the ABWR design against operating experience.

In SSAR Section 1.8, GE identifies the experience information that has been or will be included in the design of the ABWR. Experience information is routinely made available and distributed to design personnel. In addition, as a focused effort for the ABWR design activities, GE management surveyed a listing of all regulatory reports in its possession that contain operating experience information. If GE determined a report did not apply to the ABWR, the report was set aside. Each of the remaining reports on the list of potentially applicable experiences was then reviewed individually to determine technical applicability. As a result of this effort, GE prepared a listing of applicable regulatory documents (that is, information notices, generic letters, bulletins, and NUREGs) issued in 1980 or later. Some documents that addressed resolutions of concerns were not included on the resulting summary list to avoid repetition. In addition to these regulatory documents which are publicly available, GE has a collection of in-house proprietary documents that it has prepared over the years as part of its continual and ongoing assistance to various BWR licensees. GE also considered pertinent information from these experience reports.

Although the SRP provides acceptance criteria (some of which are based upon operating reactor experience) for review of a reactor facility design, this document was last revised in 1981. Therefore, potential concerns addressed in generic communications issued subsequent to the issue period of the SRP must be addressed. To ensure the staff's operating experience review is comprehensive, operating experience documents issued in 1980 and later were considered by the staff for applicability to the ABWR

design. Because the review of operating experience should include those items that are believed to have the highest safety significance, the staff limited its augmented review to those documents that rose to the status of a bulletin or a generic letter with a staff position or a request for licensee action. The last bulletin and generic letter considered in the review were NRC Bulletin 91-01 (original issuance) and GL 91-17, respectively. This consists of approximately 450 documents. Approximately 220 of these documents were excluded because they were not pertinent to the design review of the ABWR. The remaining list of 230 documents was augmented by any additional bulletins and generic letters GE thought applicable. About 160 on the combined listing were related to GSIs, USIs, TMI item implementation, or an NRC regulation. Operating experience concerns related to these issues are required to be explicitly addressed by GE and acceptability of the ABWR design with respect to these concerns is discussed in appropriate sections of this report. Thus, the staff excluded the disposition of these concerns from the task of ensuring and addressing how operating experience was considered in the ABWR design.

The remaining 71 applicable documents identify undesirable situations that have occurred and should be avoided and that could affect ABWR issues related to either equipment design, analytical methods, construction, operation or maintenance activities, or major programmatic activities. The staff reviewed these documents to determine if additional action is necessary to ensure that this experience has been reflected in the ABWR design and properly resolved. Of the 71 documents considered by the staff, 31 addressed issues that are being resolved during the ongoing preparation of TS or during a future equipment procurement process, and 27 had been superseded by technical developments after the document was issued. The remaining 13 documents raised questions that were considered by the staff in their overall evaluation process of the ABWR design.

Based on the above, the staff concludes that GE has adequately considered operating experience identified by generic letters or bulletins issued since 1980 in the ABWR design.

21 REPORT OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

The advanced boiling water reactor (ABWR) Subcommittee of the Advisory Committee on Reactor Safeguards (ACRS) conducted a review of GE Nuclear Energy's (GE's) application for design certification, which began in February 1988 and was completed in April 1994. The Subcommittee conducted a limited review of GE's certified design material (CDM) and a complete review of the various draft versions of the staff's safety evaluation reports on the ABWR standard design. The ACRS conducted its final meeting on the ABWR during its 408th meeting on April 7 and 8, 1994, and subsequently issued its letter regarding the ABWR on April 14, 1994. This letter, which follows this discussion, reflects approval of the application and includes no recommended actions by either the staff or the applicant. During the full committee meeting held on

March 10 through 12, 1994, the Chairman of the ABWR Subcommittee indicated that GE had provided him with an extensive set of draft revisions and markups to the standard safety analysis report (SSAR) and CDM in response to ACRS concerns, which were found to be acceptable but had not yet been incorporated into an SSAR amendment or final CDM submittal. The staff agreed to review GE's SSAR Amendment 34 and the associated CDM revision to ensure that the proposed revisions were included in GE's submittals. Accordingly, a review of Amendments 34 and 35, and the associated CDM was conducted and the staff determined that GE has adequately included the technical and editorial changes in its application for design certification in response to the previous ABWR Subcommittee concerns.

Report of the Advisory Committee on Reactor Safeguards

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

April 14, 1994

The Honorable Ivan Selin
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Selin:

SUBJECT: REPORT ON SAFETY ASPECTS OF THE GENERAL ELECTRIC NUCLEAR
ENERGY APPLICATION FOR CERTIFICATION OF THE ADVANCED
BOILING WATER REACTOR DESIGN

During the 408th meeting of the Advisory Committee on Reactor Safeguards, April 7-8, 1994, we completed our review of the General Electric Nuclear Energy (GENE) application for certification of its U.S. version of the Advanced Boiling Water Reactor (ABWR) standard design. This final report is intended to fulfill the requirement of 10 CFR 52.53 that the ACRS "... report on those portions of the application which concern safety." During our review we had the benefit of discussions with representatives of GENE and the NRC staff. We also had the benefit of the documents referenced.

ABWR Application

The U.S. version of the ABWR standard design utilizes a significant portion of the detailed design information developed jointly by GENE, Hitachi, and Toshiba for the international version which is being built in Japan. The application for certification of the U.S. version was filed by GENE in September 1987 under the provisions of Appendix O to 10 CFR Part 50 and the NRC Policy Statement on Nuclear Power Plant Standardization (Ref. 1). The application was docketed in February 1988. In December 1991, GENE requested that the application be considered under 10 CFR 52.45. This request was made effective in March 1992.

The application is based on the ABWR Standard Safety Analysis Report (SSAR), which was submitted in modular form between September 1987 and March 1989. Since then it has been amended frequently, the last submittal for our review was Amendment 34 in March 1994. The application also includes the ABWR Certified Design Material (CDM). The CDM contains the design information from the SSAR that will become a part of the design certification rule. The CDM has been revised, the last submittal that we received was Rev. 2 in December 1993.

The Honorable Ivan Selin

2

April 14, 1994

ABWR Design Description

The ABWR is a forced circulation boiling water reactor with a rated power of 3926 MWt. The reactor core consists of 872 8x8 fuel assemblies and 205 control rods. The reactor utilizes internal recirculation pumps and fine-motion control rod drives. It is located inside a steel-lined reinforced concrete pressure suppression containment which is enclosed by a reinforced concrete secondary containment, both of which are located in the Reactor Building. The Reactor Building also houses a standby gas treatment system, refueling area, main steam pipe tunnel, and essential systems for emergency core cooling, AC power (including diesel generators), and environmental conditioning.

The Control Building is located between the Reactor Building and the Turbine Building. The Control Building houses a continuation of the main steam pipe tunnel, the main control room, a computer facility, and essential systems for DC power, environmental conditioning, and cooling water. During emergencies, technical support is provided by the Technical Support and Operational Support Centers, which are located in the Service Building, which is immediately adjacent to the Control Building.

The Turbine Building houses equipment for power generation. Steam is supplied to an 1800 rpm turbine-generator which is oriented to minimize damage to safety-related equipment should a turbine failure occur. The Turbine Building also houses systems and equipment that provide various nonessential services for the plant. These include the standby combustion-gas-turbine generator, house boiler, air compressors, and systems for AC and DC power and environmental conditioning.

The Radwaste Building houses equipment for the collection and processing of radioactive waste generated by the plant. An underground pipe tunnel connects the Turbine and Reactor Buildings to the Radwaste Building.

The ABWR design includes a number of features that we believe will enhance safety relative to past BWR designs. Some of these features resulted from the use of PRA methodology by GENE in evaluating the ABWR design as it progressed.

- The use of reactor internal pumps removes the large reactor recirculation piping and connections to the reactor vessel, thereby reducing the size of the largest loss-of-coolant accident (LOCA).
- The use of a fine-motion control rod drive arrangement removes the scrap discharge volume and associated piping, provides two reliable means for inserting the rods, and is intended to eliminate the rod drop and rod ejection accidents.

Report of the Advisory Committee on Reactor Safeguards

The Honorable Ivan Selin

3

April 14, 1994

- The Emergency Core Cooling System and supporting auxiliaries are arranged into three physically separated electrical and mechanical divisions, only one of which is needed for handling transients and virtually all accidents.
- A combustion-gas-turbine generator is provided for enhanced on-site AC power capability.
- An AC-independent reactor water addition feature, a depressurization system, lower drywell flooders, cavity floor spreading area, sacrificial layer of basaltic concrete, and containment overpressure protection system are provided to mitigate severe accidents.
- The greatly increased application of digital control systems offers the potential for improved operator interface with the plant and the reliability of control and protection systems. In addition, the use of digital multiplexers and fiber optics reduces the amount of cabling in the plant thereby reducing the fire hazard.
- The reactor vessel is fabricated using ring forgings that eliminate the need for beltline longitudinal welds. This, in combination with improved material specifications, reduces concern for reactor vessel integrity.

Chronology of ACRS Review

Our review of the ABWR application commenced after it was docketed in February 1988. The NRC staff issued a Draft Safety Evaluation Report (DSER) on the first module of the SSAR in August 1989 (Ref. 2). We reviewed this draft and reported our findings in November (Ref. 3). At that time we questioned, in particular, the adequacy of the level of design detail available for review and recommended that the staff revisit the issue of what constitutes an "essentially complete" design.

Subsequent to November 1989, our review activities focused on several ABWR-related design concerns including Control Building flooding, physical separation, environmental protection of sensitive equipment, performance of essential chilled water systems, use of leak-before-break methodology, use of integral low-pressure turbine rotors, and the capability of the floor area beneath the reactor vessel to cope with severe accidents. These preliminary concerns were brought to the attention of the NRC staff in our July 1991 report (Ref. 4).

During 1991 the DSER was completed by the NRC staff in the form of six SECY papers (SECY-91-153, 235, 294, 309, 320, and 355). These papers generally covered most sections of the SSAR through the first eighteen amendments, but contained numerous open items. We

AIIM

Association for Information and Image Management

1100 Wayne Avenue, Suite 1100
Silver Spring, Maryland 20910
301/587-8202

Centimeter

Inches

MANUFACTURED TO AIIM STANDARDS
BY APPLIED IMAGE, INC.

10 of 10

The Honorable Ivan Selin

4

April 14, 1994

reported our findings in April 1992 (Ref. 5). In this report, we reconfirmed the preliminary concerns expressed in our July 1991 report and added several more including adequacy of the PRA, containment hydrodynamic loads, Reactor Water Cleanup System safety implications, plant design life and aging management, station grounding and surge protection, and corrosion control for structures.

In October 1992, the NRC staff issued a Draft Final Safety Evaluation Report (DFSER) (Ref. 6) covering the entire SSAR through Amendment 20. This draft superseded the six SECY papers. The final version of the staff safety evaluation report which we reviewed was the "Advance Copy of Safety Evaluation Report related to the certification of the Advanced Boiling-Water Reactor Design," dated December 1993 (Ref. 7). This copy covered the NRC staff review of SSAR information through about Amendment 32. Additional changes, including those which reflect Amendments 33 and 34, were reviewed by us as page changes to Reference 7.

Between February 1988 when the ABWR application was docketed and April 1992 when we issued our report on the DSER, our ABWR subcommittee held numerous meetings to review the SSAR and the NRC staff safety evaluations. During this same period, our subcommittee on Improved Light Water Reactors held several meetings to review the Electric Power Research Institute (EPRI) Utility Requirements Document (URD) and associated NRC staff safety evaluations for the Advanced Light-Water Reactor (ALWR) evolutionary plant. (The EPRI URD prescribes ALWR design requirements from the utility industry perspective.) Meetings were also held by our subcommittees on Auxiliary and Secondary Systems, Computers in Nuclear Power Plant Operations, Human Factors, and Severe Accidents. These subcommittees reviewed a number of specialized aspects of the proposed ABWR design including those related to fire, digital control and protection systems, human factors, and severe accidents.

Between April 1992 and today, our ABWR subcommittee held additional meetings to review design features proposed beyond Amendment 20 of the SSAR and to review the DFSER and Reference 7. This review covered significant design changes in the SSAR (through Amendment 34) and closure of all open items in the DFSER. It also included a review of written responses by GENE to numerous questions and concerns raised by the subcommittee.

During this time our subcommittee on Improved Light Water Reactors held several meetings to complete its review of the EPRI URD. In addition, ABWR-related meetings were held by our subcommittees on Auxiliary and Secondary Systems, Computers in Nuclear Power Plant Operations, Human Factors, Severe Accidents, Safeguards and Security, and our Ad Hoc Subcommittee on Design Acceptance Criteria (DAC). We did not review most of the CDM portion of the applica-

The Honorable Ivan Selin

5

April 14, 1994

tion because we were assured by the NRC staff that it did not contain design features and requirements beyond those found in the SSAR. We did, however, review and comment (Ref. 8) on the viability of the DAC process as a suitable method for establishing future design acceptance requirements in certain areas (i.e., human factors engineering, radiation protection, piping design, and instrumentation and control). We also reviewed the CDM related to these DAC areas.

During our review of the ABWR SSAR, we considered the design-specific requirements which relate to the various evolutionary and advanced light water reactor policy, technical, and licensing issues included in SECY-90-016 (Ref. 9) and its successor, SECY-93-087 (Ref. 10). These issues incorporate staff positions that deviate from or are not embodied in current regulations. Their resolutions will become "applicable regulations" through incorporation into the design certification rule for the ABWR. We have commented previously (Refs. 11 and 12) concerning these issues.

ACRS Conclusion Concerning ABWR Safety

Based on the results of our review of those portions of the GENE ABWR application which concern safety, we believe that acceptable bases and requirements have been established in the application to assure that the U.S. version of the ABWR standard design can be used to engineer and construct plants that with reasonable assurance can be operated without undue risk to the health and safety of the public.

Additional comments by ACRS Members Carlyle Michelson and Charles J. Wylie are presented below.

Sincerely,

J. Ernest Wilkins, Jr.
Chairman

Additional Comments by ACRS Members Carlyle Michelson and Charles J. Wylie

Although the Committee has arrived at a favorable conclusion concerning ABWR safety with which we agree, it is our view that this report should discuss the resolution of various issues that were considered by the Committee (Refs. 4 and 5) prior to reaching the favorable conclusion. Some of the resolutions were based on findings that were unanticipated and led to significant design changes. We believe that these findings should be made available

The Honorable Ivan Selin

6

April 14, 1994

to those who must make the final safety and design certification decisions.

As an example, it was found that the rupture of an 8-inch pipe in the non-safety-grade Reactor Water Cleanup (CUW) System which is housed inside of secondary containment creates serious environmental disruption throughout the three separate divisional areas of secondary containment which house redundant portions of the Emergency Core Cooling System (ECCS). Since this 8-inch pipe contains reactor coolant at operating temperature and pressure, the break results in an immediate loss of reactor coolant until isolated and it requires an ECCS response. Steam from the break permeates the entire secondary containment because the divisional barrier doors are forced open by a buildup of steam pressure. This occurs before the primary containment isolation valves for the CUW system have time to close. A similar situation exists for the Reactor Core Isolation Cooling (RCIC) System; however, the resulting environmental conditions for most locations are bounded by those produced by the CUW 8-inch pipe break.

Since these pipe break events cannot be confined, GENE now proposes that safety-related equipment inside of the ABWR secondary containment be environmentally qualified for steam at 15 psig. and about 248°F. It is our view that this is an acceptable, although undesirable, alternative to a design which provides separation barriers and pressure relieving pathways that are capable of isolating a sufficient amount of ECCS equipment from the harsh environment. In addition, GENE has added a third break isolation valve in the 8-inch CUW supply line and located it inside of primary containment. This valve can be closed after the blowdown is over to ensure the interruption of any prolonged loss of ECCS water to secondary containment. It is needed only if both primary containment isolation valves fail to fully close due to the severe blowdown loads or other challenges common to both valves. The added environmental qualification and the third valve are new features.

References:

1. U.S. Nuclear Regulatory Commission, Policy Statement, 10 CFR Part 50, "Nuclear Power Plant Standardization," 52 FR 34884, September 15, 1987
2. Letter dated August 17, 1989, from Charles L. Miller, NRC Office of Nuclear Reactor Regulation, to Patrick W. Marriott, General Electric Company, enclosing Draft Safety Evaluation Report Related to the Final Design Approval and Design Certification of the Advanced Boiling Water Reactor, August 1989
3. ACRS report dated November 24, 1989, from Forrest J. Remick, ACRS Chairman, to James M. Taylor, NRC Executive Director for Operations, Subject: Module I of the Draft Safety Evaluation Report for the Advanced Boiling Water Reactor Design

The Honorable Ivan Selin

7

April 14, 1994

4. ACRS report dated July 18, 1991, from David A Ward, ACRS Chairman, to James M. Taylor, NRC Executive Director for Operations, Subject: Concerns Related to the General Electric Advanced Boiling Water Reactor Design
5. ACRS report dated April 13, 1992, from David A. Ward, ACRS Chairman, to James M. Taylor, NRC Executive Director for Operations, Subject: Review of the Draft Safety Evaluation Reports on the GE Advanced Boiling Water Reactor Design
6. U. S. Nuclear Regulatory Commission, NUREG-1469, "Draft Final Safety Evaluation Report Related to the Design Certification of the General Electric Nuclear Energy Advanced Boiling Water Reactor," October 1992
7. U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, "Advance Copy of Safety Evaluation Report related to the certification of the Advanced Boiling-Water Reactor Design," December 1993
8. ACRS report dated January 14, 1994, from J. Ernest Wilkins, Jr., ACRS Chairman, to Ivan Selin, NRC Chairman, Subject: Final Report on the Use of the Design Acceptance Criteria Process in the Certification of the General Electric Nuclear Energy Advanced Boiling Water Reactor Design Approval
9. SECY-90-016, dated January 12, 1990, from James M. Taylor, NRC Executive Director for Operations, for the Commissioners, Subject: Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements
10. SECY-93-087, dated April 2, 1993, from James M. Taylor, NRC Executive Director for Operations, for the Commissioners, Subject: Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs
11. ACRS report dated April 26, 1990, from Carlyle Michelson, ACRS Chairman, to Kenneth M. Carr, NRC Chairman, Subject: Evolutionary Light Water Reactor Certification Issues and Their Relationship to Current Regulatory Requirements.
12. ACRS report dated April 26, 1993, from Paul Shewmon, ACRS Chairman, to Ivan Selin, NRC Chairman, Subject: SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs"

22 CONCLUSIONS

The staff performed its review of the U.S. ABWR standard safety analysis report, certified design material, and technical specifications in accordance with the standards for review of design certification applications set forth in 10 CFR §52.48 that are applicable and technically relevant to the U.S. ABWR standard design, including the exemptions and applicable regulations identified in Section 1.6 of this report. On the basis of its evaluation and independent analyses as discussed in this report, the staff concludes that, subject to satisfactory resolution of the confirmatory items identified in Section 1.8 of this report, GE Nuclear Energy's application for design certification meets the requirements of 10 CFR §52.47 that are applicable and technically relevant to the U.S. ABWR standard design. A copy of the report by the Advisory Committee on Reactor Safeguards required by 10 CFR §52.53 is provided in Chapter 21 of this report.

The staff also concludes that issuance of a final design approval, in accordance with Appendix O to 10 CFR Part 52, will not be inimical to the common defense and security or to the health and safety of the public. The financial qualifications of the applicable utility and the indemnity requirements of 10 CFR Part 140 will be addressed during the plant-specific licensing process for an application that references the U.S. ABWR standard design.

A final design approval, issued on the basis of this SER, does not constitute a commitment to issue a permit or license, or in any way affect the authority of the Commission, the Atomic Safety and Licensing Board, and other presiding officers, in any proceeding pursuant to Subpart G of 10 CFR Part 2.

**DATE
FILMED**

9/20/94

END