Enclosure 2

MFN 16-065

GEH's Response to NRC's Request for Additional Information

– ABWR DCD Revision 6 Markups

Table 2.4.1 Residual Heat Removal System (Continued)

Inspections, Tests, Analyses and Acceptance Criteria		
Design Commitment	**Inspections, Tests, Analyses**	**Acceptance Criteria**
4. continued c. The RHR pumps have sufficient NPSH.	4. continued c. Inspections, tests and analyses will be performed upon the as -built RHR System. NPSH tests of the pumps will be performed in a test facility. The analyses will consider the effects of: – Pressure losses for pump inlet piping and components. – Suction from the suppression pool with water level at the minimum value. – 5O% blockage of pump suction strainers. – Design basis fluid temperature (100°C). – Containment at atmospheric pressure.	4. continued c. The available NPSH exceeds the NPSH required by the pumps.

- Confirm vertical and horizontal separation between the SRV Quencher and RHR Suction Strainer

Table 2.4.2 High Pressure Core Flooder System (Continued)

Inspections, Tests, Analyses and Acceptance Criteria		
Design Commitment	**Inspections, Tests, Analyses**	**Acceptance Criteria**
3. continued d. The HPCF System flow in each division is not less than a value corresponding to a straight line between a flow of 182 m^3/h at a differential pressure of 8.12 MPa and a flow of 727 m^3/h at a differential pressure of 0.69 MPa.	3. continued d. Tests will be conducted on each division of the as-built HPCF System in the HPCF high pressure flooder mode. Analyses will be performed to convert the test results to the conditions of the Design Commitment.	3. continued d. The converted HPCF flow satisfies the following: The HPCF System flow in each division is not less than a value corresponding to a straight line between a flow of 182 m^3/h at a differential pressure of 8.12 MPa and a flow of 727 m^3/h at a differential pressure of 0.69 MPa.
e. The HPCF System has the capability to deliver at least 50% of the flow rates in item 3d with 171°C water at the pump suction.	e. Analyses will be performed of the as-built HPCF System to assess the system flow capability with 171°C water at the pump suction.	e. The HPCF System has the capability to deliver at least 50% of the flow rates in item 3d with 171°C water at the pump suction.
f. System flow into the reactor vessel is achieved within 16 seconds of receipt of an initiation signal and power available at the emergency busses.	f. Tests will be conducted on each HPCF division using simulated initiation signals.	f. The HPCF System flow is achieved within 16 seconds of receipt of a simulated initiation signal.
g. The HPCF pumps have sufficient NPSH available at the pumps.	g. Inspections, tests and analyses will be performed upon the as-built system. NPSH tests of the pumps will be performed in a test facility. The analyses will consider the effects of: – Pressure losses for pump inlet piping and components. – Suction from the suppression pool with water level at the minimum value. – 50% minimum blockage of the pump suction strainers.	g. The available NPSH exceeds the NPSH required by the pumps.

- Confirm vertical and horizontal separation between the SRV Quencher and HPCF Suction Strainer

Table 2.4.4 Reactor Core Isolation Cooling System (Continued)

Inspections, Tests, Analyses and Acceptance Criteria		
Design Commitment	**Inspections, Tests, Analyses**	**Acceptance Criteria**
3. continued j. The RCIC System pump has sufficient NPSH. (6) Confirm vertical and horizontal separation between the SRV Quencher and RCIC Suction Strainer	3. continued j. Inspections, tests, and analyses will be performed based upon the as-built system. NPSH tests of the pump will be performed at a test facility. The analyses will consider the effects of: (1) Pressure losses for pump inlet piping and components. (2) Suction from suppression pool with water level at the minimum value. (3) 50% blockage of pump suction strainers. (4) Design basis fluid temperature (77°C). (5) Containment at atmospheric pressure.	3. continued j. The available NPSH exceeds the NPSH required by the pump.
k. The RCIC System operates for a period of at least 2 hours under conditions of no AC power availability and no other simultaneous failures, accidents, or other design basis conditions.	k. Inspections and analyses of the as-built RCIC and supporting systems will be performed to determine RCIC capability.	k. The RCIC System can operate for a period of at least 2 hours under conditions of no AC power availability and no other simultaneous failures, accidents, or other design basis conditions.
l. The RCIC can be started by local operation of the RCIC System components outside the MCR.	l. Tests will be conducted locally on RCIC System components required for system operation.	l. RCIC System components required for system operation can be actuated locally.
4. If a system initiation signal occurs during the full flow test mode, the RCIC System automatically aligns to the RPV water makeup mode.	4. Test will be conducted using simulated initiation signals.	4. The RCIC System automatically aligns to RPV water makeup mode from test mode upon receipt of an initiation signal.

1.8 Conformance with Standard Review Plan and Applicability of Codes and Standards

1.8.1 Conformance with Standard Review Plan

This subsection provides the information required by 10CFR50.34(g) showing conformance with the Standard Review Plan (SRP). The summary of differences from the SRP section is presented by SRP section in Tables 1.8-1 through 1.8-18. (See Subsection 1.8.4.1 for COL license information.)

1.8.2 Applicability of Codes and Standards

Standard Review Plans, Branch Technical Positions, Regulatory Guides and Industrial Codes and Standards which are applicable to the ABWR design are provided in Tables 1.8-19, 1.8-20 and 1.8-21. Applicable revisions are also shown.

1.8.3 Applicability of Experience Information

Add: end of June 2016

Experience information is routinely made available and distributed to design personnel in the design process. ~~Nuclear field experience is maintained in hard copy form in functional component and library files and in the GEH world-wide computer retrieval system.~~

Generic Letters and IE Bulletins, Information Notices and Circulars covering the decade including 1980 through the ~~current issues (late 1991)~~ were reviewed for applicability to the ABWR design. The review was enhanced by associating related experiences and tracing referenced occurrences. This was accomplished starting with the current issues of the Generic Letters and proceeding back into the decade. The Circulars, Bulletins and Notices were reviewed in that order. Interfacing experience was included in the review. The selection of ABWR information was based on the significance to future design and operation guidance. Included is a list of NUREGs related to the closing of current safety issues. Experience that resulted in applicable rules, codes and standards was not repeated. Table 1.8-22 lists the experience information that has been included in the ABWR design or impacts the COL applicant. (See Subsection 1.8.4.2 for COL license information.)

A systematic procedure encompassing available resources was used to identify the applicability of experience information resulting in Table 1.8-22. Engineering management surveyed the indices of annual experience information to identify those very likely to be applicable to the ABWR. The remaining potentially applicable experiences were reviewed individually. Experience information not deemed applicable to the ABWR design (issues pertaining to other reactor types, scram discharge volume, etc.) were not included in Table 1.8-22. The experience information categories applicable to the ABWR design in Table 1.8-22 include experience information accommodated by a design change, covered by review of USIs/GSIs or an issue that impacts the ABWR design but must be addressed by the COL applicant. This latter category is included as COL license information.

Experiences related to identified regulatory or industry developed resolutions were eliminated to avoid repetition except for selected experiences that have a nuisance potential for reoccurring. Lead system engineers classified the more complex experiences.

Reference to the new or novel design features used in the ABWR are provided below:

Feature	Tier 2 Section
Fine Motion Control Rod Drive	4.6
Internal Reactor Pumps	5.4.1
Multiplexing	7A.2
Digital/Solid-State Control	7A.7
Lower Drywell Flooder	9.5.1.2

Add: GEH also reviewed international operating experience related to the ABWR design. Experiences related to the ABWR licensing effort in the UK were reviewed for applicability to the ABWR Certified design. The UK Office of Nuclear Regulations (ONR) issued Regulatory Issues (RI) and Regulatory Observations (RO) as a result of the UK's Generic Design Assessment (GDA) of the ABWR during UK licensing review. These RIs and ROs were systematically reviewed and evaluated by ABWR subject matter experts for applicability to the ABWR standard design. The conclusion of the evaluation is that none of the RIs and ROs requires a design change to the ABWR standard design. The RIs and ROs are either unique to the UK licensing process, are already addressed in the ABWR standard design, or are the result of unique UK licensing regulations.

1.8.4 COL License Information

1.8.4.1 SRP Deviations

The SRP sections to be addressed by the COL applicant are indicated in the comments column of Table 1.8-19 as "COL Applicant". Where applicable the COL applicant will provide the information required by 10CFR50.34(g) similar to Tables 1.8-1 through 1.8-18 (see Subsection 1.8.1).

1.8.4.2 Experience Information

The experience information to be addressed by the COL applicant are indicated in the comment column of Table 1.8-22 as "COL Applicant" (see Subsection 1.8.3).

Table 1.8-22 Experience Information Applicable to ABWR (Continued)

No.	Issue Date	Title	Comment
89-02	3/21/89	Actions to Improve the Detection of Counterfeit and Fraudulently Marketed Products Past Related Correspondence: EPRI-NP-5652, “Guideline for the Utilization of Commercial-Grade Items in Nuclear Safety-Related Applications”. Bulletins 87-02 and Supplements 1 and 2, 88-05 and Supplements 1 and 2, 88-10 IE Notices 87-66, 88-19, 88-35, 88-46 and Supplements 1 and 2, 88-48 and Supplement 1, 88-97	COL Applicant
Insert 1			
89-04	4/3/89	Guidance on Developing Acceptable Inservice Testing Program	COL Applicant
89-06	4/12/89	Task Action Plan Item I.D.2 – Safety Parameter Display System CFR 50.54(f)	1A.2.3
89-07	4/28/89	Power Reactor Safeguards Contingency Planning for Surface Vehicle Bombs	
89-07 Supp I	4/21/89	Power Reactor Safeguards Contingency Planning for Surface Vehicle Bombs	
89-08	5/2/89	Erosion/Corrosion-Induced Pipe Wall Thinning	
Insert 2			
89-10	6/28/89	Safety-Related Motor-Operated Valve Testing and Surveillance	COL Applicant
89-13	7/18/89	Service Water System Problems Affecting Safety-Related Equipment	COL Applicant
Insert 3			
89-14	8/21/89	Line Item Improvements in Technical Specifications Removal of the 3.25 Limit on Extending Surveillance Intervals	
89-15	8/21/89	Emergency Response Data System	COL Applicant
89-16	9/1/89	Installation of a Hardened Wetwell Vent	
89-18	9/6/89	Resolution of USI A-17, Systems Interactions	Subsection 19B.2.59
89-19	9/20/89	Request for Action Related to Resolution of Unresolved Safety Issue A-47, “Safety Implication of Control Systems in LWR Nuclear Power Plants”, Pursuant to 10CFR50.54(f)	Subsection 19B.2.17
89-22	10/19/89	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	
90-09	12/11/90	Alternative Requirements for Snubber Visual Inspection Intervals and Corrective Actions	
91-03	03/06/91	Reporting of Safeguards Events	COL Applicant

Table 1.8-22 Experience Information Applicable to ABWR (Continued)

Insert 4
Insert 5
Insert 6
Insert 7

No.	Issue Date	Title	Comment
91-04	04/02/91	Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle	
91-05	04/04/91	Licensee Commercial Grade Procurement and Dedication Programs	
91-06	04/29/91	Resolution of Generic Issue A-30, “Adequacy of Safety-Related DC Power Supplies”, Pursuant to 10CFR50.54(f)	Subsection 19B.2.52
91-10	07/08/91	Explosive Searches at Protected Area Portals	COL Applicant
91-11	07/19/91	Resolution of Generic Issue 48, “LCOs for Class 1E Tie Breakers”, Pursuant to 10CFR50.54(f)	Subsection 19B.2.52
91-14	09/23/91	Emergency Telecommunications	
91-16	10/03/91	Licensed Operators' and Other Nuclear Facility Personnel Fitness for Duty	COL Applicant
91-17	10/17/91	Generic Safety Issue 29, “Bolting Degradation or Failure in Nuclear Power Plants”	Subsection 19B.2.62
92-04	8/19/92	Resolution of the Issues Related to Reactor Vessel Level Instrumentation in BWRs Pursuant to 10CFR50.54(f)	
97-04	10/7/97	Assurance of Sufficient Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal Pumps	COL Applicant
98-04	7/14/98	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	COL Applicant
Type: IE Bulletins			
79-02	3/8/79	Pipe Support Base Plate Designs Using Concrete Expansion Anchor Bolts	
79-08	4/14/79	Events Relevant to BWR Identified During TMI Incident	
80-01	1/11/80	ADS Valve Pneumatic Supply	
80-03	2/6/80	Loss of Charcoal from Absorber Cells	
80-05	3/10/80	Vacuum Condition Resulting in Damage to Chemical and Volume Control System (CVCS) Holdup Tanks	COL Applicant
80-06	3/13/80	ESF Reset Controls	
80-08	4/7/80	Containment Lines Penetration Welds	COL Applicant
80-10	5/6/80	Non-Radioactive System – Potential for Unmonitored Release	COL Applicant
80-12	5/9/80	Decay Heat Removal System Operability	COL Applicant

Table 1.8-22 Experience Information Applicable to ABWR (Continued)

Insert 8

No.	Issue Date	Title	Comment
87-02, Supp 1	4/22/88	Fastener Testing to Determine Conformance with Applicable Material Specifications Past Related Correspondence: IE Notice 88-17	COL Applicant
87-02, Supp 2	6/10/88	Fastener Testing to Determine Conformance with Applicable Material Specifications	COL Applicant
88-04	5/5/88	Potential Safety-Related Pump Loss Past Related Correspondence: IE Notice 87-59	
88-07	6/15/88	Power Oscillations in Boiling Water Reactors (BWRs) Past Related Correspondence: IE Notice 88-39	
88-07, Supp 1	12/30/88	Power Oscillations in Boiling Water Reactors (BWRs)	Subsections 7.1.2.6.1.4 and 7.1.2.1.1.2.2
90-01	03/09/90	Loss of Fill-Oil in Transmitters Manufactured by Rosemount	
90-02	03/20/90	Loss of Thermal Margin Caused by Channel Box Bow	
91-01	10/18/91	Reporting Loss of Criticality Safety Controls	
2012-01	07/27/12	Design Vulnerability in Electric Power System	COL Applicant
Type: IE Information Notices			
79-22	9/14/79	Qualifications of Control Systems	COL Applicant
80-12	3/31/80	Instrumentation Failure Causes PORV Opening	
80-21	5/16/80	Anchorage and Support of Safety-Related Electrical Equipment	
80-22	5/28/80	Breakdowns in Contamination Control Programs	COL Applicant
80-40	11/7/80	Excessive N_2 Supply Pressure	
80-42	11/24/80	Effect of Radiation on Hydraulic Snubber Fluid	
81-05	3/13/81	Degraded DC Systems at Palisades	COL Applicant
81-07	3/16/81	Potential Problem with Water Soluble Purge Dam Materials Used During Inert Gas Welding	COL Applicant
81-10	3/25/81	Inadvertent Containment Spray	COL Applicant

No.	Issue Date	Title	Comment
Insert 1			
89-04 Supp. 1	4/4/95	Guidance on Developing Acceptable Inservice Testing Programs	COL Applicant
Insert 2			
89-10 Supp. 1	6/13/90	Supplement 1 to Generic Letter 89-10: Results of the Public Workshops	COL Applicant
89-10 Supp. 3	10/25/90	Generic Letter 89-10, Supplement 3, Consideration of the Results of NRC-Sponsored Tests of Motor-Operated Valves	COL Applicant
89-10 Supp. 4	2/12/92	Generic Letter 89-10, Supplement 4, Consideration of Valve Mispositioning in Boiling Water Reactors	COL Applicant
89-10 Supp. 5	6/28/93	Generic Letter 89-10, Supplement 5, Inaccuracy of Motor-Operated Valve Diagnostic Equipment	COL Applicant
89-10 Supp. 6	3/8/94	Generic Letter 89-10, Supplement 6, Information on Schedule and Grouping, and Staff Responses to Additional Public Questions	COL Applicant
Insert 3			
89-13 Supp. 1	4/4/90	Service Water System Problems Affecting Safety-Related Equipment	COL Applicant
Insert 4			
91-15	9/23/91	Operating Experience Feedback Report, Solenoid-Operated Valve Problems at US Reactors	3.9.6.2.3, COL Applicant
Insert 5			
92-01 Rev. 1 Supp.1	5/19/95	Reactor Vessel Structural Integrity	5.2

No.	Issue Date	Title	Comment
Insert 6			
92-08	12/17/92	Thermo-Lag 330-1 Fire Barriers	COL Applicant
93-05	9/27/93	Line-Item Technical Specifications Improvements to Reduce Surveillance Requirements for Testing During Power Operation	Chapter 16
93-06	10/25/93	Research Results on Generic Safety Issue 106, 'Piping and the use of Highly Combustible Gases in Vital Areas"	9.5
93-08	12/29/93	Relocation of Technical Specification Tables of Instrument Response Time Limits	COL Applicant
94-01	5/31/94	Removal of Accelerated Testing and Special Reporting Requirements for Emergency Diesel Generators	COL Applicant
94-02	7/11/94	Long-Term Solutions and Upgrade of Interim Operating Recommendations for Thermal-Hydraulic Instabilities in BWRs	4.4.3.7
94-03	7/25/94	Intergranular Stress Corrosion Cracking of Core Shrouds in Boiling Water Reactors	5.2.3.4.1
95-07	8/17/95	Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves	3.9.6.2.3, COL Applicant
95-10	12/15/95	Relocation of Selected Technical Specifications Requirements Related to Instrumentation	Chapter 16
96-01	1/10/96	Testing of Safety-Related Logic Circuits	7.1.2.1.6, Chapter 16, COL Applicant
96-03	1/31/96	NRC Generic Letter 96-03: Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits	Chapter 16, COL Applicant
96-04	6/26/96	Boraflex Degradation in Spent Fuel Pool Storage Racks	16.4.3.1
96-05	9/18/96	Periodic Verification of Design-Basis Capability of Safety-Related Power-Operated Valves	3.9.6, 3.9.7.3 COL Applicant
96-06	9/30/96	Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions	6.2, 3.11
96-06 Supp. 1	11/13/97	Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions	6.2, 3.11

No.	Issue Date	Title	Comment
<u>Insert 7</u>			
98-05	11/10/98	Boiling Water Reactor Licensees Use of the BWRVIP-05 Report to Request Relief from Augmented Examination Requirements on Reactor Pressure Vessel Circumferential Shell Welds	5.3, COL Applicant
99-02	06/03/99	Laboratory Testing of Nuclear-Grade Activated Charcoal	16.5.5.2.7
99-02 Errata	08/23/99	Laboratory Testing of Nuclear-Grade Activated Charcoal	16.5.5.2.7
03-01	06/12/03	Control Room Habitability	6.4
06-02	02/01/06	Grid Reliability and the Impact on Plant Risk and the Operability of Offsite Power	8.2, COL Applicant
06-03	04/10/06	Potentially Nonconforming Hemyc and MT Fire Barrier Configurations	9.5.13.9, COL Applicant
07-01	02/07/07	Inaccessible or Underground Power Cable Failures that Disable Accident Mitigation Systems or Cause Plant Transients	COL Applicant
08-01	01/11/08	Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal, and Containment Spray Systems	5.4.8, 19B.2.2
16-01	4/7/2016	Monitoring of Neutron-Absorbing Materials in Spent Fuel Pools	16.4.3.1

No.	Issue Date	Title	Comment
Insert 8			
93-02	05/11/93	Debris Plugging of Emergency Core Cooling Suction Strainers	6C
93-02 Supp. 1	02/18/94	Debris Plugging of Emergency Core Cooling Suction Strainers	6C
93-03	05/28/93	Resolution of Issues Related to Reactor Vessel Water Level Instrumentation in BWRs	5.2.5.2.1(12)
94-01	04/14/94	Potential Fuel Pool Draindown Caused by Inadequate Maintenance Practices at Dresden Unit 1	9.1.2 & 9.1.3
95-02	10/17/95	Unexpected Clogging of a Residual Heat Removal (RHR) Pump Strainer While Operating in Suppression Pool Cooling Mode	6C
96-02	04/11/96	Movement of Heavy Loads Over Spent Fuel, Over Fuel in the Reactor Core, or Over Safety-Related Equipment	9.1.5
96-03	05/06/96	Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling-Water Reactors	6C
2005-02	07/18/05	Emergency Preparedness and Response Actions for Security-Based Events	COL Applicant
2011-01	05/11/11	Mitigating Strategies	1D, COL Applicant

Safety Issues Index (Continued)

Title		NRC Priority	Tier 2 Subsection
105	Interfacing Systems LOCA at BWRs	High	19B.2.45 COL App.
106	Piping and Use of Highly Combustible Gases in Vital Areas	Medium	19B.2.46
118	Tendon Anchorage Failure	Resolved	19B.2.48
124	Auxiliary Feedwater System Reliability	Resolved	19B.2.51
128	Electrical Power Reliability	Resolved	19B.2.52
142	Leakage Through Electrical Isolators in Instrumentation Circuits	Medium	19B.2.53
143	Availability of Chilled Water Systems	High	19B.2.54
145	Actions to Reduce Common Cause Failures	Resolved	19B.2.55 COL App.
153	Loss of Essential Service Water in LWRs	High	19B.2.57 COL App.
155.1	More Realistic Source Term Assumptions	Resolved	19B.2.58
	Add Insert 9		
Human Factors Issues			
HF.1.1	Shift Staffing	Resolved	18.8.2
HF.4.4	Guidelines for Upgrading Other Procedures	High	18.8.1 18E.1.7
HF.5.1	Local Control Stations	High	18.8.11
HF.5.2	Review Criteria for Human Factors Aspects of Advanced Controls and Instrumentation	High	18.8.9
Issues Resolved With No New Requirements			
A-17	Systems Interaction in Nuclear Power Plants	Resolved	19B.2.59
A-29	Nuclear Power Plant Design for Reduction of Vulnerability to Industrial Sabotage	Resolved	19B.2.60 COL App.
B-5	Ductility of Two-Way Slabs and Shells and Buckling Behavior of Steel Containments	Resolved	19B.2.61
C-8	Main Steamline Leakage Control Systems	Resolved	19B.2.61.1
29	Bolting Degradation or Failure in Nuclear Power Plants	Resolved	19B.2.62
82	Beyond Design Basis Accidents in Spent-Fuel Pools	Resolved	19B.2.63
113	Dynamic Qualification Testing of Large Bore Hydraulic Snubbers	Resolved	19B.2.64

Insert 9
Safety Issues Index

Title		NRC Priority	Tier 2 Subsection
New Generic Issues			
186	Potential Risk and Consequences of Heavy Load Drops in Nuclear Power Plants	TBD	19B.2.74
189	Susceptibility of Ice Condenser and Mark III Containments to Early Failure from Hydrogen Combustion during a Severe Accident	TBD	19B.2.75
191	Assessment of Debris Accumulation on PWR Sump Performance	TBD	19B.2.76
193	BWR ECCS Suction Concerns	TBD	19B.2.77 Tier 1 Table 2.4.1 Item 4c, Table 2.4.2 Item 3g, Table 2.4.4 Item 3j
199	Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States for Existing Plants	TBD	19B.2.78 COL App. Items 2.3.1.2 and 2.3.2.22.

(Figures 12.3-56 through 12.3-73) as well as the specific area radiation channels for each building, the detector map location, the channel sensitivity range, and the local alarm areas (Tables 12.3-3 through 12.3-7).

References

19B.2.72-1 NUREG-0660, “NRC Action Plan Developed as a Result of the TMI-2 Accident”, U.S. NRC, May 1980.

19B.2.72-2 NUREG-0737, “Clarification of TMI Action Plan Requirements”, U.S. NRC, November 1980.

19B.2.73 III.D.3.3(2): Set Criteria Requiring Licensees to Evaluate Need for Additional Survey Equipment

Issue

NUREG-0660 (Reference 19B.2.73-1) is a guideline to improve nuclear power plant worker radiation protection to allow workers to take effective action to control the course and consequences of an accident, as well as to keep exposures as low as reasonably achievable (ALARA) during normal operation and accidents.

Acceptance Criteria

This issue required the NRR to set criteria requiring licensees to evaluate in their plants the need for additional survey equipment and radiation monitors in vital areas and requiring, as necessary, installation of area monitors with remote readout. The NRR evaluated the need to specify the minimum types and quantities of portable monitoring instrumentation, including very high dose rate survey instruments. Operating reactors were reviewed for conformance with Standard Review Plan (SRP) Section 12.3.4, “Area Radiation and Airborne Radioactivity Monitoring Instrumentation”. The NRR revised the SRP Sections 12.5 and 12.3.4 to incorporate additional monitor requirement criteria.

Resolution

Item III.D.3.3(2) which concerns licensees evaluate the need for additional radiation survey equipment is resolved in Subsection 12.3.4. This item also concerned the need to specify the minimum types and quantities of portable monitoring instrumentation, including very high dose rate survey instruments. As noted in Subsections 12.5.2, 19A.2.39 and 19A.3.5, COL applicants will provide the portable instruments in operating reactors that accurately measure radio-iodine concentration in plant areas under accident conditions.

References

19B.2.73-1 NUREG-0660, “NRC Action Plan Developed as a Result of the TMI-2 Accident”, U.S. NRC, May 1980.

19B.2.73-2 NUREG-0737, “Clarification of TMI Action Plan Requirements”, U.S. NRC, November 1980.

Insert 10

19B.2.74 186: Potential Risk and Consequences of Heavy Load Drops in Nuclear Power Plants

Issue

This issue was identified by NRR in April 1999 when the concern was raised that licensees operating within the regulatory guidelines of GL 85-11 may not have taken adequate measures to assess and mitigate the consequences of dropped heavy loads. Prior to the issuance of GL 85-11, GLs 80-113, 81-07, and 83-42 were issued with requirements for operating licensees following the resolution of Issue A-36. In April 1996, NRC Bulletin 96-02 was issued to alert licensees of potential high consequences that could result from a cask drop and to remind them of complying with existing regulatory guidelines on the control and handling of heavy loads.

Acceptance Criteria

The acceptance criteria for the resolution of Issue 186 is that heavy load handling systems shall be designed to provide the equipment, procedures and operator training such that no credible drop can cause a release of radioactivity, a criticality accident, an inability to cool fuel within the reactor vessel or spent-fuel pool, or prevent a safe shutdown of the reactor. Where applicable, the design shall conform to the industrial and electrical codes, the relevant requirements of General Design Criteria 2, 4, and 61 of 10 CFR 50, Appendix A (Reference 19B.2.12-4) and NUREG-0612.

Resolution

This item was resolved in the ABWR DCD Tier 2, Section 9.1.5.5, in the original design certification:

> 9.1.5.5 Safety Evaluations
>
> The cranes, hoists, and related lifting devices used for handling heavy loads either satisfy the single-failure guidelines of NUREG-0612, Subsection 5.1.6, including NUREG-0554 or evaluations are made to demonstrate compliance with the recommended guidelines of Section 5.1, including Subsections 5.1.4 and 5.1.5. The equipment handling components over the fuel pool are designed to meet the single- failure-proof criteria to satisfy NUREG-0554. Redundant safety interlocks and limit switches are provided to prevent transporting heavy loads other than spent fuel by the refueling bridge crane over any spent fuel that is stored in the spent-fuel storage pool. A transportation routing study will be made of all planned heavy load handling moves to evaluate and minimize safety risks. Safety evaluations of related light loads and refueling handling tasks in which heavy load equipment is also used are covered in Subsection 9.1.4.3. The CRD and RIP maintenance equipment on the rotating bridge below the RPV used during refueling operation will be withdrawn through the personnel equipment tunnel to outside primary containment.

Therefore, this issue is resolved for the ABWR.

References

19B.2.74-1 NUREG-0933, "A Prioritization of Generic Safety Issues (with Supplements 1-34)," U.S. NRC, June 2016.

19B.2.75 189: Susceptibility of Ice Condenser and Mark III Containments to early failure from hydrogen combustion during a severe accident.

Issue

It was discovered in the study associated with NUREG/CR-6427 that, for ice condenser containments, the early containment failure probability is dominated by non-DCH hydrogen combustion events. To deal with large quantities of hydrogen, these containments are equipped with AC-powered igniters, which are intended to control hydrogen concentrations in the containment atmosphere by initiating limited "burns" before a large quantity accumulates. During accident sequences associated with station blackouts, where the igniter systems are not available due to no electrical power, this can become a problem.

Acceptance Criteria

Because of the potential for significant hydrogen generation as a result of an accident, 10 CFR 50.44, “Standards for Combustible Gas Control System in Light-Water-Cooled Power Reactors” (Reference 19B.2.18-1), and General Design Criterion 41, “Containment Atmosphere Cleanup”, in Appendix A to 10 CFR Part 50 (Reference 19B.2.18-2), require that systems be provided to control hydrogen concentrations in the containment atmosphere following a postulated accident to ensure that containment integrity is maintained.

Paragraph (f)(2)(ix) of 10 CFR 50.34, “Contents of Applications; Technical Information” (Reference 19B.2.18-4), requires that provision be made for a hydrogen control system that can safely accommodate hydrogen generated by the equivalent of a 100% fuel-clad metal-water reaction.

An inerted containment and the provision for permanently installed hydrogen recombiners are acceptable as hydrogen control measures.

Resolution

This item was resolved in the ABWR DCD Tier 2, Section 19B.2.18A-48, in the original design certification:

The ABWR containment is considerably different from an Ice Condenser Containment. The ABWR Containment also differs from the Mark III Containment in that it is inerted to prevent hydrogen combustion. In the ABWR, there are no design basis events that result in core uncovery or core heatup sufficient to cause significant metal-water reaction. Therefore, per Regulatory Guide 1.7, the design basis metal/water reaction is that equivalent to the reaction of the active clad to a depth of 0.0058 mm. This is equivalent to 0.72% of the active clad.

Therefore, this issue is resolved for the ABWR.

References

19B.2.75-1 NUREG-0933, “A Prioritization of Generic Safety Issues (with Supplements 1-34),” U.S. NRC, June 2016.

19B.2.76 191: Assessment of Debris Accumulation on PWR Sump Performance

A study was deemed to be required to determine whether PWR ECCS sumps are adequate to ensure proper ECCS operation. Based on the existence of an action plan to address the safety concerns, the issue was considered nearly-resolved in September 1996. It was later given a HIGH priority ranking in SECY-98-166.

The staff's Technical Assessment concluded that GSI-191 was a credible concern for the population of domestic PWRs, and that detailed plant-specific evaluations were needed to determine the susceptibility of each U.S.-licensed PWR to ECCS sump blockage.

Acceptance Criteria

Not applicable. Issue does not apply to ABWRs.

Resolution

This issue is specific to PWRs. GSI 193 addresses BWR ECCS Suction Concerns

Therefore, Issue 191 is resolved for ABWR.

References

19B.2.76-1 NUREG-0933, "A Prioritization of Generic Safety Issues (with Supplements 1-34)," U.S. NRC, June 2016.

19B.2.77 193: BWR ECCS Suction Concerns

Issue

This issue addressed the possible failure of low pressure emergency core cooling systems due to unanticipated, large quantities of entrained gas in the suction piping from suppression pools in BWR Mark I containments.

The swell/exclusion zone in the BWR Mark I torus after a LOCA is considered to be limited to less than one diameter of the down-comer pipe. The ABWR Containment is not as limiting as the Mark I and therefore this condition may not be present in the ABWR.

Acceptance Criteria

Not applicable. The ABWR containment differs from the Mark I Containment and the arrangement of the horizontal vents alleviates the problem identified for the Mark I Containment Downcomers. This issue is resolved for ABWR renewal application.

Resolution

For containment suppression pool LOCA analyses, an NRC SER for two GE topical reports (NEDO-30832 and NEDO- 31695-A) accepts the elimination of suppression pool local temperature limits with the proviso that the ECCS suction strainer inlet be below the quencher outlet. If this is not the case for a specific installation, the new strainers may need to be evaluated for the potential effect of air and or steam ingestion from an SRV quencher into the strainer as this could potentially affect ECCS pump/system performance.

Tier 1, Tables 2.4.1, 2.4.2 and 2.4.4 (ITAAC) include a requirement for the respective ECCS pump suction Strainer for a verification of adequate vertical and horizontal separation between the ECCS suction strainer and the SRV quencher to prevent the potential effect of air and or steam ingestion.

Therefore, Issue 193 is resolved for ABWR with actions identified for the COL Holder.

References

19B.2.77-1 NUREG-0933, "A Prioritization of Generic Safety Issues (with Supplements 1-34)," U.S. NRC, June 2016.

19B.2.77-2 NEDC-32721P-A, "Application Methodology for the General Electric Stacked Disk ECCS Suction Strainer", Revision 2, March 2003

19B.2.77-3 NEDO-30832, "Elimination of Limit on BWR Suppression Pool Temperature For SRV Discharge with Quenchers", Revision 0, December 1984

19B.2.77-4 NEDO-31695-A, "BWR Suppression Pool Temperature Technical Specification Limits", Revision 0, May 1995

19B.2.78 199: Implications of Updated Seismic Hazard Estimates in Central and Eastern U.S for Existing Plants

Issue

Recent data and models indicate that estimates of the potential for earthquake hazards for some NPPs in the Central and Eastern United States (CEUS) may be larger than previous estimates.

Regulatory Guide 1.165, "Identification and Characterization of Seismic Sources and Determination of Safe Shutdown Earthquake Ground Motion," developed in the early 1990s, specifies a reference probability for exceedance of a safe-shutdown earthquake (SSE) ground motion (i.e., seismic hazard) at a median annual value of 1E-5. This reference probability value is based on the annual probability of exceeding the SSEs for 29 CEUS nuclear power sites and is used to establish the SSEs for future nuclear facilities. Based on preliminary results from work performed by the United States Geological Survey (USGS) in 2004, it appears that the reference probability for the 29 CEUS sites has increased to about 6 to 7E-5. The increase in the reference probability value is primarily due to recent developments in the modeling of earthquake ground motion in the CEUS. When the staff first identified this issue, no new plants had applied for a construction permit or early site permit (ESP) since Title 10 of the *Code of Federal Regulations* (10 CFR) Part 100, "Reactor Site Criteria," was revised and Regulatory Guide 1.165 was issued in 1997. When the staff began review of the ESP applications, the staff realized the impact of the revised regulation and the regulatory guide as they relate to future plants and operating reactors.

From the staff's review of the ESP applications with support from the 2004 USGS draft report, it appeared that the perception of seismic hazard for operating plants in the CEUS region had increased. Based on the evaluations of the Individual Plant Examination of External Events (IPEEE) Program, the staff had determined that seismic designs of operating plants in the CEUS provided an adequate level of protection. However, in light of the preliminary results from the USGS work of 2004 and the ESP applications, the staff also recognized that the probability of exceeding the SSE at some of the currently operating sites in the CEUS is higher than previously understood. Therefore, the staff initiated this GI to assess the impact of increased estimates of seismic hazards on selected current NPPs in the CEUS region that might be impacted by the updated seismic research, information, and models.

Acceptance Criteria

This is a site-specific issue that would be addressed in a COL application through the comparison of site characteristics with the site parameters. The applicable COL Information Items are set forth in Sections 2.3.1.2 and 2.3.2.22 of the ABWR DCD.

Resolution

To the extent that any COLA may reference the ABWR certified design, the applicant would determine how GSI-199 is addressed for the proposed site.

Therefore, Issue 199 is resolved for ABWR with actions identified for the COL Application.

References

19B.2.78-1 NUREG-0933, “A Prioritization of Generic Safety Issues (with Supplements 1-34),” U.S. NRC, June 2016.