20. GENERIC ISSUES

20.1 Introduction

This chapter discusses the staff's evaluation of unresolved safety issues (USIs) and generic safety issues (GSIs), Three Mile Island (TMI) action plan items addressed in Title10 of the *Code of Federal Regulations* (10 CFR) 50.34(f), and incorporation of operating experience in the economic simplified boiling-water reactor (ESBWR) design submitted by GE-Hitachi Nuclear Energy (GEH). Since there are a large number of generic issues (GIs) relevant to the ESBWR design, this chapter predominantly directs the reader to other chapters and sections of the safety evaluation report (SER) that present the staff's evaluation of GIs. However, this section of the report does specifically address some GIs.

Tables 20.1-1, 20.1-2, and 20.1-3 list all GIs relevant to the ESBWR design and the SER chapters or sections where they are addressed. Table 20.1-1 lists GSIs and USIs, which include task action plan items, new GIs, TMI action plan items, and human factor issues as identified in NUREG-0933, "A Prioritization of Generic Safety Issues." Table 20.1-2 lists TMI action plan items addressed in 10 CFR 50.34(f). Table 20.1-3 lists generic letters (GLs) and bulletins (BLs) that deal with operating experience.

The staff's evaluations of GIs described in this chapter are grouped according to issue type. Section 20.2 contains task action plan items which include both USIs and GSIs. Section 20.3 addresses new GIs, which are categorized as GSIs. Section 20.4 addresses TMI action plan items, which includes those required by 10 CFR 50.34(f) and those GSIs identified in NUREG-0933.

20.1.1 Compliance with 10 CFR 52.47(a)(21)

10 CFR 52.47(a)(21) requires an application for design certification (DC) to include proposed technical resolutions of the USIs and medium- and high-priority GSIs as defined in NUREG-0933, "A Prioritization of Generic Safety Issues." These issues must be technically relevant to the design and are identified in the applicable NUREG-0933 supplement that was current 6 months before the docket date of the application.

In Design Control Document (DCD) Tier 2, Table 1.11-1 in Section 1.11 and Table 1A-1 in Appendix 1A the applicant addresses the USIs and GSIs relevant to the ESBWR design. The staff evaluation of the resolution of the USIs and GSIs are described in the SER sections listed in Table 20.1-1. Table 20.1-1 provides the issue designation, title, and the SER section that addresses the issue. The USIs and GSIs listed in Table 20.1-1 include task action plan items, new GIs, and TMI action plan items.

20.1.2 Compliance with 10 CFR 52.47(a)(8)

According to 10 CFR 52.47(a)(8) a DC applicant must demonstrate compliance with any technically relevant parts of the TMI action plan requirements found in 10 CFR 50.34(f). The applicant addressed these requirements in DCD, Tier 2, Table 1A-1 in Appendix 1A. Because of the overlap between the TMI action plan requirements and the GSIs identified in NUREG-0933, Table 20.1-2 lists all the relevant parts of the TMI action plan

items found in 10 CFR 50.34(f) in tabular form. Table 20.1-2 provides the issue designation, the 10 CFR 50.34(f) requirements, and the SER section in which they are addressed. Staff's evaluation of the resolution of the TMI Action Plan items are described in the designated SER sections listed in Tables 20.1-1 and 20.1-2.

20.1.3 Incorporation of Operating Experience

As part of its program to disseminate information on operating experience to the nuclear industry, the U.S Nuclear Regulatory Commission (NRC) issues generic communications when a significant safety-related event or condition at a facility may potentially apply to other facilities. The generic communications are issued in form of GLs, BLs, and information notices (INs). The NRC issues GLs and BLs when the event or condition requires the licensees to inform the NRC of what actions they have taken or will take to address the event or condition that is potentially significant to safety. The agency issues INs when it has determined that licensees should be informed of an event or condition but the communication does not contain any requests for action by the licensees. Potential safety issues highlighted in NRC generic communications may be incorporated into formal requirements or may eventually become a USI or GSI.

In the staff requirements memorandum (SRM), dated February 15, 1991, concerning SECY-90-377, "Requirements for Design Certification Under 10 CFR Part 52," dated November 8, 1990, the commission directed the staff to ensure that the DC process preserves operating experience insights in the certified design. In the NRC program to review and incorporate operating experience, the BLs and GLs that are issued to the nuclear industry convey the most safety-significant lessons distilled from many sources of information. In contrast, INs do not require action by the licensees.

The applicant addressed incorporation of operating experience in DCD, Tier 2, Tables 1C-1 and 1C-2 in Appendix 1C. The SER sections listed in Table 20.1-3 describe the staff's evaluation of the applicant's incorporation of operating experience in the ESBWR design .The table provides issue designation, title, and, the SER section that addresses the issue.

Table 20.1-1 USIs and GSIs in NUREG 0933 Relevant to the ESBWR Design

Issue Designation	Title	SER Chapter/ Section
	Task Action Plan Items	
A-1	Water Hammer	3.12, 10.3, 10.47
A-6	Mark I Short Term Program	6.2.1.3.3
A-7	Mark I Long Term Program	6.2.1.3.3
A-8	Mark II Containment Pool Dynamic Loads-Long Term Program	6.2.1.3.3
A-9	ATWS (Former USI)	15.6.4
A-10	BWR Feed Water Nozzle Cracking	5.3.1.2, 10.4.7
A-11	Reactor Vessel Materials Toughness (Former USI)	5.3.1.2
A-13	Snubber Operability Assurance	3.9.6, 20.2
A-17	Systems Interactions in Nuclear Power Plants (Former USI)	20.2
A-19	Digital Computer Protection System	7.1
A-23	Containment Leak Testing	6.2.3
A-24	Qualification of Class 1E Safety-Related Equipment	3.11
A-25	Non-Safety Loads on Class 1E Power Sources	8.3.1
A-28	Increase in Spent Fuel Pool Storage Capacity	9.1.3
A-29	Nuclear Power Plant Design for the Reduction of Vulnerability to Industrial Sabotage	13.6
A-30	Adequacy of Safety-Related DC Power Supplies	8.3.2
A-31	RHR Shutdown Requirements	20.2
A-34	Instruments for Monitoring Radiation and Process Variables During Accidents	7.5
A-35	Adequacy of Offsite Power Systems	8.2
A-36	Control Heavy Loads Near Spent Fuel	9.1.5
A-39	Determination of Safety Relief Valve Pool Dynamic Loads and Temperature Limits	6.2.1.3.3
A-40	Seismic Design Criteria Short Term Program	3.7.1, 3.7.2, 3.7.3
A-42	Pipe Cracks in Boiling Water Reactors	5.2.3
A-44	Station Blackout	8.4
A-45	Shutdown Decay Heat Removal Requirements	20.2
A-46	Seismic Qualification of Equipment in Operating Plants (Former USI)	3.10.3.6
A-47	Safety Implications of Control Systems (Former USI)	7.7
A-48	Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment	6.2.5
B-6	Loads, Load Combinations, Stress Limits	3.8.1, 3.8.2, 3.8.3, 3.8.4, 3.8.5
B-10	Behavior of BWR Mark III Containments	6.2.1.3.3
B-12	Containment Cooling Requirements (Non-LOCA)	6.2.2.3
B-17	Criteria for Safety-Related Operator Actions	18.15.2

Issue Designation	Title	SER Chapter/ Section
B-19	Thermal-Hydraulic Stability	4.4
B-26	Structural Integrity of Containment Penetrations	3.81, 3.8.2
B-36	Develop Design, Testing, and Maintenance Criteria for Atmosphere Cleanup System Air Filtration and Adsorption Units for Engineered Safety Feature Systems and For Normal Ventilation Systems	6.4
B-48	BWR Control Rod Drive Mechanical Failures	4.5.1
B-55	Improved Reliability of Target Rock Safety Relief Valves	20.2
B-60	Loose Parts Monitoring System	4.4
B-63	Isolation of Low Pressure Systems Connected to the Reactor Coolant Pressure Boundary	20.2
B-66	Control Room Infiltration Measurements	6.4
B-67	Effluent and Process Monitoring Instrumentation	11.5
C-1	Assurance of Continuous Long Term Capability of Hermetic Seals on Instrumentation and Electrical Equipment	20.2
C-2	Study of Containment Depressurization by Inadvertent Spray Operation to Determine Adequacy of Containment External Design Pressure	6.2.1
C-4	Statistical Methods for ECCS Analysis	6.3, 21.6
C-5	Decay Heat Update	6.3
C-6	LOCA Heat Sources	6.3
C-11	Assessment of Failure and Reliability of Pumps and Valves	20.2
C-17	Interim Acceptance Criteria for Solidification Agents for Radioactive Solid Wastes	11.4
D-3	Control Rod Drop Accident	15.4
	New Generic Issues	
6	Separation of Control Rod from Its Drive and BWR High Rod Worth Events	4.6
15	Radiation Effects on Reactor Vessel Supports	5.3.3
29	Bolting Degradation or Failure in Nuclear Power Plants	3.13
45	Inoperability of Instrumentation Due to Extreme Cold Weather	7.1
51	Proposed Requirements for Improving Reliability the Open Cycle Service Water System	9.2.1
57	Effects of Fire Protection System Actuation on Safety-Related Equipment	20.3
64	Identification of Protection System Instrument Sensing Lines	7.1
67.3.3	Steam Generator Staff Actions - Improved Accident Monitoring	7.5
75	Generic Implication of ATWS Events at the Salem Nuclear Plant	20.3

Issue Designation	Title	SER Chapter/ Section
78	Monitoring of Fatigue Transient Limits for Reactor Core Coolant System	3.12, 16
80	Pipe Break Effects on Control Rod Drive Hydraulic Lines in the Drywells of BWR Mark I and Mark II Containments	20.3
83	Control Room Habitability	6.4
86	Long Range Plan for Dealing with Stress Corrosion Cracking in BWR Piping	5.2.3
103	Design for Probable Maximum Precipitation	2.4.3
105	Interfacing Systems LOCA at LWRs	20.3
106	Piping and the Use of Highly Combustible Gases in Vital Areas	20.3
107	Main Transformer Failures	8.3.1
111	Stress Corrosion Cracking of Pressure Boundary Ferritic Steels in Selected Environment	5.2.3
113	Dynamic Qualification Testing of Large Bore Hydraulic Snubbers	3.9.3
119.1	Piping Rupture Requirements and Decoupling of Seismic and LOCA Loads	3.6.2
119.2	Piping Damping Values	3.12
119.3	Decoupling the OBE from the SSE	3.12
119.4	BWR Piping Materials	5.2.3, 6.1.1
120	On-Line Testability of Protection Systems	16
142	Leakage through Electrical Isolators in Instrumentation Circuits	7
143	Availability of Chilled Water Systems and Room Cooling	9.2.7
146	Support Flexibility of Equipment and Components	3.9.2
153	Loss of Essential Service Water in LWRs	9.2.1
156.6.1	Systematic Evaluation Program - Piping Break Effects on Systems and Components	3.6.2, 3.9.3, 3.8.1, 3.8.3
157	Containment Performance	19.2
166	Adequacy of Fatigue Life of Metal Components	3.12.6.7
173.A	Spent Fuel Storage Pool – Operating Facilities	9.1.3
186	Potential Risk and Consequences of Heavy Loads in Nuclear Power Plants	9.1.5
189	Susceptibility of Ice Condenser and Mark III Containments to Early Failure from Hydrogen Combustion During a Severe Accident	20.3
191	Assessment of Debris Accumulation on PWR Sump Performance	6.2.1
193	BWR ECCS Suction Concerns	20.3
	Three Mile Island Action Plan Items	
I.C.5	Procedures for Feedback of Operating Experience to Plant Staff	20.4

Issue Designation	Title	SER Chapter/ Section
I.C.9	Long -Term Program Plan for Upgrading of Procedures	20.4
I.D.1	Control Room Design Review	18.15.3
I.D.2	Plant Safety Parameter Display Console Description	18.15.3
I.D.3	Safety System Status Monitoring	7
I.F.1	Expand QA list	20.4
I.F.2	Develop More Detailed QA Criteria	20.4
II.B.1	Reactor Coolant System Vents	5.4.12
II.B.2	Plant Shielding To Provide Access to Vital Areas and Protect Safety Equipment for Post-Accident Operation	3.11, 12.4.3.5
II.B.3	Post- Accident Sampling	11.5
II.B.8	Rulemaking Proceedings on Degraded Core Accidents	19.2
II.D.1	Testing Requirements	5.2.2
II.D.3	Relief and Safety Valve Position Indication	5.2.2
II.E.4.2	Isolation Dependability	6.2.4
II.E.4.4	Purging	6.2.4
II.F.1	Additional Accident Monitoring Instrumentation	7.5, 12.4.3.4
II.F.2	Identification of and Recovery from Conditions Leading to Inadequate Core Cooling	7.1.1.3.4
II.F.3	Instruments for Monitoring Accident Conditions	7.5
II.J.3.1	Organization and Staffing To Oversee Design and Constructions	20.4
II.K.1(22)	Describe Automatic and Manual Actions for Proper Functioning of Auxiliary Heat Removal System When Feedwater System Not Operable	20.4
II.K.2(10)	Hard-Wired Safety Grade Anticipatory Reactor Trips	7.4
II.K.3(13)	Separation of HPCI and RCIC System Initiation Levels	20.4
II.K.3(16)	Reduction of Challenges and Failures of Relief Valves- Feasibility Study and System Modification	5.2.2
II.K.3(18)	Modification of ADS Logic- Feasibility Study and Modification for Increased Diversity for Some Events Sequences	7.3
II.K.3(21)	Restart of Core Spray and LPCI Systems on Low Level-Design and Modification	6.3
II.K.3(23)	Central Water Level Recording	7
II.K.3(24)	Confirm Adequacy of Space Cooling for HPCI and RCIC Systems	6.3
II.K.3(28)	Study and Verify Qualification of Accumulators on ADS Valves	20.4
II.K.3(45)	Evaluate Depressurization with Other Than Full ADS	5.3.2
III. A.1.2(1)	Technical Support Center	13.3
III.A.1.2(2)	Onsite Operational Support Center	13.3
III.D.1.1	Primary Coolant Sources Outside the Containment Structure	20.4

Issue Designation	Title	SER Chapter/ Section
III.D.3.3	In-Plant Radiation Monitoring	12.6
III.D.3.4	Control Room Habitability	6.4
	Human Factors Issues	
HF1.1	Shift Staffing	18.6, 18.15.1
HF4.4	Guidelines for Upgrading Other Procedures	18.9, 18.15.2
HF4.5	Application of Automation and Artificial Intelligence	18.3
HF5.1	Local Control Stations	18.15.2
HF5.2	Review Criteria for Human Factors Aspects of Advanced Controls and Instrumentation	18.8, 18.15.2

Table 20.1-2 10 CFR 52.47(a)(8) TMI Action Plan Items

TMI Requirement	10 CFR 50.34(f)	SER Chapter/Section
I.C.5	(3)(i)	20.4
I.C.9	(2)(ii)	20.4
I.D.1	(2)(iii)	18.15.3
I.D.2	(2)(iv)	18.15.3
I.D.3	(2)(v)	7
I.F.1	(3)(ii)	20.4
I.F.2	(3)(iii)	20.4
II.B.1	(2)(vi)	5.4.12
II.B.2	(2)(vii)	3.11, 12.4.3.5
II.B.3	(2)(viii)	11.5
II.B.8	(1)(i) & (xii), (2)(ix), (3)(iv) & (v)	19.2
II.D.1	(2)(x)	5.2.2
II.D.3	(2)(xi)	5.2.2
II.E.4.2	(2)(xiv)	6.2.4
II.E.4.4	(2)(xv)	6.2.4
II.F.1	(2)(xvii)	7.5, 12.4.3.4
II.F.2	(2)(xviii)	7.1.1.3.4
II.F.3	(2)(xix)	7.5
II.J.3.1	(3)(vii)	20.4
II.K.1(22)	(2)(xxi)	20.4
II.K.2(10)	(2)(xxiii)	7.4
II.K.3(13)	(1)(v)	20.4
II.K.3(16)	(1)(vi)	5.2.2
II.K.3(18)	(1)(vii)	7.3
II.K.3(23)	(2)(xxiv)	7
II.K.3(21)	(1)(viii)	6.3
II.K.3(24)	(1)(ix)	6.3
II.K.3(28)	(1)(x)	20.4
II.K.3(45)	(1)(xii)	5.2.3
III. A.1.2(1)	(2)(xxv)	13.3
III.D.1.1	(2)(xxvi)	20.4
III.D.3.3	(2)(xxvii)	12.6
III.D.3.4	(2)(xxviii)	6.4

Table 20.1-3 Generic Letters and Bulletins

Issue Designation	Title	SER Chapter/Section
	Generic Letters	
GL 80-09	Low Level Radioactive Waste Disposal	11.4
GL 80-34	Clarification of NRC Requirements for Emergency Response Facilities at Each Site	13.3
GL 80-113	Controls of Heavy Loads	9.1.5
GL 81-04	Emergency Procedures and Training for Station Blackout Events	18.15.4
GL 81-03	Implementation of NUREG-0313, Revision 1, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping (Generic Task A-42)"	5.2.3
GL 81-07	Control of Heavy Loads	9.1.5
GL 81-10	Post TMI Requirements for the Emergency Operations Facility	13.3
GL 81-11	"BWR Feedwater Nozzle and Control Rod NozzleDrive Return Line Nozzle Cracking" (NUREG - 0619)	10.4.7
GL 81-20	Safety Concerns Associated with Pipe Breaks in the BWR Scram System	20.5.1
GL 81-37	ODYN Code Re-analysis Requirements	20.5.1
GL 81-38	Storage of Low-Level Radioactive Wastes at Power Reactor Sites	11.4
GL 81-39	NRC Volume Reduction Policy	11.4
GL 82-09	Environmental Qualification of Safety-Related Electrical Equipment	3.11
GL 82-21	Technical Specifications for Fire Protection Audits	16
GL 82-23	Inconsistency Between Requirement of 10 CFR 73.40(D) and Standard Technical Specifications for Performing Audits of Safeguards Contingency Plans	16
GL 82-27	Transmittal of NUREG 0763, "Guidelines for Confirmatory In-Plant Tests of Safety Relief Valve Discharges for BWR Plants," and NUREG 0783, "Suppression Pool Temperature Limits for BWR Containments"	6.2.1.1.6
GL 82-33	Supplement 1 to NUREG- 0737 – Requirements for Emergency Response Capability	7.1.1.3.5,13.3 and 18.15.3
GL 83-05	Safety Evaluation of "Emergency Procedure Guidelines Revision 2," NEDO-24934, June 1982	18.9, 13.5
GL 83-13	Clarification of SRs Surveillance Requirements for HEPA Filters and Charcoal Adsorber Units in STS and ESP Standard Technical Specifications of ESF Cleanup Systems	6.2.3
GL 83-28	Required Actions Based on Generic Implications of Salem ATWS Events	20.5.1

Issue Designation	Title	SER Chapter/Section
GL 83-33	NRC Positions on Certain Requirements of Appendix R to 10 CFR Part 50	20.5.1
GL 84-15	Proposed Staff Actions To Improve and Maintain Diesel Generator Reliability	8.3.1
GL 84-23	Reactor Vessel Water Level Instrumentation in BWRs	20.5.1
GL 85-01	Fire Protection Policy Steering Committee Report	20.5.1
GL 86-10	Implementation of Fire Protection Requirements	20.5.1
Supp1 to GL 86-10	Fire Endurance Test Acceptance Criteria for Fire Barrier Systems Used To Separate Redundant Safe Shutdown Trains Within the Same Fire Area, March 25, 1994	20.5.1
GL 87-06	Periodic Verification of Leak Tight Integrity of Pressure Isolation Valves	20.5.1
GL -87-09	Sections 3.0 and 4.0 of Standard Technical Specifications (STS) on the Applicability of Limiting Conditions for Operation and Surveillance Requirements	16
GL 88-01	NRC Position on Intergranular Stress Corrosion Cracking (IGSCC) in BWR Austenitic Stainless Steel Piping	5.2.3
GL 88-14	Instrument Air Supply System Problems Affecting Safety-Related Equipment	9.3.6
GL 88-15	Electric Power Systems- Inadequate Control Over Design Processes	8.3.1
GL 88-16	Removal of Cycle-Specific Parameter Limits from Technical specifications	16
GL 88-18	Plant Record Storage on Optical Disks	20.5.1
GL 88-20	Individual Plant Examination for Severe Accident Vulnerabilities	19.1
GL 89-01	Implementation of Programmatic and Procedural Controls for Radiological Effluent Technical Specification	11.4, 11.5, 16.5
GL 89-02	Actions To Improve the Detection of Counterfeit and Fraudulently Marketed Products	20.5.1
GL 89-04	Guidance on Developing Acceptable Inservice Testing Programs	20.5.1
GL 89-06	Task Action Plan Item I.D.2 -Safety Parameter Display System – 10 CFR 50.549(f)	18.15.4
GL 89-07	Powers Reactor Safeguards Contingency Planning for Surface Vehicle Bombs	13.6
GL 89-08	Erosion/Corrosion - Induced Pipe Wall Thinning	6.6, 10.3, 10.47
GL 89-10	Safety-Related Motor-Operated Valve Testing and Surveillance	3.9.6
GL 89-13	Service Water System Problems Affecting Safety-Related Equipment	9.2.1

Issue Designation	Title	SER Chapter/Section
Supp to GL 89-13	Service Water Systems Problems Affecting Safety-Related Equipment, April 4, 1990	9.2.1
GL 89-14	Line - Item Improvements in Technical Specifications- Removal of the 3.25 Limit on Extending Surveillance Intervals	16
GL 89-15	Emergency Response Data System	9.5
GL 89-16	Installation of a Hardened Wetwell Vent	19.2
GL 89-18	Resolution of Unresolved Safety Issue A-17, “System Interactions in Nuclear Power Plants”	20.5.1
GL 89-19	Request for Action Related to Resolution of Unresolved Safety-Related Issue A-47, “Safety Implication of Control Systems in LWR Nuclear Power Plants” Pursuant to 10 CFR 50.54(f)	7.7
GL 89-22	Potential for Increased Roof Loads and Plant Area Flood Runoff Depth at Licensed Nuclear Power Plants Due to Recent Change in Probable Maximum Precipitation Criteria Developed by the National Weather Service	3.8.4
GL 90-09	Alternative Requirements for Snubber Visual Inspection Intervals and Corrective Actions	3.9.6
GL 91-05	Licensee Commercial- Grade Procurement and Dedication Programs	20.5.1
GL 91-03	Reporting of Safeguards Events	13.6
GL 91-04	Changes in Technical Specification Surveillance Intervals To Accommodate a 24 Month Fuel Cycle	16
GL 91-06	Resolution of Generic Issue A-30, “Adequacy of Safety-Related DC Power Supplies,” Pursuant to 10 CFR 50.54(f)	20.5.1
GL 91-10	Explosives Searches at Protected Area Portals	13.6
GL 91-11	Resolution of Generic Issues 48, “LCOs for Class 1E Vital Instrumentation Bus,” and 49, “Interlocks and LCOs for Class 1E Tie Breakers” Pursuant to 10 CFR 50.54(f)	8.3.1
GL 91-14	Emergency Telecommunications	9.5
GL 91-16	Licensed Operators’ and Other Nuclear Facility Personnel Fitness for Duty	20.5.1
GL 91-17	Generic Safety Issue 29, “Bolting Degradation or Failure in Nuclear Power Plants”	3.13
GL 92-01	Reactor Vessel Structural Integrity	5.3.1.2
GL 92-04	Resolution of the Issues Related to Reactor Vessel Water Level Instrumentation in BWRs Pursuant to 10 CFR 30.54(f)	20.5.1
GL 92-08	Thermo-Lag 330-1 Fire Barriers (BL 92-001)	20.5.1
GL 93-05	Line-Item in Technical Specifications Improvements To Reduce Surveillance Requirements for Testing During Power Operations	16

Issue Designation	Title	SER Chapter/Section
GL 93-06	Research Results on Generic Safety Issue 106, "Piping and the Use of Highly Combustible Gases in Vital Areas," October 25, 1993	20.5.1
GL 93-08	Relocation of Technical Specifications Tables of Instrument Response Time Limits	16
GL 94-01	Removal of Accelerated Testing and Special Reporting Requirements for Emergency Diesel Generators	8.3.1
GL 94-02	Long-Term Solutions and Upgrade of Interim Operating Recommendation for Thermal-Hydraulic Instabilities in Boiling Water Reactors	4.4
GL 94-03	Intergranular Stress Corrosion Cracking of Core Shrouds in BWRs	4.5.2
GL 95-07	Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves	3.9.6
GL 96-01	Testing of Safety-Related Logic Circuits	7.1
GL 96-03	Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits	16
GL 96-04	Boraflex Degradation in Spent Fuel Pool Storage Racks	9.1.2
GL 96-05	Periodic Verification of Design Basis Capability of Safety-Related Motor -Operated Valves	3.9.6
GL 96-06	Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions	6.2.1
GL 97-04	Assurance of Sufficient Net Positive Suction Head for Emergency Core Cooling System and Containment Heat Removal Pumps	6.2.2, 6.3
GL 98-04	Potential for Degradation of the Emergency Core Cooling System and the Containment Spray System after a Loss-of-Coolant Accident Because of Construction and Protective Coating Deficiencies and Foreign Material in Containment	6.2.2, 6.3
GL 99-02	Laboratory Testing of Nuclear- Grade Activated Charcoal	6.2.3
GL 2003-01	Control Room Habitability	6.4
GL 2006-03	Potentially Nonconforming Hemyc and MT Fire Barrier Configurations, April 10, 2006	20.5.1
	Bulletins	
BL 79-02	Pipe Support Base Plate Designs Using Concrete Expansion Anchor Bolts	3.12
BL 80-01	Operability of ADS Valve Pneumatic Supply	20.5.2
BL 80-03	Loss of Charcoal from Standard Type II, 2 Inch, Tray Adsorber Cells	6.2.3
BL 80-06	Engineered Safety Features (ESF) Reset Controls	7.3
BL 80-08	Examination of Containment Liner Penetration Welds	20.5.2

Issue Designation	Title	SER Chapter/Section
BL 80-10	Contamination of Nonradioactive System, and Resulting Potential for Unmonitored, Uncontrolled Release of Radioactivity to the Environment	11.2, 11.3, 11.4, 11.5
BL 80-15	Possible Loss of Emergency Notification System (ENS) with Loss of Offsite Power	9.5
BL 80-24	Prevention of Damage Due to Water Leakage Inside Containment	9.2.7
BL 80-25	Operating Problems with Target Rock Safety-Relief Valves at Boiling Water Reactors	20.5.2
BL 81-01	Surveillance of Mechanical Snubbers	3.9.6
BL 81-03	Flow Blockage of Cooling Water to Safety System Components by Corbicula sp. (Asiatic Clam) and Mytilus sp. (Mussel)	9.2.1
BL 82-04	Deficiencies in Primary Containment Electrical Penetration Assemblies	8.3.1
BL 85-03	Motor-Operated Valve Common Mode Failures During Plant Transients Due to Improper Switch Settings	20.5.2
BL 86-01	Minimum Flow Logic Problems That Could Disable RHR Pumps	20.5.2
BL 87-01	Thinning of Pipe Walls In Nuclear Power Plants	6.6, 10.3.6
BL 88-07	Power Oscillations in Boiling Water Reactors	4.4
BL 88-08	Thermal Stresses in Piping Connected to Reactor Coolant Systems	3.12
BL 90-02	Loss of Thermal Margin Caused by Channel Box Bow	4.2
BL 93-02	Debris Plugging of Emergency Core Cooling Systems Suction Strainers	6.2.1
BL 93-03	Resolution of Issues Related to Reactor Vessel Water Level Instrumentation in Boiling Water Reactors	20.5.2
BL 94-01	Potential Fuel Pool Draindown Caused by Inadequate Maintenance Practices at Dresden Unit 1	9.1.3
BL 95-02	Unexpected Clogging of a Residual Heat Removal Pump Strainer While Operating in Suppression Pool Cooling Mode	6.2.1
BL 96-02	Movement of Heavy Loads over Spent Fuel, over Fuel in the Reactor Core, or over Safety-Related Equipment	9.1.5
BL 96-03	Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling Water Reactors	6.2.1
BL 05-02	Emergency Preparedness and Response Actions for Security- Based Events	13.3

20.2 Task Action Plan Items

This section addresses the staff's evaluation of USIs and GSIs that are categorized as "task action plan items" in NUREG-0933.

A-17: Systems Interactions in Nuclear Power Plants

As discussed in NUREG-0933, Issue A-17 addresses concerns about adverse system interactions (ASIs) in nuclear power plants. Depending on how they propagate, ASIs can be classified as functionally coupled, spatially coupled, and induced-human-intervention coupled. As discussed in NUREG-1229, "Regulatory Analysis for Resolution of USI A-17," issued August 1989, and GL 89-18, "Resolution of Unresolved Safety Issue A-17, Systems Interactions in Nuclear Power Plants," dated September 6, 1989, Issue A-17 concerns ASIs caused by water intrusion, internal flooding, seismic events, and pipe ruptures.

A nuclear power plant is comprised of numerous structures, systems, and components (SSCs) that are designed, analyzed, and constructed using many different engineering disciplines. The degree of functional and physical integration of these SSCs into any single power plant may vary considerably. Concerns have been raised about the adequacy of this functional and physical integration and the coordination process. The Issue A-17 program was initiated to integrate the areas of systems interactions and to consider viable alternatives for regulatory requirements to ensure that ASIs have been or will be minimized in operating and new plants. Within the framework of the program, the staff requested, as stated in NUREG-0933, that plant designers consider the operating experience discussed in GL 89-18 and use the probabilistic risk assessment (PRA) required for future plants to identify vulnerability and reduce ASIs.

Issue A-17 concerns the need to investigate the potential that unrecognized subtle dependencies, or systems interactions, among SSCs in a plant could lead to safety-significant events. In NUREG-1174, "Evaluation of Systems Interactions in Nuclear Power Plants: Technical Findings Related to Unresolved Safety Issue A-17," issued May 1989, inter-system dependencies are categorized based on the way they propagate into functionally coupled, spatially coupled, and induced-human-intervention coupled systems interactions. The occurrence of an actual ASI or the existence of a potential ASI, as well as the potential overall safety impact, is a function of an individual plant's design and operational features. For the ESBWR with new or differently configured passive and active systems, a systematic search for ASIs is necessary.

The applicant used a systematic process to analyze specific features and actions that are designed to prevent postulated adverse interactions. In its response to Request for Additional Information (RAI) 19.1.0-2, the applicant submitted an assessment of significant adverse interactions.

The purpose of the applicant's assessment was to identify possible adverse interactions among safety-related systems (passive systems) and between safety-related and non-safety-related systems (active systems), and to evaluate the potential consequences of such interactions. The assessment evaluated the gravity-driven cooling system (GDCS), automatic depressurization system (ADS), isolation condenser system (ICS), standby liquid control system (SLCS), and passive containment cooling system (PCCS). Interaction of these systems with other systems such as fuel and auxiliary pools cooling

system (FAPCS), direct current power, suppression pool, main steam, containment, high pressure nitrogen supply system and radiation monitoring system, was studied by the applicant. The staff reviewed this study as part of their review of the regulatory treatment of non-safety systems (RTNSS) as described in Chapter 22.5.5 of this report. For the purpose of the staff's analysis, an adverse system interaction exists, if the action or condition of an active, interfacing system causes a loss of safety function of a passive system.

The applicant addresses Issue A-17 in DCD, Tier 2, Table 1.11-1. Based on the above information and its evaluation in 22.5.5 of this report, the staff concludes that the applicant has adequately assessed possible ASIs and their potential consequences and issue A-17 is resolved for the ESBWR design.

A-31: RHR Shutdown Requirements (former USI)

As discussed in NUREG-0933, Issue A-31 addresses the ability of a plant to transfer heat from the reactor to the environment after shutdown, which is an important safety function. This issue was resolved in 1978 with the issuance of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants" (known as the SRP) Section 5.4.7, "Residual Heat Removal (RHR) System."

The safe shutdown of a nuclear power plant following an accident not related to a loss-of-coolant accident (LOCA) has typically been interpreted by the staff as achieving "hot-shutdown" condition. The NRC has placed an emphasis on the hot-standby condition of a power plant in the event of an accident or other abnormal occurrence, as well as on long-term cooling, which is typically achieved by the RHR system. The RHR system starts to operate when the reactor coolant pressure and temperature are substantially lower than the values for the hot-standby condition. Although it may generally be considered safe to maintain a reactor in hot-standby condition for a long time, experience shows that certain events have occurred that required eventual cooldown or long-term cooling until the reactor coolant system (RCS) was cold enough for personnel to inspect and repair the problem.

In GDC 34, the NRC requires an RHR system to be provided with suitable redundancy in components and features to ensure that, with or without onsite or offsite power, it can accomplish its safety functions so as not to exceed the specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary (RCPB). The Electric Power Research Institute (EPRI) Utility Requirements Document (URD) proposes that the safe shutdown condition be defined as less than 215.6 degrees Celsius (°C) (420 degrees Fahrenheit (°F)) for the passive advanced light- water reactor (ALWR) designs. In its evaluation of the URD, the staff concluded that cold shutdown is not the only safe stable shutdown condition able to maintain the fuel and RCPB within acceptable limits. In SECY-94-084, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs," Section C, "Safe Shutdown Requirements," dated March 28, 1984, the staff recommended, and the Commission approved, that the EPRI-proposed 215.6 °C (420 °F) criterion or below, rather than the cold-shutdown condition described in SRP 5.4.7, be accepted as a safe stable condition, which the passive ICS system must be capable of achieving and maintaining following non-LOCA events. The staff's acceptance is predicated on an acceptable passive safety system performance and an acceptable resolution of the issue of RTNSS for reactor water cleanup (RWCU) system.

SECY-94-084 also states that the passive safety system capabilities can be demonstrated by appropriate evaluations during detailed design analyses, including the following two analyses:

(1) a safety analysis to demonstrate that the passive systems can bring the plant to a safe stable condition and maintain this condition such that no transients will violate the specified acceptable fuel design limits and pressure boundary design limit, and that no high-energy piping failure initiated by this condition will violate the 10 CFR 50.46 criteria, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors"

(2) a probabilistic reliability analysis, including events initiated from the safe-shutdown conditions, to ensure conformance with the safety goal guidelines and to determine the reliability and availability missions of risk-significant systems and components as a part of the effort for RTNSS

In DCD, Tier 2, Table 1.11, the applicant addresses this issue. The applicant states that the ESBWR is a passive plant and does not have the traditional RHR system. The applicant stated that the isolation condensers (ICs) can achieve and maintain a safe stable condition for at least 72 hours without operator action following non-LOCA events. DCD, Tier 2, Section 5.4.6 discusses the ICS. DCD, Tier 2, Sections 5.4.7 and 5.4.8 discusses the non-safety-related normal RWCU system. For normal shutdown and cooldown, residual and decay heat is removed via the main condenser and the shutdown cooling (SDC) mode of the RWCU system/SDC. The RWCU system consists of two redundant trains. In the event of loss-of-preferred power, the RWCU/SDC system, in conjunction with the ICs, is capable of bringing the reactor pressure vessel (RPV) to the cold shutdown condition in a day and a half, assuming limiting single active failure, and with the ICs removing the initial heat load.

The staff agrees with the applicant that for the ESBWR design, cold-shutdown conditions can be achieved using reliable, but non-safety-related systems, which have redundancy similar to that of the current generation safety-related systems and are supplied with alternating current power from either onsite or offsite sources. Sections 5.4.7 and 5.4.8 of this report provides the staff's evaluation of the RWCU system. Section 5.4.6 of this report provides the staff's evaluation of the ICS. The staff concludes that the ESBWR design complies with GDC 34, "Residual Heat Removal," by using a more reliable and simplified system for both hot-standby and long-term cooling modes of the RWCU system. The staff also concludes that it is not necessary that these passive systems achieve cold shutdown as discussed in SECY-94-084.

Section 22.5 of this report discusses the RTNSS issue in terms of the availability of the RWCU/SDC and ICS system during shutdown and refueling conditions. Based on the above discussion and the staff's evaluation in Sections 5.4.6, 5.4.7, 5.4.8, and 22.5 of this report, the staff considers Issue A-31 resolved for the ESBWR design.

<u>A-45: Shutdown Decay Heat Removal Requirements</u>

In March 1981, NUREG-0705, "Identification of New Unresolved Safety Issues Relating to Nuclear Power Plants—Special Report to Congress," designated this issue as a USI. The NRC initiated a program to evaluate the adequacy of the decay heat removal (DHR)

function in operating light-water reactors (LWRs) and to assess the value and impact (i.e., the benefit and cost) of alternative measures to improve the overall reliability of the DHR function.

According to NUREG-0933, the program employed PRAs and deterministic evaluations of those DHR systems and support systems required to achieve hot shutdown and cold shutdown conditions in both pressurized water reactors (PWRs) and boiling-water reactors (BWRs). Systems analysis techniques were used to assess the vulnerability of DHR systems to various internal and external events. The analyses were limited to transients, small-break LOCAs, and special emergency challenges such as fires, floods, earthquakes, and sabotage. Cost-benefit analysis techniques were used to assess the net safety benefit and cost of alternative measures to improve the overall reliability of the DHR function.

Establishing a safe-shutdown condition requires maintenance of the reactor in a subcritical condition and adequate cooling to remove residual heat. One of the functional requirements for the ESBWR is that the plant can be brought to a stable condition using the safety-grade systems for all events. Because of the functional limitations of the safety-related passive plant designs, the Commission, in a SRM issued June 30, 1994, approved the position proposed in SECY-94-084, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs." This position accepts 420 ^{0}F (215.6 ^{0}C) or below, rather than the cold shutdown (less than 200 ^{0}F) specified in SRP Branch Technical Position RSB BTP 5-1, as the safe stable condition that the passive decay heat removal system must be capable of achieving and maintaining following non-LOCA events. The SLCS establishes safe shutdown by providing the necessary reactivity control to maintain the core in a subcritical condition while ICS provides residual heat removal capability to maintain adequate core cooling. DCD, Tier 2, Sections 5.4, 6.3, 7.4, discusses the systems required for safe shutdown:

- ICS
- SLCS
- safety/relief valves (SRVs)
- depressurization valves (DPVs)
- GDCS
- PCCS

The passive ICS is automatically initiated upon closure of the main steam isolation valves (MSIVs) to remove decay heat following scram and isolation, and ICS condensate flow provides initial reactor coolant inventory makeup to the RPV. If the water reaches Level 1 in the reactor, the ADS, with the operation of the SRVs and DPVs, is initiated to depressurize the RPV.

To resolve USI A-45, one of the alternatives proposed by the staff in NUREG-0933 was to have each licensee perform a risk assessment for its plant. The regulation in 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," requires the DC applicant to perform a risk assessment. The Staff's evaluation of the ESBWR PRA is included in Section 19 of this report. Based on its evaluation in Section 19 of this report, the staff considers issue A-45 resolved.

B-55: Improved Reliability of Target Rock Safety Relief Valves

As discussed in NUREG-0933, Issue B-55 addresses the failure of a pressure relief system valve to open on demand which results in a decrease in the total available pressure-relieving capacity of the system. Similarly, spurious openings of pressure relief system valves, or failures of valves to properly reseat after opening, can result in inadvertent RCS blowdown with unnecessary thermal transients on the reactor vessel and the vessel internals, unnecessary hydrodynamic loading of the containment systems' pressure suppression chamber (torus) and its internal components, and potential increases in the release of radioactivity to the environs. In addition, if the failed valve also serves as part of the ADS, the ability of the ADS to perform its emergency core cooling function could be degraded.

In resolving the issue, the staff found that licensees had significantly improved the performance of Target Rock SRVs and were continuing to evaluate and improve their performance. Licensee compliance with existing regulations, such as Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50 and 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," were sufficient for the staff to pursue additional improvements on a plant-specific basis, if needed. Thus, the issue was resolved with no new or revised requirements.

In DCD, Tier 2, Table 1.11-1, the applicant indicated that Issue B-55 was resolved with no new requirements. ESBWR DCD, Tier 2, Section 3.9.3.5, "Valve Operability Assurance," specifies that the qualification programs for valve designs developed for the ESBWR that were not previously qualified will meet the requirements of American Society of Mechanical Engineers (ASME) Standard QME-1-2007, "Qualification of Active Mechanical Equipment Used in Nuclear Power Plants." For valves that were previously qualified, the ESBWR DCD specifies key features of lessons learned from nuclear power plant operations and research programs included in QME-1-2007 as part of its design specifications. For example, qualification specifications (e.g., design specifications) consistent with Appendix QV-I, "Qualification Specification for Active Valves," and Appendix QV-A, "Functional Specification for Active Valves for Nuclear Power Plants," to QME-1-2007 will be prepared for previously qualified valves to ensure operating conditions and safety functions for which the valves are to be qualified are communicated to the manufacturer or qualification facility. Suppliers will submit, for GEH review and approval, application reports as described in QME-1-2007 that describe the basis for the application of specific predictive methods and/or qualification test data to a valve application. In September 2009, the NRC issued Revision 3 to Regulatory Guide (RG) 1.100, "Seismic Qualification of Electric and Active Mechanical Equipment and Functional Qualification of Active Mechanical Equipment for Nuclear Power Plants," which accepts the use of the QME-1-2007 standard, with certain staff positions, for the functional design and qualification of safety-related pumps, valves, and dynamic restraints.

ESBWR DCD, Tier 2, Section 3.9.3.5 provides additional qualification provisions for specific valve types, such as safety relief valves. As discussed in Section 3.9.6 of this SER, the NRC staff considers the provisions in the ESBWR DCD for functional design and qualification of valves to be acceptable for the ESBWR DC in that the provisions incorporate the lessons learned from valve operating experience and research programs through application of the ASME QME-1-2007 standard for new valve qualification and

key features of QME-1-2007 for previously qualified valves, where applied consistent with NRC acceptance of the standard in Revision 3 to RG 1.100. As described in DCD Section 14.2.8.1.1, valve operability is also verified during the pre-operational test program. The SRVs are tested in accordance with quality control procedures to detect defects and to provide operability before installation. Based on its evaluation of valve qualification in Section 3.9.6 and pre-operational testing in Section 14.2.3.1 of this report, the staff finds issue B-55 is resolved for the ESBWR design.

B-63: Isolation of Low Pressure Systems Connected to the Reactor Coolant Pressure Boundary

As discussed in NUREG- 0933, Issue B-63 addresses the adequacy of the isolation of low-pressure systems that are connected to the RCPB. Design pressures in several systems connected to the RCPB in operating plants are considerably below the RCS operating pressure. The NRC has established acceptance criteria in SRP 3.9.6 to address the isolation of low pressures systems connected to the RCPB Isolation design function, quality, and testing of pressure isolation valves (PIVs) are evaluated and documented in Section 3.9.6 of this SER. The staff finds this acceptable because PIVs to be used in the ESBWR satisfy SRP 3.9.6 for functional design, qualification, and inservice testing (IST).

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 during equipment maintenance, such errors could lead to the loss of a seal and of equipment functionality. The focus 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.

DCD, Tier 2, Appendix 3H, "Equipment Qualification Design Environmental Conditions," defines the environmental conditions with respect to limiting design conditions for all safety-related mechanical and electrical equipment. Environmental conditions are tabulated by zones contained in the referenced building arrangements 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 DCD, Tier 2, Appendix 3H. The environmental qualification document will include a list of all safety-related electrical and mechanical equipment required for safe shutdown that is located in a harsh environment as stated in DCD, Tier 2, Section 3.11.1.

Safety-related electrical equipment that is located in a harsh environment is qualified by test or other methods, as described in Institute of Electrical and Electronic Engineers (IEEE) 323, “Qualifying Class 1E Equipment for Nuclear Power Generating Stations,” and permitted by 10 CFR 50.49(f). The NRC-approved report, NEDE-24326-1-P describes the qualification methodology in detail. This report also addresses compliance with the applicable portions of Appendix A to 10 CFR Part 50, and the quality assurance (QA) criteria of Appendix B to 10 CFR Part 50. Additionally, the report describes

conformance to NUREG-0588, “Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment,” issued November 1979, and the RGs and IEEE standards referenced in SRP Section 3.11. Details on the staff's evaluation of environmental qualification of safety-related electrical equipment are provided in Section 3.11 of this report. Based on the above discussion and the staff’s evaluation in Section 3.11 of this report, the staff concludes that GEH has adequately addressed this issue for the ESBWR design.

C-11: Assessment of Failure and Reliability of Pumps and Valves

In DCD, Tier 2, Table 1.11-1, provides the resolution of task action plan Item C-11, “Assessment of Failure and Reliability of Pumps and Valves,” for the ESBWR. In particular, GEH indicated that this item was generically resolved with no new requirements. The ESBWR DCD Tier 2 addresses the design and performance of pumps and valves in several subsections. The NRC staff discusses the functional design, qualification, and IST of pumps and valves in Section 3.9.6 of this report. ESBWR DCD, Tier 2, Section 3.9.6.2, “Inservice Testing of Pumps,” notes that no pumps are included in the IST Program because the ESBWR design does not require the use of pumps to mitigate the consequences of any design basis accidents, or to achieve or maintain the safe shutdown condition. Because of the ESBWR DCD provisions for valve design and qualification in accordance with ASME Standard QME-1-2007, and because COL Information Item 3.9.9-3-A, “Inservice Testing Programs,” requires COL applicants to provide a full description of the IST Program, the staff concludes that GEH has adequately addressed this issue in the ESBWR design.

20.3 New Generic Issues

This section addresses the staff’s evaluation of USIs and GSIs that are categorized as “new generic issues” in NUREG-0933.

Issued 57: Effects of Fire Protection System Actuation on Safety-Related Equipment

As discussed in NUREG-0933, Issue 57 provides guidance on avoiding damage to required safety-related equipment due to fire suppression system discharge. Appendix 9A, Section 9A.4, of the ESBWR DCD describes the design features to ensure that suppression system discharge will not prevent safe shutdown. The staff’s evaluation of DCD, Tier 2, Appendix 9A and Section 9A.4 can be found is Section 9.5.1 of this report. Based on its review in Section 9.5.1, the staff finds this GSI is resolved for the ESBWR design.

Issue 75: Generic Implications of ATWS Event at the Salem Nuclear Plant

As discussed in NUREG-0933, Issue 75 addresses the generic implications of two events at Salem Unit 1 where there were failures to scram automatically because of the failure of both reactor trip breakers to open upon receipt of an actuation signal. This issue was expanded to include a number of concerns raised by the staff, that were closely related to the design and testing of the reactor protection system (RPS). The requirements for this issue were stated in GL 83-28, “Required Actions Based on Generic Implications of Salem ATWS Event,” dated July 8, 1983.

In DCD, Tier 2, Table 1.11, the applicant addresses this issue. The RPS designs for BWRs are substantially different from the reactor trip system design used in Salem Unit 1. Sections 3.1.2.5 (and the preceding Sections 3.1.2.2 to 3.1.2.4) and Table 3.1 of NUREG-1000, Volume 1, "Generic Implications of ATWS Events at the Salem Nuclear Power Plant," issued April 1983, describe the basic differences between BWR designs used at the time of the Salem events and the reactor trip system designs then used by PWRs.

The applicant maintains that the ESBWR further improves upon the BWR RPS designs used at the time of the Salem anticipated transient without scram (ATWS) events. The RPS is designed to provide a reliable single failure-proof capability to automatically or manually initiate a reactor scram while maintaining protection against unnecessary scrams resulting from a single-failure even when bypassed and/or when one of the four automatic RPS trip logic systems is out-of-service. This is accomplished through the combination of fail-safe equipment design, the redundant two-out-of-four sensor channel trip decision logic, and the redundant two-out-of-four trip systems output scram logic arrangement utilized in the RPS design.

Staff's evaluation of the RPS system is included in Section 7 of this report. Based on its review of DCD Section 7.2.2, the staff finds that the ESBWR RPS design provides a reliable single-failure-proof capability to automatically or manually initiate a reactor scram while maintaining protection against unnecessary scrams resulting from single failures. This is reliable because the RPS meets the requirements of 10 CFR 50.55(a)(h). The RPS remains single-failure-proof even when one entire division of channel sensors is out of service. Based on the above information and the staff's evaluation in Section 7 of this report, the staff finds that Issue 75 is resolved for the ESBWR design.

<u>Issue 80: Pipe Break Effects on Control Rod Drive Hydraulic Lines in the Drywells of BWR Mark I and II Containments</u>

In DCD, Tier 2, Table 1.11, the applicant addresses this issue. Issue 80 as described in NUREG-0803, "Generic Safety Evaluation Report Regarding Integrity of BWR Scram System Piping," does not apply to the ESBWR design. The ESBWR design does not include scram discharge volume piping. The water displaced by the control rod drive (CRD) during the scram will be routed to the RPV.

<u>Issue 105: Interfacing Systems LOCA at LWRs</u>

GEH addresses its evaluation of this issue in Appendix 3K of the DCD. For advanced reactor designs, the staff stated its position regarding intersystem LOCA (ISLOCA) protection in SECY-90-016, "Evolutionary Light Water (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements," as well as in SECY-93-087, "Policy, Technical Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs.

In SECY 90-16, the staff stated that 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 full RCS pressure.

In SECY-90-87, the staff further indicated that enhancements of isolation capability or the number of inter-system 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 staff further stated that 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 should provide justification demonstrating that it is not practicable to reduce the pressure challenge any further for each interfacing system and component that does not meet the RCS URS. This justification should be based upon an engineering feasibility analysis and not solely on the ratio of risk to benefit.

Accordingly, an applicant should 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:

> (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 de-energized, and (3) 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.

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 the ESBWR. GEH stated in DCD, Tier 2, Appendix 3K that the design pressure for the low-pressure piping systems that interface with the RCS pressure boundary is equal to 0.4 times the normal reactor operating pressure of 1,025 pound per square inch gauge (psig), that is, 410 psig and the minimum wall thickness of the low-pressure piping should be no less than that of a standard weight pipe. The design of the piping is to be in accordance with Section III of the ASME Boiler & Pressure Vessel Code. Furthermore, the staff will continue to require periodic surveillance and leak rate testing of the pressure isolation valves through technical specifications, as part of the ISI program. Class 300 valves will be used for the interface systems.

The staff determined that a class 300 valve is adequate for ensuring the pressure of the low-pressure piping system under full reactor pressure. Non-piping components also will be designed to 410 psig. This is accomplished in the DCD by the boundary symbols on system drawings. The boundary symbol on the system drawing applies to the piping and components that extend away from the boundary symbol, including along any

branch line, until another boundary symbol occurs on the drawing. The components include flanges and pump seals.

The staff determined that 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. Also, it is considered impractical to upgrade the suppression pool and primary containment. The suppression pool provides a low-pressure sink and the ESBWR containment is designed to 45 psig and is designed to seismic Category I requirements. Based on the staff guidance described above, GEH evaluated, in DCD Appendix 3K, the following systems that interface with the RCS to verify that they are designed for an ISLOCA "to the extent practicable":

- CRD system
- SLCS
- RWCU/SDC system
- FAPCS
- Nuclear boiler system (NBS)
- Condensate & Feed water system

The pressure of each system piping boundary was reviewed to identify where changes were needed to provide the URS protection.

Based on the preceding information and and its evaluation in Sections 3.12.6.19 and 3.12.3.6.20 of this report, the staff concludes that the ESBWR design meets the criteria of SECY-90-016 regarding ISLOCA prevention and mitigation.

Issue 106: Piping and the use of Highly Combustible Gases in Vital Areas

As discussed in NUREG-0933, this GSI provides guidance on systems and procedures for highly combustible gas used in vital areas of the plant. The staff included this GSI in its review of the ESBWR design as described in DCD, Tier 2, Section 9.5.1, and evaluated GEH's design with respect to the design of systems and the use of highly combustible gases in vital areas. Section 9.5.1 of this report documents the staff evaluation. This subject is also addressed by resolution of GL 93-006 in Section 20.5.1 of this report. Based on its review discussed in Sections 9.5.1 and 20.5.1, the staff finds that this GSI is resolved for the ESBWR design.

Issue 189: Susceptibility of Ice Condenser and Mark III Containments to Early Failure from Hydrogen Combustion During a Severe Accident

In DCD, Tier 2, Table 1.11, the applicant addresses this issue.

Prevention of hydrogen combustion in the ESBWR containment is achieved by using an inerted containment in accordance with 10 CFR 50.44(c)(2). So this issue is resolved for the ESBWR design

Issue 193: BWR ECCS Suction Concerns

GSI-193 addresses the possibility of air intrusion into the emergency core cooling system (ECCS) suction piping and the possible degradation of the ECCS pumps as the

result of cavitations. Since the ESBWR does not have ECCS pumps, the staff's finds that this GSI is not applicable to ESBWR. However the staff recognizes that there are effects on ECCS systems due to air intrusion. Non-condensable gas (i.e. air) might enter the PCCS and degrade its performance. The PCCS helps to suppress pressure increase in the drywell as well as supply liquid inventory to the reactor vessel during long-term core cooling. Section 6.2 and 6.3 of this report describes the staff's evaluation of possible air intrusion in the PCCS and its effects on containment pressure suppression and long-term core cooling, respectively. As discussed in Sections 6.2 and 6.3 of this report, the staff considers Issue 193 resolved for the ESBWR design.

20.4 Three Mile Island (TMI) Action Plan Items

This section addresses the staff's evaluation of GSIs that are categorized as "TMI action plan items" in NUREG-0933 and TMI requirements found in 10 CFR 50.34(f).

I.C.5: Procedures for Feedback of Operating Experience to Plant Staff

The regulation in10 CFR 50.34(f)(3)(i) requires the provision of 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. According to Table 1A-1 in Appendix 1A to DCD, Tier 2, Chapter 1, the ESBWR design engineers are continually involved in reviewing industry experience from sources such as NRC BLs, licensee event reports, NRC request for information letters to holders of operating licenses, *Federal Register* information, and GLs. GEH made a commitment to address these procedures in DCD, Tier 2, Sections, 13.2.3, 13.5.2 and 18.3. The staff's evaluation of this commitment appears in Section 13.2.3, 13.5.2, 18.3 of this report. Based on its review in Sections 13.2.3, 13.5.2 and 18.3, the staff finds that TMI Action Item I.C.5 is resolved for the ESBWR design.

I.C.9: Long-Term Program Plan for Upgrading of Procedures

The regulation in 10 CFR 50.34(f)(2)(ii) requires establishment of a program, to begin during construction and follow into operation, for integrating and expanding current efforts to improve plant procedure. Table 1A-1 in Appendix 1a to DCD, Tier 2, Chapter 1 addresses this issue and references DCD, Tier 2, Section 13.5. GEH addresses procedures in DCD, Tier 2, Sections 13.5, 18.9, and in the Procedures Development Implementation Plan (NEDO-33274). The staff's evaluation appears in Sections 13.5 and 18.9 of this report. Based on its review in Sections 13.5 and 18.9, the staff finds that TMI Action Item I.C.9 is resolved for the ESBWR design.

I.F.1: Expand QA List

As required by 10 CFR 52.47(a)8), an applicant for DC must demonstrate compliance with any technically relevant portions of the TMI requirements in 10 CFR 50.34(f). As required by 10 CFR 50.34(f)(3)(ii), an application must provide sufficient information to "ensure that the quality assurance (QA) list required by Criterion II, Appendix B, 10 CFR Part 50 includes all structures, systems, and components important to safety (I.F.1)." This requirement was intended to expand the QA list to ensure that non-safety-related SSCs that are important to safety are subject to appropriate QA controls.

The NRC staff reviewed the QA controls described in the ESBWR DCD Sections 17.4 and 17.5 that are applicable to the non-safety-related SSCs to verify that adequate controls are specified to ensure the reliability and availability of risk-significant, non-safety-related SSCs. The staff determined that quality programs, such as the reliability assurance program and the RTNSS, are sufficient to provide reasonable assurance that non-safety-related SSCs that are important to safety will perform satisfactorily in service. Section 17.4 of this report discusses the staff's evaluation of the ESBWR reliability assurance program. Based on the existence of alternate quality programs that provide reasonable assurance that non-safety-related SSCs important to safety will perform satisfactorily in service, the staff concludes that the ESBWR DCD meets the requirements of 10 CFR 50.34(f)(3)(ii). The staff finds that GEH has adequately addressed this TMI requirement.

Issue I.F.2: Develop More Detailed QA Criteria

As required by 10 CFR 52.47(a)(8), an applicant for DC must demonstrate compliance with any technically relevant portions of the TMI requirements set forth in 10 CFR 50.34(f). As stated in 10 CFR 50.34(f)(3)(iii), an application must provide sufficient information to demonstrate that the applicant has established a QA program that considers the following:

> (A) ensuring independence of the organization performing checking functions from the organization responsible for performing the functions; (B) performing quality assurance/quality control functions at construction sites to the maximum feasible extent; (C) including QA personnel in the documented review of and concurrence in quality related procedures associated with the design, construction and installation; (D) establishing criteria for determining QA programmatic requirements; (E) establishing qualification requirements for QA and QC personnel; (F) sizing the QA staff commensurate with its duties and responsibilities; (G) establishing procedures for maintenance of "as built" documentation; and (H) providing a QA role in design and analysis activities.

The requirements in 10 CFR 50.34(f)(3)(iii) are intended to improve the QA program to provide greater assurance that plant design, construction, and operational activities are conducted in a manner commensurate with their importance to safety. The NRC staff reviewed the requirements of 10 CFR 50.34(f)(3)(iii) to determine which requirements were technically relevant to a DC applicant. The NRC staff found that the requirements contained in 10 CFR 50.34(f)(3)(iii)(B),(D),and(E) pertain to QA activities during plant construction and operation; and therefore, were not technically relevant to a DC applicant. Similarly, the requirements of 10 CFR 50.34(f)(3)(iii)(G) are associated with control of "as-built" documentation and, therefore, are not technically relevant to DC. However, the staff found that 10 CFR(f)(3)(iii), Items A, C, F, and H are technically relevant to the DC.

As required by 10 CFR 52.47(a)(21), an application for DC must contain proposed technical resolutions of those medium- and high-priority GSIs that are identified in the version of NUREG-0933, current on the date 6 months prior to the docket date of application and that are technically relevant to the design. The intent of I.F.2 was to improve the QA program for design, construction, and operations to provide greater assurance that plant design, construction, and operational activities were conducted in a manner commensurate with their importance to safety. Item I.F.2 was to provide more

explicit and detailed criteria concerning the elements that were found in a well-conducted QA programs. As discussed in NUREG-0933, the NRC staff resolved four issues associated with Item I.F.2 by establishing new requirements in SRP Chapter 17 to reflect the 4 relevant sections of 10 CFR 50.34(f)(3)(iii). These issues include the following:

(1) Item I.F.2(2) - Include QA personnel in review and approval of plant procedures..

(2) Item I.F.2(3) - Include QA personnel in all design, construction, installation, testing, and operation activities..

(3) Item I.F.2(6) - Increase the size of the QA staff, and.

(4) Item I.F.2(9) - Clarify organizational reporting levels for the QA organization.

In DCD, Tier 2, Appendix 1A, the applicant stated that the ESBWR QA plan described in DCD, Tier 2, Section 17, meets the requirements of 10 CFR 50.34(f)(3)(iii) as they apply to the design of the ESBWR. Most requirements in 10 CFR 50.34(f)(3)(iii) overlap with the requirements in 10 CFR Part 50, Appendix B. However, some 10 CFR 50.34(f)(iii), Item I.F.2 requirements go beyond the requirements of Appendix B and are implemented separately.

As discussed above, the four new sections of NUREG-0933 covers the requirements in 10 CFR 50.34(f)(3)(iii)(A), (C), (F) and (H). NUREG-0933 classifies the remainder of the issues associated with Item I.F.2 as low-priority issues that the DC applicant is not required to address. The staff concluded that because Items I.F.2(2), (3), (6), and (9) were resolved by a revision to SRP Chapter 17 in NUREG-0800 and a review of the QA program conducted in accordance with SRP Section 17.3 would therefore verify compliance with these requirements. The staff's evaluation of GEH's QA program is provided in Chapter 17 of this report. As stated in Section 17.1.3 of this report, the staff conducted three inspections to verify GEH's implementation of their QA program for the ESBWR DC. During these inspections, the staff also verified that the requirements of 10 CFR 50.34(f)(3)(iii), identified as Items I.F.2(2), (3), (6), and (9) above, were implemented for the activities related to the ESBWR DC. Issue I.F.2 is resolved for the ESBWR design because the requirements in 10 CFR 50.34(f)(3)(iii), 10 CFR 52.479(a)(8), and 10 CFR 52.47(a)(21) are met.

II.J.3.1: Organization and Staffing to Oversee Design and Construction

The regulation in10 CFR 50.34(f)(3)(vii) requires the applicant to provide a description of the management plan for design and construction activities that includes the following:

- 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.

In Table 1A-1 in DCD, Tier 2, Appendix 1A, GEH stated that the ESBWR design team has developed a management plan for the ESBWR project which consists of a properly structured organization with open lines of communication, clearly defined responsibilities, and well-coordinated technical efforts, and appropriate control channels. GEH further stated in Table 1A-1 that the procedures to be used in the construction and operation phases of the plant are discussed in DCD, Tier 2, Section 13.5 and the startup procedures are discussed in DCD, Tier 2, Section 14.2. Sections 13.5 and 14.2 of this report present the staff's evaluation of these procedures. Based on its review in Sections 13.5 and 14.2, the staff finds that TMI Action Item II.J.3.1 is resolved for the ESBWR design.

<u>II.K.1(22): Describe Automatic and Manual Actions for Proper Functioning of Auxiliary. Heat Removal Systems When FW Not Operable</u>

Paragraph (2)(xxi) of 10 CFR 50.34(f) requires design of TMI-2 Action Item II.K.1(22): 10 CFR 50.34 (f)(2)(xxi): Design 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.

The applicant stated in DCD, Tier 2, Table 1A-1 that no short term manual actions are necessary during loss of feedwater system. Sufficient systems exist to automatically mitigate the consequences of a loss of feedwater event. DCD Chapter 15 describes an analysis peformed for a loss-of-feedwater event. If the main feedwater system is not operable, a reactor scram and initiation of the ICS will occur because either (1) A detected Loss of All Feedwater, or (2) the reactor water level will fall as a result of void collapse, boil-off and absence of makeup water. When reactor vessel water Level 3 is reached, a reactor scram is automatically initiated. Reactor water level continues to decrease because of void collapse, boil-off until the low-low level set point (Level 2) is reached. At this point, reactor isolation also occurs, but with a time delay for the MSIVs. When ICs receive an initiation signal, the condensate return valves will open in 30 seconds, placing the ICS in full operation at which time the water level stabilizes. (High pressure CRD makeup, if available, will prevent the water level from falling to a point where ADS and GDCS are initiated. If the reactor pressure is low, the low-pressure coolant injection (LPCI) mode of the FAPCS also can be used to maintain the RPV level). If the ICs are not operable, the SRVs will open on high vessel pressure approximately 5 minutes later. The SRVs open and close to maintain vessel pressure. When the reactor vessel water level reaches 1 (at this level, MSIVs close immediately, if not already closed), an ADS timer is initiated. When the ADS timer is timed out, the ADS and SLCS system actuation sequence is initiated, and the GDCS timer is initiated. When the GDCS timer is timed out, the GDCS injection valves open. Vessel pressure then decreases below the static head of GDCS, and the GDCS reflooding flow into the vessel begins. The core remains covered throughout the sequence of events and no core heatup occurs.

The staff reviewed the applicant's description above and the staff's evaluation of "Loss of feedwater flow" is described in Section 15.2.5.3 of this report. The staff noted that the ESBWR design incorporates appropriate automatic and manual action capability to ensure proper heat removal when the main feedwater system is not operable. Based on the information above and its evaluation in Section 15.2.5.3 of this report, the staff concludes that GEH has adequately addressed the requirements of this TMI-2 action item for the ESBWR design.

II.K.3(13): Separation HPCI and RCIC System initiation Levels Such that RCIC Initiation at a higher water level than HPCI/HPCS

The ESBWR is a passive plant and has no high-pressure coolant injection (HPCI), high-pressure core spray (HPCS), or reactor core isolation cooling (RCIC) systems. Therefore, TMI Action Item II.K.3(13) is not applicable to the ESBWR design. The staff concludes that this TMI Action Item II.K.3(13) is resolved for the ESBWR design.

II.K.3(28): Study and Verify Qualification of Accumulators on ADS Valves

The Applicant is required by 10 CFR 50.34(f)(1)(x), to perform a study to ensure that the ADS, valves, accumulators, and associated equipment and instrumentation will be capable of performing their intended functions during and following an accident situation. The study must give no credit for non-safety-related equipment or instrumentation, and must account for normal expected air (or nitrogen) leakage through valves. In ESBWR DCD, Tier 2, Table 1A-1, GEH discussed the resolution of this issue, also referred to as TMI Action Plan Item II.K.3(28).

In particular, GEH stated that the ESBWR ADS is made up of SRVs and squib-activated DPVs to depressurize the reactor. Following their actuation, the DPVs will not reclose until being refurbished. Each of the ADS SRVs is equipped with a pneumatic accumulator and check valve for the ADS, and manual opening functions. These accumulators ensure that the valves can be opened following failure of the gas supply to the accumulators. The accumulator capacity is sufficient for one actuation at drywell design pressure. The valves have been designed to achieve the maximum practical number of actuations consistent with state-of-the-art technology.

The DPVs are of a non-leak, non-simmer, non-maintenance design. They are straight-through, squib-actuated, non-reclosing valves with a metal diaphragm seal. The SRVs and DPVs, and their associated controls and actuation circuits are located or protected so that their function cannot be impaired by the consequences of accidents. ADS components are qualified to withstand the harsh environments postulated for design-basis accidents inside the containment, including high temperature and pressure, and radiation environments. GEH referred the reader to Sections 5.2.2.2, 6.3.2.8, and 7.3.1.1 of the DCD for additional information. The NRC staff discusses the SRVs and DPVs in Section 3.9.6 of this report. Based on its review in Section 3.9.6 of this report, the staff concludes that GEH has adequately addressed the requirements of TMI Action Item II.K(3)28.

III.D.1.1: Primary Coolant Sources Outside the Containment

The regulation in10 CFR 50.34(f)(2)(xxvi) requires the provision of leakage control and detection in the design of systems outside of containment that contain (or might contain)

accident source term radioactive materials following an accident. Applicants are required to submit a leakage control program, including initial test program, a schedule for retesting these systems, and the actions to be taken to minimize leakage from such systems. The goal is to minimize potential exposures to workers and the public and to provide assurance that excessive leakage will not prevent the use of systems needed in an emergency. In Revision 3 of the DCD, GEH cited two means of satisfying the requirements of 10 CFR 50.34(f)(2)(xxvi). In DCD, Tier 2, Table 1A-1 GEH listed Appendix J testing and the Leak Detection and Isolation System (LD&IS) as a means to satisfy issue III.D.1.1. In RAI 20-12, the staff indicated that the Appendix J testing program and LD&IS have little bearing on issue III.D.1.1 and do not satisfy the 10 CFR 50.34(f)(2)(xxvi) requirements. The staff requested that GEH address Issue III.D.1.1 without relying on Appendix J testing and the LD&IS. In response to RAI 20.0-12, the applicant agreed that it should revise the response to TMI Action Plan Item III.D.1.1 to address detecting and limiting system leakage during plant operation. GEH indicated that it would accomplish this by defining a program to reduce leakage to as-low-as practical levels for all required post accident systems outside of containment that could contain highly radioactive fluid. GEH defined the program in Revision 4, DCD, Tier 2, Table 1A-1. In this revision of Table 1A-1 it identified the ICS, FAPCS, and containment monitoring system (CMS) systems outside containment that contain or might contain source term radioactive materials following an accident.

However, after further review of Revision 4 of the DCD, the staff raised a concern as to whether GEH had identified all of the appropriate systems outside of containment that may contain source term radioactive materials following an accident. In RAI 20.0-16, the staff asked GEH to describe and justify the screening process used to determine which systems should be leak tested and meet the criteria described in the clarification section in NUREG-0737, “Clarification of TMI Action Plan Requirements,” issued November 1980 f for TMI III.D.1.1. The staff also requested that GEH identify the systems that require leak testing and justify the leak testing it proposed to perform for systems included under this item.

In response to RAI 20.0-16, the applicant indicated that the clarification section of NUREG-0737 for TMI Item III.D.1.1 provides a detailed list of systems that should be leak tested. The applicant further explained that during the preparation of the response to RAI 20.0-12, it reviewed the detailed list of systems and functions from the clarification section of NUREG-0737 and identified the corresponding ESBWR systems. As stated in its response to RAI 20-12, ICS, FAPCS, and CMS are the systems requiring leak testing. The applicant also indicated in its response to RAI 20-16 that it added the RWCU/SDC system to the list as a result of a design change to improve the post-LOCA reduction in containment pressure.

The applicant further explained that the screening process used to develop the list of systems consisted of reviewing the list of systems mentioned in the clarification section of NUREG-0737 for TMI Item III.D.1.1 and identifying the comparable ESBWR systems used to perform those design functions. In the response, GEH included a table showing how the systems in the TMI Item III.D.1.1 compare to the ESBWR design. GEH identified the systems that contain radioactive materials that are excluded from the program and discussed the justification for the exclusion in their response. GEH indicated that the NBS (the main steam and feedwater) contains radioactive materials during normal operations, but is automatically isolated during severe transients and accidents. Therefore, the NBS is not included on the list of systems requiring leak

testing under TMI Item III.D.1.1. GEH also identified the offgas system as another system that contains radioactive materials during normal operation but is excluded from the list of systems requiring leak testing. GEH indicated that, historically the offgas system has been excluded from periodic leak testing for BWRs. GEH further explained that the offgas system has a design pressure of around 2.4 megapascal gauge (350 psig), and is isolated from the RPV during serious transients and accidents after MSIV and high activity levels are alarmed in the main control room (MCR). GEH also indicated in its response that ESBWR technical specifications, DCD, Tier 2, Chapter 16, Section 5.5.2 implements the program for minimizing leakage from the systems identified. Staff found GEH response to RAI 20.0-16 acceptable because the applicant explain their screening process and identified all the appropriate systems in the ESBWR design that should be leak tested and meet the criteria described in the clarification section in NUREG-0737, "Clarification of TMI Action Plan Requirements."

SRP Section 12.3 and 12.4 states that the applicant should provide a dose assessment of major functions such as operations, radwaste handling, normal maintenance, special maintenance, refueling, and inservice inspection in accordance with the provisions of RG 8.19. Accordingly, the staff issued RAI 20.0-16, Supplement 1, requesting GEH to provide a listing of the estimated collected doses associated with the leak testing program and to verify that the dose assessment described in DCD, Tier 2, Section 12.4 account for the occupational radiation exposures associated with the leak testing program. In response to RAI 20.0-16, Supplement 1, the applicant stated that Tables 12.4-2, 12.4-3, 12.4-4, 12.4-6, and 12.4-7 in DCD, Tier 2, Section 12.4 already contain the occupational doses associated with the inspection of the systems and the leak test program. The staff found the response to RAI 20.0-16, Supplement 1, acceptable because tables in DCD, Tier 2, Section 12.4 accounted for the occupational radiation exposures associated with the leak test program. Therefore, the staff concludes that GEH has adequately addressed TMI Action Item III.D.1.1 for the ESBWR design.

20.5 Operating Experience

This section addresses the staff's evaluation of the applicant's incorporation of operating experience insights into the design.

20.5.1 Generic Letters

GL 81-20, "Safety Concerns Associated with Pipe Breaks in the BWR Scram System," April 10, 1981

This GL is not applicable to the ESBWR design.

GL 81-37, "ODYN Code Re-analysis Requirements," December 29, 1981

DCD, Tier 2, Table 1C-1, addresses this Issue.

This GL requires all BWR licensees to reanalyze the limiting transients with the ODYN code. GEH analyzes the transients in ESBWR using the TRACG04 code. This code supersedes the ODYN code and therefore, the staff finds GL 81-037 is not applicable to the ESBWR design.

GL 83-28, "Required Actions Based on Generic Implications of Salem ATWS Events," July 8, 1983

DCD, Tier 2, Table 1C-1, addresses this Issue. Based on information in NUREG-1000, the staff identified actions to be taken by applicants as a result of the Salem ATWS events. These actions address issues related to reactor trip system reliability and general management capability. The actions covered by this letter fall into the following four areas:

(1.) Post-Trip Review - —This action addresses the program, procedures, and data collection capability to ensure that the causes for unscheduled reactor shutdowns, as well as the response of safety-related equipment, are fully understood before plant restart.

(2.) Equipment Classification and Vendor Interface - —This action addresses the programs for ensuring that all components necessary for accomplishing required safety-related functions are properly identified in documents, procedures, and information-handling systems that are used to control safety-related plant activities. In addition, this action addresses the establishment and maintenance of a program to ensure that vendor information for safety-related components is complete.

(3.) Post maintenance Testing - —This action addresses post-maintenance operability testing of safety-related components.

(4.) Reactor Trip System Reliability Improvements - —This action is aimed at ensuring that vendor -recommended reactor trip breaker modifications and associated RPS changes are completed in PWRs, that a comprehensive program of preventive maintenance and surveillance testing is implemented for the reactor trip breakers in PWRs, that the shunt trip attachment activates automatically in all PWRs that use circuit breakers in their reactor trip system, and to ensure that on-line functional testing of the reactor trip system is performed on all LWRs.

As discussed in the staff evaluation of Generic Issue 75 in Section 20.3 of this report, the staff concludes that this issue is resolved.

GL 83-33, "NRC Position on Certain Requirements of Appendix R to 10 CFR Part 50," October 19, 1983

This GL was superseded by GL 86-10. For more information, refer to staff's evaluation of GL 86-10 later in this section.

GL 84-23, "Reactor Vessel Water Level Instrumentation in BWRs," October 26, 1984

DCD, Tier 2, Table 1C-1, addresses this issue.

This GL identifies the following potential improvement categories for the operating reactors:

- Improvements to plant(s) that will reduce level indication errors caused by high drywell temperature. These improvements include

prevention of reference leg overheating or reduction of the vertical drops in the drywell. (Vertical drop should be measured from the condensation pot to the drywell exit point. Maximum drop would allow an indicated level at the bottom of the normal operating range when the actual level is just above lower tap for worst flashing condition.) Those plants for which the vertical drop in the drywell has already been minimized will not have to make additional changes for the drywell heating effect.

- Review of plant experience relating to mechanical level indication equipment. Plant experience shows mechanical level equipment is more vulnerable to failure or malfunction than analog equipment. A number of plants have already connected analog trip units to their level transmitters to improve reliability and accuracy. Those plants that use mechanical level indication should replace the mechanical level indication equipment with analog level transmitters unless operating experience confirms high reliability.

- Changes to the protection system logic that may be needed for those plants in which operator action may be required to mitigate the consequences of a break in a reference leg and a single failure in a protection system channel associated with an intact reference leg. Changes will generally result in additional transmitters to satisfy the single failure criterion.

The staff reviewed the ESBWR design for the reactor vessel water level measurement system requirements specified in GL 84-23. In RAI 20.0-7, the staff stated that confirmation was needed regarding the adequacy of the differential pressure method for the RPV level measurement and asked the applicant to explain in detail the systems design, operation and operator actions during transients, and demonstrate that the RPV level system was robust. In response to RAI 20.0-7, the applicant submitted additional information about the RPV level measurement system. The applicant stated that in the ESBWR design, the direct RPV water level measurement instrumentation system detects conditions of adequate core cooling. The RPV water level is the primary variable in the BWR for indicating the availability of adequate core cooling. Four independent divisions of differential pressure sensing instruments provide water level sensing. They are designed to be adequately redundant and unambiguous so that ESBWR level indication is accurate and reliable. Each division of level sensing instruments includes a differential pressure instrument for one of four measurement regions including fuel zone, wide range, narrow range (primarily used for power operation level indication and feedwater control logic), and shutdown range (used during refuel operations). Each division has its own set of RPV sensing line nozzle connections. RAI 20.0-7 was resolved because the ESBWR has addressed the issue of false high water level indication upon vessel depressurization or as the result of events that cause vessel pressure reduction transients. GEH also provided the instrument line vertical drop in conformance with the guidelines in RG 1.151.

The staff concluded that when implemented in the ESBWR, these improvements will increase assurance that the level instrumentation will detect inadequate core cooling, as specified in NUREG- 0737, Item II.F.2, and thereby satisfy this requirement. Section 7.1.1.3.4 of this report presents the staff evaluation of TMI-2 Action Item II.F.2. Based on the staff's evaluation of Item II.F.2, the staff considers this GL resolved for the ESBWR design.

GL 85-01, "Fire Protection Policy Steering Committee Report," January 9, 1985

The NRC never formally issued this GL. The content of the draft version of GL 85-01 subsequently became GL 86-10, which is addressed below. Therefore, the staff considers this GL resolved for the ESBWR design.

GL86-10, "Implementation of Fire Protection Requirements," April 24 1986

This GL applies to existing plants licensed before to January 1, 1979. SRP Section 9.5.1 and RG 1.189, "Fire Protection for Nuclear Power Plants," provide the corresponding guidance for new reactors. Section 9.5.1 of this report addresses the staff evaluation of the applicant's compliance with the fire protection requirements. Therefore, based on the staff's evaluation in Section 9.5.1 of this report, the staff considers this GL is resolved for the ESBWR design.

Supplement 1 to GL 86-10, "Fire Endurance Test Acceptance Criteria for Fire Barrier Systems Used to Separate Redundant Safe Shutdown Trains Within the Same Fire Area," March 25, 1994

This supplement provides guidance on the qualification testing of fire barrier systems. RG 1.189 includes the guidance provided by this supplement. However, the ESBWR DCD does not identify any applications for these systems. Any proposed use of such systems will be identified by the applicant and the design evaluated by the staff during the review of the COL application in response to COL Information Item 9.5.1-5-A. Therefore, the staff considers this GL resolved for the ESBWR design.

GL 87-06, "Periodic Verification of Leak Tight Integrity of Pressure Isolation Valves," March 13 1987

ESBWR DCD, Tier 2, Table 1C-1 indicates that GL 87-06, does not apply to the ESBWR. In its response to RAI 20.0-11, the applicant stated that the basis for this statement is provided in Section 3K.2 of DCD, Tier 2, Appendix 3K, which specifies that the periodic surveillance and leak rate testing requirements for high-pressure to low-pressure isolation valves do not apply to the ESBWR, because the ESBWR design does not contain a pressure isolation valve between the RCPB and a low pressure piping system. GEH stated that it would revise Table 1C-1 to reference DCD, Tier 2, Appendix 3K, to support the statement that the GL is not applicable to the ESBWR. As a result, ESBWR DCD, Tier 2, Revision 6, Table 1C-1 references Appendix 3K in an acceptable manner. Therefore, RAI 20.0-11 is closed. The staff considers this GL resolved for the ESBWR design.

GL 88-18, “Plant Record Storage on Optical Disks,” October 20, 1988

The purpose of GL 88-18 is to inform all licensees that the staff approves the use of plant record storage on optical disks for record keeping when appropriate QA controls are applied. In DCD, Tier 2, Appendix 1C, the applicant stated that it is the responsibility of the COL applicant and licensee to supplement DCD Subsection 17.1.17, which states that the GEH QA Program Description (NEDO-11209-04A) establishes control requirements of QA records used during the design of the ESBWR. The NRC staff agrees that NEDO-11209-04A establishes control requirements for QA records and that the COL applicant and licensee will also need to establish control requirements for QA records consistent with the guidance in GL88-18, if applicable. This will be addressed by COL Information Items 17.2-1-A, 17.2-2-A, 17.3-1-A. Therefore, the staff considers this GL resolved for the ESBWR design because COL applicants will address this GL.

GL 89-02, “Actions To Improve the Detection of Counterfeit and Fraudulently Marketed Products,” March 21, 1989

The purpose of GL 89-02 is to share with all licensees some of the elements of programs that appear to be effective in detecting counterfeit or fraudulently marketed products and in ensuring the quality of vendor products.

In DCD, Tier 2, Appendix 1C, the applicant stated that it is the COL applicant and the licensee’s responsibility to address the guidance of GL 89-02. The NRC staff agrees with the applicant that GL 89-02 is not applicable to the ESBWR DCD review since this is a procurement issue related to components, which is the responsibility of the COL applicant and licensee. COL applicants will consider GL 89-02 when addressing COL Information Items 17.2-1-A, 17.2-2-A and 17.3-1-A. The staff considers this GL resolved for the ESBWR design because it will be addressed by these COL information items in DCD, Tier 2, Sections 17.2 and 17.3.

GL 89-04, “Guidance on Developing Acceptable IST Programs,” April 3, 1989

The NRC staff issued GL 89-04, and its Supplement 1 to provide information for nuclear power plant licensees to use in satisfying the NRC regulations for IST programs. In response to RAI 20.0-9, the applicant stated that ESBWR DCD, Tier 2, Appendix 1C, “Industry Operating Experience,” would be revised to clarify the GLs and BLs within the scope of the COL application. As a result, ESBWR DCD, Revision 6, Tier 2, Table 1C-1, “Operating Experience Review Results Summary – Generic Letters,” lists GL 89-04 and its Supplement 1 for consideration by the COL applicant, and refers to ESBWR DCD, Tier 2, Section 3.9.6, “Inservice Testing of Pumps and Valves,” to address the issues in this GL. The NRC staff considers the reference in Table 1C-1 to the appropriate DCD section for GL 89-04 for consideration by the COL applicant as specified in COL Information Item 3.9.9-3-A, to be acceptable because the staff has revised SRP Section 3.9.6 to update the guidance for staff’s review of IST programs described by DC and COL applicants, and to incorporate lessons learned from GL 89-04 and other applicable GLs that address nuclear power plant operating experience Therefore, RAI 20.0-9 is closed. The staff considers this GL resolved for the ESBWR design.

GL 89-18, “Resolution of Unresolved Safety Issue A-17, ’System Interactions in Nuclear Power Plants,’” September 6, 1989

The discussion of task action plan Item A-17 is addressed in Section 20.2 of this report. Based on the staff’s evaluation of USI A-17 in Section 20.2 of this report, the staff concludes that this GL is resolved.

GL 91-05, “Licensee Commercial-Grade Procurement and Dedication Programs,” April 9, 1991

The purpose of GL 91-05 is to allow licensees sufficient time to fully understand and implement guidance developed by industry to improve procurement and commercial grade dedication programs. In DCD, Tier 2, Appendix 1C, the applicant stated that it is the responsibility of the licensee to address the guidance of GL 91-05.

The NRC staff agrees with the applicant that GL 91-05 does not apply to the ESBWR DCD review because this is a procurement issue and GEH is not procuring any commercial grade items as part of the DC. The licensee is responsible for procurement issues, which may include commercial grade dedication. GL 91-05 will be addressed by COL Information Items 17.2-1-A, 17.2-2-A and 17.3-1-A. Therefore, the staff considers this GL resolved for the ESBWR design because it will be addressed by the COL applicants.

GL 91-16, “Licensed Operators’ and Other Nuclear Facility Personnel Fitness for Duty,” October 3, 1991

GL 91-016 does not apply to the ESBWR DCD review because the subject matter of GL 91-016, the fitness-for-duty of licensed operators and other nuclear facility personnel is not relevant to DCs.

GL 92-04, “Resolution of the Issues Related to Reactor Vessel Water Level Instrumentation in BWRs Pursuant to 10 CFR 50.54(f),” August 19, 1992

DCD, Tier 2, Table 1C-1, addresses this issue. The NRC issued the GL 92-04 to request information regarding the adequacy of and corrective actions for BWR water level instrumentation with respect to the effects of noncondensable gases on system operation. As discussed in NRC IN No. 92-54 “Level Instrumentation Inaccuracies Caused by Rapid Depressurization,” dated July 24, 1992, the staff was concerned that noncondensable gases may become dissolved in the reference leg of BWR water level instrumentation and lead to a false high level indication after a rapid depressurization event. The dissolved gases, which accumulate over time during normal operation, can rapidly come out of solution during depressurization and displace water from the reference leg. A reduced reference leg level will result in a false indication of a high level. This is important to safety because water level signals are used for actuating automatic safety systems and to guide operators during and after an event.

The staff later issued BL 93-03, "Resolution of Issues Related to Reactor Vessel Water Level Instrumentation in BWRs," dated May 28, 1993, requesting hardware modifications for operating reactors. As described in DCD, Tier 2, Section 7.7.1.2.2, GEH 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 ESBWR RPV level instrumentation system design incorporates the modifications recommended by the staff and the staff finds that the design addresses the concerns identified in GL 92-04 and BL 93-03. The staff concludes that this GL is resolved for the ESBWR design.

GL 92-08, "Thermo-Lag 330-1 Fire Barriers," December 17, 1992 (BL 92-001)

This GL provided information on testing performed to determine the fire endurance capability of Thermo-Lag 330-1 fire barriers. The ESBWR DCD does not identify any applications for this type of fire barrier system. Any proposed use of such systems will be identified by COL applicants and the design evaluated by the staff at the COL application stage in response to COL Information Item 9.5.1-5-A. Therefore, this GL is resolved for the ESBWR design.

GL 93-06, "Research Results on Generic Safety Issue 106, Piping and the Use of Highly Combustible Gases in Vital Areas," October 25, 1993

This GL provides guidance on meeting GSI 106. The staff included GL 93-06 in its review of the ESBWR design and evaluated the GEH design with respect to the use of highly combustible gases in vital areas. DCD, Tier 2, Table 9A.5-1 provides the locations and amounts of highly combustible gases, and describes safety features used in the ESBWR design to contain and mitigate a potential explosion and fire in areas with highly combustible gases. The staff has determined that the elements of the ESBWR design that contain and mitigate the hazards of highly combustible gases are adequate. The staff's review of this information is provided in Section 9.5.1 of this report. Based on the information above and the staff's evaluation in Section 9.5.1, the staff considers this GL resolved for the ESBWR design.

GL 2006-03, "Potentially Nonconforming Hemyc and MT Fire Barrier Configurations," April 10, 2006

The ESBWR DCD does not identify any applications for this type of fire barrier system. Any proposed use of such systems will be identified by the applicant and the design evaluated by the staff at the COL application stage. Therefore the staff considers this GL resolved for the ESBWR design.

20.5.2 Bulletins

Bulletin 80-01, "Operability of ADS Valve Pneumatic Supply," January 11, 1980

In Table 1C-2 of DCD Tier 2, GEH discusses its consideration of BL 80-01, "Operability of ADS Valve Pneumatic Supply," which specified that nuclear power plant licensees determine that hard-seat check valves were installed to isolate the ADS from the pneumatic supply system, determine if periodic leak tests were performed to assure availability emergency pneumatic supply, review seismic qualification of ADS pneumatic supply system, and evaluate ADS operability and take appropriate action. In Table 1C-2, GEH indicates that the design of the pneumatic supply to the ADS valves addresses the concerns with the potential loss of pneumatic pressure. In addition, the ESBWR has diverse means of depressurizing the RPV using the DPVs. The NRC staff has reviewed the GEH response to BL 80-01. In addition to the indicated GEH

response, ESBWR DCD, Tier 2 Section 3.9.3.3.5, “Valve Operability Assurance,” specifies the application of ASME Standard QME-1-2007, “Qualification of Active Mechanical Equipment used in Nuclear Power Plants,” for valve designs not previously qualified, and requires the application of key aspects of the standard for valves previously qualified. Further, the inservice testing program will assess the operational readiness of the SRVs and DPVs on a periodic basis as discussed in Section 3.9.6 of this report. Because of the ESBWR DCD provisions for valve design and qualification in accordance with ASME Standard QME-1-2007, and because COL Information Item 3.9.9-3-A requires COL applicants to provide a full description of the IST Program, the staff considers this BL resolved for the ESBWR design.

BL 80-25, “Operating Problems with Target Rock Safety-Relief Valves at BWRs,” December 19, 1980

In Table 1C-2 of DCD Tier 2, GEH discussed the evaluation of BL 80-25 for the ESBWR design. In particular, GE stated that this BL does not apply to the ESBWR design because a different valve type is used and referenced Section 5.4.13 of the DCD. The NRC staff discusses the SRVs in Section 3.9.6 of this report.

BL85-03, “Motor Operated Valve Common Mode Failures During Plant Transient Due to Improper Switch Settings,” November 15, 1985

In a previous revision to ESBWR DCD, Tier 2, Table 1C-1, “Operating Experience Review Results Summary – IE Bulletins,” GEH indicated that BL 85-03, and its Supplement 1 are not applicable to the ESBWR in that they involved an administrative, maintenance, or procurement communication. In its response to RAI 20.0-10, the applicant stated that it would revise Tables 1C-1 and 1C-2 to clarify the applicability of GLs and BL to COL applications, including BL 85-03 and its Supplement 1. ESBWR DCD, Tier 2, Revision 6, Tables 1C-1 and 1C-2 have incorporated these changes in an acceptable manner, including indication that BL 85-03 and its Supplement 1 are applicable to the COL application and are addressed in Section 3.9.6. Therefore, RAI 20.0-10 is closed. Because of the ESBWR DCD provisions for valve design and qualification in accordance with ASME Standard QME-1-2007, and because COL Information Item 3.9.9-3-A requires COL applicants to provide a full description of the IST Program (including the MOV Testing Program), the staff considers this BL resolved for the ESBWR design.

BL 86-01, “Minimum Flow Logic Problems that Could Disable RHR Pumps,” May 23, 1986

DCD, Tier 2, Table 1C-2 addresses this issue.

In this BL the staff addressed concerns regarding RHR pumps, which also function as Low Pressure Injection pumps during LOCA, running dead-headed due to a postulated single failure of a flow sensing instrument. The ESBWR RHR pumps do not function as ECCS pumps; therefore this item is not applicable.

BL 93-03, "Resolution of Issues Related to Reactor Vessel Water Level Instrumentation in Boiling Water Reactors," May 28, 1993

This subject is also addressed by resolution of GL 92-04 in Section 20.5.1 of this report. Based on its review discussed in Sections 20.5.1 of this report, the staff finds that this BL is resolved for the ESBWR design.

20.6 Conclusion

On the basis of its of review of the BLs and GLs issued between January 1, 1980 and February 24, 2005, and the review of the ESBWR DCD Tier 2, the staff concludes that GEH adequately addressed operating experience in the ESBWR design as required by 10 CFR 52.47(a)(22). The applicant also addressed all the relevant TMI action plans items found in 10 CFR 50.34, "Contents of Application's Technical Information," and proposed technical resolutions of the USIs and medium- and high-priority GSIs as defined in NUREG-0933 "A Prioritization of Generic Safety Issues." The staff concludes that the applicant has adequately demonstrated that the ESBWR design complies with the requirements of 10 CFR 52.47(a)(8) and 10 CFR 52.47(a)(21).