19B Resolution of Applicable Unresolved Safety Issues and Generic Safety Issues

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A-8 Mark I Containment Pool Dynamic Loads—Long-Term Program	Resolved	19B.2.4
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A-10 BWR Feedwater Nozzle Cracking	Resolved	19B.2.6
A-13 Snubber Operability Assurance	Resolved	19B.2.7
A-24 Qualification of Class 1E Safety-Related Equipment	Resolved	19B.2.8
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A-47 Safety Implications of Control Systems	Resolved	19B.2.17 COL App.
A-48 Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment	Resolved	19B.2.18
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105 Interfacing Systems LOCA at BWRs	Resolved	19B.2.45†
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HF.1.1 Shift Staffing	Resolved	18.8.2
HF.4.4 Guidelines for Upgrading Other Procedures	Resolved	18.8.1 18E.1.7
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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.
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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
120 On-Line Testability of Protection Systems	Resolved	19B.2.49†
121 Hydrogen Control for Large, Dry PWR Containments	Resolved	19B.2.50
151 Reliability of Anticipated Transient without Scram Recirculation Pump Trip in BWRs	Resolved	19B.2.56†
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I.A.1.1 Shift Technical Advisor	Resolved	COL App.
I.A.1.2 Shift Supervisor Administrative Duties	Resolved	COL App.
I.A.1 3 Shift Manning	Resolved	Technical Specifications
I.A.1.4 Long-Term Upgrading	Resolved	Appendix 18E
I.A.2.1(1) Qualifications-Experience	Resolved	COL App.
I.A.2.1(2) Training	Resolved	COL App.
I.A.2.1(3) Facility Certification of Competence and Fitness of Applicants for Operator and Senior Operator Licenses	Resolved	COL App.
I.A.2.3 Administration of Training Programs	Resolved	COL App.
I.A.2.6(1) Revise Regulatory Guide 1.8	Resolved	Not Applicable*
I.A.3.1 Revise Scope of Criteria for Licensing Examinations	Resolved	Not Applicable*
I.A.4.1(2) Interim Changes in Training Simulators	Resolved	COL App.
I.A.4.2(1) Research on Training Simulators	Resolved	19A.3.1
I.A.4.2(2) Upgrade Training Simulator Standards	Resolved	19A.3.1
I.A.4.2(3) Regulatory Guide on Training Simulators	Resolved	19A.3.1
I.A.4.2(4) Review Simulators for Conformance to Criteria	Resolved	19A.3.1
I.C.1(1) Small-Break LOCAs	Resolved	COL App.

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Title	NRC Priority	Tier 2 Subsection
I.C.1(2) Inadequate Core Cooling	Resolved	COL App.
I.C.1(3) Transients and Accidents	Resolved	1A.2.1
I.C.2 Shift and Relief Turnover Procedures	Resolved	COL App.
I.C.3 Shift Supervisor Responsibilities	Resolved	COL App.
I.C.4 Control Room Access	Resolved	COL App.
I.C.5 Procedures for Feedback of Operating Experience to Plant Staff	Resolved	19A.3.6
I.C.6 Procedures for Verification of Correct Performance of Operating Activities	Resolved	COL App.
I.C.7 NSSS Vendor Review of Procedures	Resolved	Not Applicable to ABWR
I.C.8 Pilot-Monitoring of Selected Emergency Procedures for Near-Term Operating License Applicants	Resolved	Not Applicable (historical)
I.D.1 Control Room Design Reviews	Resolved	1A.2.2
I.D.2 Plant Safety Parameter Display Console	Resolved	1A.2.3
I.D.3 Safety System Status Monitoring	Resolved	19A.2.17
I.D.5(2) Plant Status and Post-Accident Monitoring	Resolved	19B.2.65
I.D.5(3) On-Line Reactor Surveillance System	Resolved	19B.2.66
I.F.2(2) Include QA Personnel in Review and Approval of Plant Procedures	Resolved	19A.2.43
I.F.2(3) Include QA Personnel in All Design, Construction, Installation, Testing, and Operation Activities	Resolved	19A.2.43
I.F.2(6) Increase the Size of Licensees' QA Staff	Resolved	19A.2.43
I.F.2(9) Clarify Organizational Reporting Levels for the QA Organization	Resolved	19A.2.43
I.G.1 Training Requirements	Resolved	1A.2.4
I.G.2 Scope of Test Program	Resolved	19B.2.67
II.B.1 Reactor Coolant System Vents	Resolved	1A.2.5
II.B.2 Plant Shielding to Provide Access to Vital Areas and Protect Safety Equipment for Post-Accident Operation	Resolved	1A.2.6
II.B.3 Post-Accident Sampling	Resolved	1A.2.7

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II.B.4 Training for Mitigating Core Damage	Resolved	COL App.
II.B.8 Rulemaking Proceeding on Degraded Core Accidents	Resolved	19A.2.1 19A.2.2 19A.2.43 19A.2.45
II.D.1 Testing Requirements	Resolved	1A.2.9
II.D.3 Relief and Safety Valve Position Indication	Resolved	1A.2.10
II.E.4.1 Dedicated Penetrations	Resolved	1A.2.13
II.E.4.2 Isolation Dependability	Resolved	1A.2.14
II.E.4.4 Purging	Resolved	19A.2.27
II.E.6.1 Test Adequacy Study	Resolved	19B.2.68
II.F.1 Additional Accident Monitoring Instrumentation	Resolved	1A.2.15
II.F.2 Identification of and Recovery from Conditions Leading to Inadequate Core Cooling	Resolved	1A.2.16
II.F.3 Instruments for Monitoring Accident Conditions	Resolved	1A.2.17
II.J.4.1 Revise Deficiency Reporting Requirements	Resolved	Not Applicable*
II.K.1(5) Safety-Related Valve Position Description	Resolved	1A.2.18 18.8.7
II.K.1(10) Review and Modify Procedures for Removing Safety-Related Systems from Service	Resolved	1A.3.2
II.K.1(13) Propose Technical Specifications Changes Reflecting Implementation of All Bulletin Items	Resolved	19B.2.69
II.K.1(22) Describe Automatic and Manual Actions for Proper Functioning of Auxiliary Heat Removal Systems When FW System Not Operable	Resolved	1A.2.20
II.K.1(23) Describe Uses and Types of RV Level Indication for Automatic and Manual Initiation Safety Systems	Resolved	1A.2.21
II.K.3(3) Report Safety and Relief Valve Failures Promptly and Challenges Annually	Resolved	1A.3.4
II.K.3(11) Control Use of PORV Supplied by Control Components, Inc. Until Further Review Complete	Resolved	19B.2.70
II.K.3(13) Separation of HPCI and RCIC System Initiation Levels	Resolved	1A.2.22

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Title	NRC Priority	Tier 2 Subsection
II.K.3(15) Modify Break Detection Logic to Prevent Spurious Isolation of HPCI and RCIC Systems	Resolved	1A.2.23
II.K.3(16) Reduction of Challenges and Failures of Relief Valves-Feasibility Study and System Modification	Resolved	1A.2.24
II.K.3(17) Report and Outage of ECC Systems—Licensee Report and Technical Specification Changes	Resolved	1A.2.25
II.K.3(18) Modification of ADS Logic—Feasibility Study and Modification for Increased Diversity for Some Event Sequences	Resolved	1A.2.26
II.K.3(21) Restart of Core Spray and LPCI Systems on Low Level-Design and Modification	Resolved	1A.2.27
II.K.3(22) Automatic Switchover of RCIC System Suction—Verify Procedures and Modify Design	Resolved	1A.2.28
II.K.3(24) Confirm Adequacy of Space Cooling for HPCI and RCIC Systems	Resolved	1A.2.29
II.K.3.(25) Effect of Loss of AC Power on Pump Seals	Resolved	1A.2.30
II.K.3(27) Provide Common Reference Level for Vessel Level Instrumentation	Resolved	1A.2.21
II.K.3(28) Study and Verify Qualification of Accumulators on ADS Valves	Resolved	1A.2.31
II.K.3(30) Revised Small-Break LOCA Methods to Show Compliance with 10 CFR 50, Appendix K	Resolved	1A.2.32
II.K.3(31) Plant-Specific Calculations to Show Compliance with 10 CFR 50.46	Resolved	1A.2.33
II.K.3(44) Evaluation of Anticipated Transients with Single Failure to Verify No Significant Fuel Failure	Resolved	1A.2.33.1
II.K.3(45) Evaluate Depressurization with Other Than Full ADS	Resolved	19A.2.11
II.K.3(46) Response to List of Concerns from ACRS Consultant	Resolved	1A.2.33.3
III.A.1.1(1) Implement Action Plan Requirements for Promptly Improving Licensee Emergency Preparedness	Resolved	Not Applicable*
III.A.1.2(1) Technical Support Center	Resolved	19A.3.4
III.A.1.2(2) On-Site Operational Support Center	Resolved	19A.3.4
III.A.1.2(3) Near-Site Emergency Operations Facility	Resolved	19A.3.4

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Title	NRC Priority	Tier 2 Subsection
III.A.2.1(1) Publish Proposed Amendments to the Rules	Resolved	Not Applicable*
III.A.2.1(2) Conduct Public Regional Meetings	Resolved	Not Applicable†
III.A.2.1(3) Prepare Final Commission Paper Recommending Adoption of Rules	Resolved	Not Applicable†
III.A.2.1(4) Revise Inspection Program to Cover Upgraded Requirements	Resolved	Not Applicable*
III.A.2.2 Development of Guidance and Criteria	Resolved	Not Applicable*
III.A.3.3(1) Install Direct Dedicated Telephone Lines	Resolved	COL App.
III.A.3.3(2) Obtain Dedicated, Short-Range Radio Communication Systems	Resolved	Not Applicable*
III.D.1.1(1) Review Information Submitted by Licensee Pertaining to Reducing Leakage from Operating Systems	Resolved	1A.2.34
III.D.3.3(1) Issue Letter Requiring Improved Radiation Sampling Instrumentation	Resolved	19A.2.39
III.D.3.3(2) Set Criteria Requiring Licensees to Evaluate Need for Additional Survey Equipment	Resolved	19A.2.39
III.D.3.3(3) Issue a Rule Change Providing Acceptable Methods for Calibration of Radiation-Monitoring Instruments	Resolved	19A.3.5
III.D.3.3(4) Issue a Regulatory Guide	Resolved	19A.3.5
III.D.3.4 Control Room Habitability	Resolved	1A.2.36

* NRC action.

† Not listed in Applicability of NUREG-0933 Issues to Operating and Future Reactor Plants.

19B.1 Introduction

19B.1.1 Purpose

The ABWR has proposed technical resolutions of those Unresolved Safety Issues (USI) and medium and high priority Generic Safety Issues (GSI) which are identified in the version of NUREG-0933 through Supplement 15 (Reference 19B.1.1-1) and which are technically relevant to the ABWR design in accordance with 10 CFR 52.47(a)(21). These same USI and GSI not categorized as "Resolved" for the initial ABWR certification were updated as applicable based on NUREG-0933 through Supplement 33 (Reference 19B.1.1-4). NUREG-0933 and associated correspondence (References 19B.1.1-2 and 19B.1.1-3) were reviewed and evaluated for the ABWR. The TMI issues satisfying Section II of NUREG-0800, Standard Review Plan, are addressed in Appendix 1A; and those satisfying 10 CFR 50.34(f) are addressed in Appendix 19A. The remaining issues are addressed in Subsection 19B.2.

The following guidelines were used in the review of NUREG-0933 to eliminate potentially non-relevant issues to the ABWR design:

(1) Priority rating of low, dropped, or not yet prioritized

(2) Operational, environmental, licensing, or other NRC impact with no plant design content

(3) No design content applicable to the ABWR design except for NRC identified issues

(4) Resolved with no new requirements except for ACRS and NRC selected issues

In addition, the NRC staff assisted in identifying relevant and current issues and resolutions. The group of issues remaining are identified in the Safety Issues Index and are evaluated in the referenced subsection. Where COL applicant is indicated in the Tier 2 subsection column, the issue is included in Subsection 19B.3 for the COL applicant to address and evaluate. These COL issues pertain to operating personnel including staffing, training, qualification and licensing; operating procedures including post accident operation, severe accident safety reviews, improved emergency preparedness and radiation effects and deficiency reporting; and assisting in the development of regulatory documentation. The COL applicant is required to provide the resolution for each issue as described below.

The documentation of the issue evaluation is comprised of four sections:

- ISSUE,
- ACCEPTANCE CRITERIA,
- RESOLUTION, and
- REFERENCES.

The ISSUES statement is a brief summary description of the issue. The ACCEPTANCE CRITERIA are taken from NUREG-0933 and GIMCS (Reference 19B.1.1-2) resolution references and where there is no formal NRC resolution, accepted industry codes and standards and good engineering practices. The RESOLUTION contains the technical resolution of the issue for the ABWR Standard Plant design. The REFERENCES identifies documentation other than Tier 2.

References

19B.1.1-1 NUREG-0933, "A Prioritization of Generic Safety Issues (with Supplements 1-15)", U.S. NRC, April 1993.

19B.1.1-2 "Generic Issue Management Control System" - Fourth Quarter FY-93, Update, Memorandum for James M. Taylor from E. S. Beckjord dated March 30, 1993.

19B.1.1-3 "Advanced Light Water Reactor Utility Requirements Document", Volume II, Electric Power Institute, Advanced LWR Program.

19B.1.1-4 NUREG-0933, "Resolution of Generic Safety Issues (with Supplements 1-33)", U.S. NRC, August 2010.

19B.2 Safety Issues

19B.2.1 Issue, Acceptance Criteria and ABWR Resolution

19B.2.2 A-1: Water Hammer

Issue

Unresolved Safety Issue (USI) A-1 in NUREG-0933 (Reference 19B.2.2-1) addresses identifying the probable causes of water hammer and minimizing the susceptibility of fluid systems and components to water hammer by correcting design and operational deficiencies.

Water hammer is defined as a rapid deviation in pressure caused by a change in the velocity of a fluid in a closed volume. There are various types of water hammer, including steam condensation-induced water hammer, which occurs in the secondary side of a PWR steam generator at the connection to the feedwater line. This type of water hammer involves steam generator feedrings and piping. Water hammer has been observed in many fluid systems including residual heat removal, containment spray, service water, feedwater systems, and main steam lines. In addition to condensation-induced water hammer, other forms of initiating events which cause water hammer can occur, such as steam driven slugs of water, pump startup with partially empty lines, and rapid valve cycling.

Regardless of the initiating event, water hammer and the resulting fluid accelerations can cause damage to the affected fluid system. The level of severity of damage depends upon the event, and can range from minor damage such as overstressed pipe hangers to major damage to restraints, piping and components.

According to NUREG-0927 (Reference 19B.2.2-2), water hammer can be induced by operator/maintenance actions and by design inadequacies. Experience has shown that water hammer events reported on LERs are about equally divided between operator or maintenance actions and design deficiencies. The NRC implemented Standard Review Plan (SRP) changes relative to the design, operation, and maintenance of new plants to minimize the probability and effects of water hammer, and issued a Branch Technical Position (BTP) for pre-operational tests.

Acceptance Criteria

Reference 19B.2.2-1 concluded that USI A-1 was resolved by the publication of SRP sections:

SRP	Revision
3.9.3	1
3.9.4	2
5.4.6	3
5.4.7	3
6.3.1	2
9.2.1	3
9.2.2	2
10.3	3
10.4.7	3

Compliance with these SRPs becomes the acceptance criteria for resolving this issue.

Resolution

The ABWR design complies with the above listed SRPs and therefore the water hammer issue is resolved. Of all the ABWR systems, the systems discussed below are the only systems considered as having a potential for water hammer.

Potential water hammer conditions are prevented by implementation of the following analyses, design features, and pre-operational tests. Tier 2 section references are given.

(1) Water hammer evaluation is required for specific piping regions as follows:

(a) Condensate and Feedwater System. The feedwater lines are demonstrated to have low probability of failure from water hammer effects. Subsections 10.4.7.3, 3E.6.2.2, 3E.6.2.7, 20.3.10 RAI-10 Question-Response 430.89.

(b) Main steam lines are analyzed for dynamic loadings due to fast closing of the turbine stop valves. Subsection 5.4.9.1(4).

(c) All components of the main steam supply system are designed to accommodate the loads and stresses resulting from steam hammer. Subsection 10.3.3.

(2) Not Used.

(3) Applicable systems are filled with water, and kept filled with water, which prevents water hammer when pumps are started from a standby condition. Systems described in Tier 2 are as follows:

(a) Operating procedures will be developed so that all divisions of the Reactor Service Water (RSW) System are maintained full of water to prevent water hammer [Subsection 9.2.15.1.1(5)]

(b) Operating procedures will be developed so that all components of the Turbine Service Water (TSW) System are maintained full to prevent water hammer [Subsection 9.2.16.2.2(5)]

(c) Residual Heat Removal (RHR) [Subsections 5.4.1.1.4, 6.3.2.2.5, and 14.2.12.1.8(3)(m)]

(d) High Pressure Core Flooder (HPCF) [Subsections 6.3.2.2.5, and 14.2.12.1.10(3)(n)]

(e) Reactor Core Isolation Cooling (RCIC) [Subsections 5.4.6.2.5.1, 6.3.2.2.5, and 14.2.12.1.9(3)(n)]

(f) HVAC Emergency Cooling Water System [Subsection 9.2.13.1.2(6)]

(4) Condensation Induced Water Hammer (CIWH) for the ECCS systems (RHR, HPCF, and RCIC) was evaluated for the ABWR (Reference 19B.2.2-5). The conclusion was that the ECCS injection piping configuration was not susceptible to CIWH.

(a) For the RHR System low pressure flooder (LPFL) mode, the water in the sloped, but nearly horizontal, injection line flashes to steam during reactor depressurization. An analysis was performed that indicated about 80% of the water remained in the pipe after depressurization. Therefore, slow injection of cold water by the LPFL injection valve into the horizontal LPFL pipe partially filled with saturated water will not cause CIWH.

(b) For the HPCF System, in the event of a LOCA, the high pressure flooder spargers located inside the RPV shroud are immersed in a two-phase mixture. During the flashing period prior to HPCF initiation, the RPV is depressurizing and water in the piping can flash. A steam bubble can form at the piping's high point. For the HPCF high pressure system, injection begins within a few seconds, and the entrance of subcooled water could cause decompression inside the pipe. Any water slug accelerated from the reactor side towards the upstream piping will flash into a two-phase mixture because the water is in a saturated condition. A slug of two-phase mixture, which is highly compressible, colliding with another surface has been analyzed and found to produce a pressure pulse of the order of 6.8947E+04 Pa to 1.3789E+05 Pa. This analysis was done earlier for typical BWR-5 and BWR-6 piping using TRACB01 computer code. 1.3789E+5 Pa pressure pulses are not considered significant, and it is concluded that CIWH is not a problem for the HPCF injection piping.

(c) For the RCIC System during system initiation, the water level in the reactor is a Level 2 or higher, which is higher than the feedwater nozzle height. The fluid condition at the feedwater sparger is water when RCIC water is pumped into the vessel. Therefore, CIWH will not occur at the time of RCIC makeup water injection into the reactor vessel.

(5) Pre-operational tests are specified for the purpose of verifying the piping keep-fill methods are operational. Filled pipelines preclude water hammer associated with pump startup.

(a) RHR [Subsection 14.2.12.1.8(3)(m)]

(b) HPCF [Subsection 14.2.12.1.10(3)(n)]

(c) RCIC [Subsection 14.2.12.1.9(3)(n)]

References

19B.2.2-1 NUREG-0933, "A Prioritization of Generic Safety Issues (with Supplements 1-15)", U.S. NRC, April 1993.

19B.2.2-2 NUREG-0927, Revision 1, "Evaluation of Water Hammer Occurrence in Nuclear Power Plants", U.S. NRC, March 1984.

19B.2.2-3 10 CFR 50 Appendix A, "General Design Criteria for Nuclear Power Plants", Code of Federal Regulations, Office of the Federal Register, National Archives and Records Administration.

19B.2.2-4 NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants—LWR Edition", U.S. NRC.

19B.2.2-5 Jack Fox, GE, to Chet Poslusny, NRC, "Submittal Supporting Accelerated ABWR Review Schedule – Water Hammer Evaluation", May 11, 1993.

19B.2.3 A-7: Mark I Long-Term Program

Issue

During testing for an advanced BWR containment system design (MARK III), suppression pool hydrodynamic loads were identified which had not been considered in the original design of the MARK I containment system. To address this issue, a MARK I Owners Group was formed and the assessment was divided into a short-term and long-term program. The results of the NRC staff's review of the MARK I Containment Short-Term Program are described in NUREG-0408 (Reference 19B.2.3-6). The long-term program (LTP) was conducted to provide a generic basis to define suppression pool hydrodynamic loads and the related structural acceptance criteria, such that a comprehensive reassessment of each MARK I containment system would be performed. A series of experimental and analytical programs were conducted by the MARK I Owners Group to provide the necessary bases for the generic load definition and structural assessment techniques. The generic methods proposed by the MARK I Owners Group, as modified by the NRC staff's requirements, will be used to perform plant-unique analyses, which will identify the plant modifications, if any, that will be needed to restore the originally intended margin of safety in the MARK I containment designs. This item was originally identified in NUREG-0371 (Reference 19B.2.3-5) and was later determined to be a Unresolved Safety Issue (USI).

Acceptance Criteria

The objectives of the LTP were to establish design basis (conservative) loads that are appropriate for the anticipated life of each Mark I boiling water reactor (BWR) facility (40 years) and to restore the originally intended design safety margins for each Mark I containment system. The principal thrust of the LTP has been the development of generic methods for the definition of suppression pool hydrodynamic loadings and the associated structural assessment techniques for the Mark I configuration.

Resolution

On the basis of the review of the experimental and analytical programs conducted by the Mark I Owners Group, the NRC staff concluded that, with one exception, the proposed suppression pool hydrodynamic load definition procedures, as modified by the NRC Acceptance Criteria in Appendix A of Reference 19B.2.3-1, will provide a conservative estimate of these loading conditions. The exception is the lack of an acceptable specification for the downcomer condensation oscillation loads. In addition, the staff requested confirmatory programs to justify the adequacy of the loading specifications in the following three areas:

(1) adequacy of the data base for specifying torus wall pressures during condensation oscillations,

(2) possibility of asymmetric torus loading during condensation oscillations, and

(3) effect of fluid compressibility in the vent system on pool-swell loads.

These programs were documented in Reference 19B.2.3-3. This report supplements the Mark I SER (NUREG-0661) by addressing the outstanding issues relating to the Mark I containment LTP, namely the downcomer condensation oscillation load definition and the confirmatory analyses and test programs that are intended to justify the adequacy of the load specifications.

The Mark I torus pool and vent configuration is not similar to the ABWR annular pool and vent design. The Mark I pool loads are not directly applicable to the ABWR because of these configuration differences and the results thus obtained cannot be used directly in the ABWR design. Nevertheless, since the line clearing phenomena for single SRV discharge conditions are the same, the results obtained from the Mark I Owner's Group program were used as a database for definition of the SRV loads that are applicable to the ABWR and utilized in resolution of Issue A-39 (Subsection 19B.2.13).

References

19B.2.3-1 NUREG-0661, "Safety Evaluation Report, Mark I Long Term Program, Resolution of Generic Technical Activity A-7", U.S. NRC, July 1980.

19B.2.3-2 NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants—LWR Edition", U.S. NRC.

19B.2.3-3 NUREG-0661, Supplement 1, "Safety Evaluation Report for the MARK I Containment Long-Term Program", U.S. NRC, August 1982.

19B.2.3-4 NUREG-0933, *"*A Prioritization of Generic Safety Issues (with Supplements 1-15)", U.S. NRC, April 1993.

19B.2.3-5 NUREG-0371, "Task Action Plans for Generic Activities Category A," U.S. NRC, 1978.

19B.2.3-6 NUREG-0408, *"Mark I Containment Short-Term Program",* U.S. NRC, 1977.

19B.2.4 A-8: Mark II Containment Pool Dynamic Loads Long-Term Program

Issue

As a result of the GE testing program for the MARK III pressure-suppression containment program, new containment loads associated with a postulated LOCA were identified in 1975 which had not been explicitly included in the original design of MARK I and MARK II containments. These loads result from the dynamic effects of drywell air and steam being rapidly forced into the suppression pool during a postulated LOCA event. Other pool dynamic loads previously unaccounted for result from the actuation of safety/relief valves (SRVs) in the MARK II containment. The review and evaluation of the MARK I loads were addressed in USI A-7 and SRV loads for all suppression-type containments were addressed in USI A-39 (Reference 19B.2.4-3). This item was originally identified in NUREG-0371 (Reference 19B.2.4-4) and was later determined to be a USI.

Acceptance Criteria

The NRC established an acceptance criteria for Mark II LOCA-Related Pool Dynamic Loads addressing pool swell loads, condensation oscillation loads, and chugging loads (Reference 19B.2.4-1, Appendix A, and Reference 19B.2.4-2). The original design of the Mark II containment system considered only those loads normally associated with design-basis accidents. These included pressure and temperature loads associated with a LOCA, seismic loads, dead loads, jet impingement loads, hydrostatic loads due to water in the suppression chamber, overload pressure test loads, and construction loads. However, since the establishment of the original design criteria, additional loading conditions have been identified that must be considered for the pressure-suppression containment-system design.

Resolution

The ABWR pool swell response calculations to quantify pool swell loads were based on a simplified, one-dimensional analytical model which was reviewed and approved by the NRC staff (Reference 19B.2.4-5). Since the ABWR vent design system utilizes horizontal vents (like Mark III containments) rather than vertical, additional studies were performed to assure the applicability of the Mark II model to the ABWR.

This issue is resolved for the ABWR.

References

19B.2.4-1 NUREG-0808, "MARK II Containment Program Evaluation and Acceptance Criteria", U.S. NRC, August 1981.

19B.2.4-2 NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants—LWR Edition", U.S. NRC.

19B.2.4-3 NUREG-0933, "A Prioritization of Generic Safety Issues (with Supplements 1-15)", U.S. NRC, April 1993.

19B.2.4-4 NUREG-0371, "Task Action Plans for Generic Activities Category A", U.S. NRC, 1978.

19B.2.4-5 NEDE-21544-P, "Mark II Pressure Suppression Containment Systems: Analytical Model of the Pool Swell Phenomena", U.S. NRC, December 1976.

19B.2.5 A-9: Anticipated Transients Without Scram, (ATWS)

Issue

This issue, A-9 (Reference 19B.2.5-1), addresses the concern that the reactor can attain safe shutdown after incurring an anticipated transient (such as a loss of feedwater, loss of condenser vacuum, or loss of offsite power) with a failure of the reactor protection system to shutdown the reactor. The technical report on ATWS (WASH-1270) (Reference 19B.2.5-2) discussed the probability of an ATWS event as well as an appropriate safety objective for the event. In 1975 the staff published a status report on each vendor analysis which included guidelines on analysis models and ATWS safety objectives. This issue was resolved by the NRC with the publication of a final rule, 10 CFR 50.62 (Reference 19B.2.5-3).

Acceptance Criteria

The acceptance criteria for the resolution of this issue is that the reactor must be capable of reaching a safe shutdown condition as identified in 10 CFR 50.62 after incurring an anticipated transient and a failure to scram. Specifically, 10 CFR 50.62 requires the BWR to have automatic recirculation pump(s) trip, an alternate rod insertion system and an automatic standby liquid control system.

Resolution

For ATWS prevention/mitigation for the ABWR, the following are provided:

- An ARI system diverse and independent of the reactor protection system,
- Electric insertion of the fine motion control rod drives which is also diverse and independent of the reactor protection system,
- Automatic recirculation pump trip, and
- Automatic initiation of the standby liquid control system.

These features are described in Section 15.8 and fulfill the requirements of 10 CFR 50.62 to resolve this issue for the ABWR, and the details are discussed in Reference 19B.2.5-4.

References

19B.2.5-1 NUREG-0933, "A Prioritization of Generic Safety Issues (with Supplements 1-15)", U.S. NRC, April 1993.

19B.2.5-2 WASH-1270, "Anticipated Transients Without Scram for Water-Cooled Reactors", U.S. NRC, September 1973.

19B.2.5-3 10 CFR 50.62, "Requirements for Reduction of Risk from Anticipated Transients Without Scram (ATWS) Events for Light-Water-Cooled Nuclear Power Plants."

19B.2.5-4 NEDE-31096-A, GE Licensing Topical Report, "ATWS-Response to NRC ATWS Rule, 10CFR50.62", February, 1987.

19B.2.6 A- 10: BWR Feedwater Nozzle Cracking

Issue

Inspections of operating BWRs conducted up to April 1978 revealed cracks in the feedwater nozzles of 20 reactor vessels. Most of these BWRs contained 4 nozzles with diameters ranging from 250A to 300A (10 inches to 12 inches). Although most cracks range from 12.7 to 19.05

mm (0.5 inch to 0.75 inch) in depth (including cladding), one crack penetrated the cladding into the base metal for a total depth of approximately 38.1 mm (1.5 inch).

It was determined that cracking was due to high cycle fatigue caused by fluctuations in water temperature within the vessel in the nozzle region. These fluctuations occurred during periods of low feedwater temperature when flow is unsteady and intermittent. Once initiated, the cracks enlarge from high pressure and thermal cycling associated with startups and shutdowns. This item was originally identified in NUREG-0371 and was later determined to be an unresolved safety issue (USI) (References 19B.2.6-1 and 19B.2.6-2)

Acceptance Criteria

The acceptance criteria are based on developing a design that provides protection to the feedwater nozzle from the water temperature fluctuations. The feedwater nozzles experience thermal stress because the incoming feedwater is colder than that in the vessel. It is much colder during startups before the feedwater heaters are in service and during shutdown after the heaters have been taken out of service. Turbulent mixing of the hot water returning from the steam separators and dryers and the incoming cold feedwater causes thermal stress cycling in the nozzle bore unless it is thoroughly protected by the sparger thermal sleeve.

In previous designs bypass leakage past the junction of the thermal sleeve and nozzle safe end has been the primary source of cold water impinging on the nozzle bore. A secondary source is the layer of water that sheds off after being cooled by contact with the outer surface of the sleeve.

Resolution

The welded double sleeve design gives a low fatigue usage factor in the nozzle bore and at the inner nozzle corner. The design protects the nozzle from fluctuating temperatures and, therefore, the issue of high cycle fatigue in the feedwater nozzle has been resolved for the ABWR.

The ABWR utilizes a double feedwater nozzle thermal sleeve as can be seen on Figure 19B-1. An inner thermal sleeve leading the cooler feedwater to the feedwater sparger is welded to the nozzle safe end. The welded thermal sleeve design was adopted to assure that there is no leakage of cold feedwater between the thermal sleeve and the safe end. A secondary thermal sleeve is placed concentrically in the annulus between the inner thermal sleeve and the nozzle bore to prevent cold water that may be shedding from the outside surface of the inner sleeve impinge on the nozzle bore and the inside nozzle corner.

The material of the nozzle forging is SA-508, Class 3 low alloy steel and that of the safe end is SA-508, Class 1 carbon steel. The carbon steel safe end is welded to the nozzle forging with a carbon steel weld. The nozzle itself has no cladding.

Welded thermal sleeves have been successfully used in at least three domestic reactors and in BWR/5s in Japan since 1977. The welded double thermal sleeve with no cladding inside the nozzle is considered an improvement of the welded single sleeve design in that the outer thermal sleeve provides additional protection against high cycle fatigue in the nozzle bore and the inside nozzle corner. The double thermal sleeve as applied to the ABWR has not been used in earlier plants although Monticello and Tsuruga (Japan) are using similar designs.

The ABWR feedwater nozzle and thermal sleeve design does not correspond to any design mentioned in Table 2 of NUREG-0619. The closest design is considered to be "Welded, clad removed (spargers have top mounted elbows)". Hence, the proposed program for ISI of the ABWR feedwater nozzles and spargers is based on this design. Based upon programs approved by the NRC allowing relief from periodic PT inspections, the following program is proposed:

UT examination from the external surface of the nozzle safe ends, nozzle bores and nozzle blend radius every second outage. If indications are found in the safe ends, evaluate per Section XI of the ASME Code. If recordable indications are interpreted as cracks in any nozzle, proceed with repair as outlined in NUREG-0619, Paragraph 4.3.2.3.

Visual inspection of flow holes and welds in sparger arms and sparger tees every fourth outage.

Visual inspection of accessible areas of the nozzles from the ID surface on the same schedule as core internal components.

It is believed that UT examination of the nozzle bore using advanced techniques give better results than PT inspection of accessible areas. This method has successfully been tried out on several domestic reactors. Depending upon actual operating experience, it may be possible to extend the period between UT examinations.

Reference

19B.2.6-1 NUREG-0619, "BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking", U.S. NRC, November 1980.

19B.2.6-2 NUREG-0371, "Task Action Plans for Generic Activities (Category A)", U.S. NRC, November 1978.

19B.2.7 A-13: Snubber Operability Assurance

Issue

Generic Safety Issue (GSI) A-13 in NUREG-0933 (Reference 19B.2.7-1), addresses snubber selection and operability for safety related systems and components by identifying the need for:

(1) A consistent means of determining snubber operability through standardized functional testing.

(2) A set of criteria for selection and specification.

(3) Preservice and inservice inspection programs.

Snubbers are utilized primarily as seismic and pipe whip restraints at operating plants. Their safety function is to operate as rigid supports for restraining the motion of systems or components under dynamic load conditions such as earthquakes and severe hydraulic transients, e.g., pipe breaks.

According to NUREG-0933, a substantial number of Licensee Event Reports (LERs), concerning snubber operability, were issued by utilities. A review of these LERs showed that a variety of methods were employed to determine the operability of the snubbers and that different types of snubbers were used for systems with similar configurations.

Acceptance Criteria

The acceptance criteria for the resolution of GSI A-13 is that the design, specification, installation, and in-service operability of snubbers must meet the intent of the guidance given in SRP Section 3.9.3 (Reference 19B.2.7-2).

Specifically, during the design of safety systems or components for which snubbers are to be used, sufficient consideration should be given as to their unique application, i.e., their response to normal, upset, and faulted conditions and the effect of these responses on the associated system and/or component.

Resolution

For the ABWR design, snubbers are minimized by using design optimization procedures. However, where required, snubber supports are used as shock arrestors for safety-related systems and components. Snubbers are used as structural supports during a dynamic event such as earthquake or pipe break, but during normal operation act as passive devices which accommodate normal expansions and contractions without resistance.

Assurance of snubber operability for the ABWR design is provided by incorporating analytical, design, installation, in-service, and verification criteria to meet the intent of the draft Regulatory Guide (Reference 19B.2.7-3) as described in Subsection 3.9.3.4.1(3). The elements of snubber operability assurance include:

(1) Consideration of load cycles and travel that each snubber will experience during normal plant operating conditions.

(2) Verification that the thermal growth rates of the system do not exceed the required lock-up velocity of the snubber.

(3) Appropriate characterization of snubber mechanical properties in the structural analysis of the snubber-supported system.

(4) For engineered, large bore snubbers, issuance of a design specification to the snubber supplier, describing the required structural and mechanical performance of the snubber with respect to: activation level, release rate, spring rate, dead band, and drag

as specified in the draft Regulatory Guide SC-708-4 (Reference 19B.2.7-3). Subsequent verification that the specified design and fabrication requirements were met.

In summary, during the design of safety-related systems or components for which snubbers are to be used, sufficient consideration is given as to their unique application, (i.e., their response to normal, upset and faulted conditions and the effect of these responses on the associated system and/or component). Thus the design, specification, installation, and in-service operability of snubbers meets the intent of SRP Section 3.9.3 and this issue is resolved for the ABWR design.

References

19B.2.7-1 NUREG-0933, "A Prioritization of Generic Safety Issues (with Supplements 1-15)", U.S. NRC, April 1993.

19B.2.7-2 NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants—LWR Edition", U.S.NRC.

19B.2.7-3 DRAFT Regulatory Guide (SC-708-4), February 1981.

19B.2.8 A-24: Qualification of Class 1E Safety Related Equipment

Issue

Safety Issue A-24 in NUREG-0933 (Reference 19B.2.8-1) addresses the adequacy of environmental qualification methods and acceptance criteria for Class 1E electrical equipment.

The Nuclear Regulatory Commission (NRC) initially required license applicants to qualify all safety-related equipment to IEEE Std 323-1974 (Reference 19B.2.8-2). Some of the industry qualification methods and concepts proposed in accordance with this standard, such as testing margins, aging effects, and the simulation of worst case environments, were not resolved to the satisfaction of the NRC. It was therefore decided that a generic approach should be developed under A-24 to expedite the review and assessment of equipment qualification methods used by vendors.

All major Nuclear Steam Supply Systems (NSSS) vendors and architect engineers submitted topical reports on their methods of environmental qualification which were reviewed by the NRC and the results documented in NUREG-0588 (Reference 19B.2.8-3). In a subsequent rulemaking, 10 CFR 50.49 (Reference 19B.2.8-4) established the requirement for an environmental qualification program for Class 1E electrical equipment together with rules for its content. References 19B.2.8-2 and 19B.2.8-3 comprise the bases for the rules. Regulatory Guide 1.89 was then revised (Reference 19B.2.8-5) to described an acceptable method for complying with 10 CFR 50.49.

Dynamic and seismic qualification of Class 1E electrical equipment was not included in the scope of 10 CFR 50.49. Existing dynamic and seismic qualification requirements are identified in Regulatory Guide 1.100 (Reference 19B.2.8-6).

Acceptance Criteria

The acceptance criteria for the resolution of Issue A-24 is that safety-related electrical equipment shall be environmentally qualified in accordance with 10 CFR 50.49, and dynamically and seismically qualified in accordance with the acceptance criteria of Regulatory Guide 1.100.

Resolution

The ABWR safety-related electrical equipment is environmentally qualified in accordance with 10 CFR 50.49 as described in Subsection 3.11 and dynamically and seismically qualified in accordance with Regulatory Guide 1.100 as described in Subsection 3.10.

References

19B.2.8-1 NUREG-0933, "A Prioritization of Generic Safety Issues (with Supplements 1-15)", U.S. NRC, April 1993.

19B.2.8-2 IEEE Std. 323, "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations", Institute of Electrical and Electronics Engineers.

19B.2.8-3 NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment", U.S. NRC, July 1981.

19B.2.8-4 10 CFR 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plant", Office of the Federal Register, National Archives and Records Administration.

19B.2.8-5 Regulatory Guide 1.89, "Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants", U.S. NRC.

19B.2.8-6 Regulatory Guide 1.100, "Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants", U.S. NRC.

19B.2.9 A-25: Non-Safety Loads on Class 1E Power Sources

Issue

Generic Safety Issue (GSI) A-25 in NUREG-0933 (Reference 19B.2.9-1), addresses the potential safety degradation of a Class 1E Power system caused by its connection to a non-safety-related power source or load.

There are two approaches to assuring the reliability of the safety-related system Class 1E power supplies for future plants. The first approach is to allow only Class 1E loads to be connected to Class 1E power supplies. [In previous designs, non-safety electrical equipment was connected to Class 1E power supplies (i.e., the emergency diesel generators) to provide a source of power during loss-of-offsite power (LOPP) events.]

The second approach is to limit the connection of non-safety-related electrical equipment to the Class 1E power systems and assure that when this equipment is connected to the Class 1E power systems that the equipment and the connections conform to the requirements for independence, electrical isolation, and physical separation. These requirements are identified in IEEE Standard 384 (Reference 19B.2.9-2), and guidance is provided in Regulatory Guide 1.75 (Reference 19B.2.9-3). [Supplemental information on Class 1E safety systems may be found in IEEE Standard 603, IEEE Standard 279, and IEEE Standard 308, (References 19B.2.9-4, 19B.2.9-5 and 19B.2.9-6, respectively).]

Both industry and the NRC, through IEEE Standard 384 and Regulatory Guide 1.75, have determined that these design requirements provide an acceptable means of achieving an adequate level of reliability for the Class 1E power supplies. Therefore, a commensurate level of safety for the safety systems is assured.

Acceptance Criteria

The acceptance criteria for the resolution of GSI A-25 is that the reliability and level of safety of Class 1E power sources and the safety systems which they supply may not be degraded by the sharing of loads between safety-related systems and non-safety-related systems.

Specifically, the second approach, identified in the issue statement, shall be used in establishing an acceptable level of reliability and safety for Class 1E power sources and safety-related systems.

This shall be accomplished by assuring that the interface between safety-related and non-safety-related equipment on Class 1E power sources and safety-related systems is adequately controlled by meeting the independence, electrical isolation, and physical separation requirements identified in IEEE Standard 384 and other applicable standards (References 19B.2.9-2, 19B.2.9-4, and 19B.2.9-6) taking into consideration the guidance provided in Regulatory Guide 1.75.

Resolution

The ABWR design assures the reliability and safety of the Class 1E power sources and safety-related systems by a highly selective connection (i.e., only one subsystem) of non-safety-related equipment and strict control of the interface between this subsystem and Class 1E power system. Each safety related system conforms to the requirements of IEEE Standard 384 (Reference 19B.2.9-2) and meets RG 1.75 (Reference 19B.2.9-3) and addresses IEEE Standard 603 (Reference 19B.2.9-4).

The ABWR design incorporates three independent Class 1E diesel generators (DGs) and a non-Class 1E combustion turbine generator (CTG). The CTG is designed to automatically and independently assume the plant investment protection (PIP) loads, should a LOPP event occur.

This is in much the same manner as the DGs assume the Class 1E loads for the same event. Therefore, it is not necessary for the Class 1E buses to assume the PIP loads. (See Subsections 8.2.1 and 8.3.1.)

The ABWR design excludes non-Class 1E from the Class 1E busses, with the exception of the fine-motion control rod drive (FMCRD) subsystem, the associated AC standby lighting system, and the associated DC emergency lighting system. The reliability of the FMCRD subsystem is enhanced for the anticipated transient without scram (ATWS) event by using Class 1E power for the drive motors.

The load breakers in the Division 1 bus are part of the isolation scheme between the Class 1E power and the non-Class 1E FMCRD loads. The Class 1E load breakers provide the needed isolation between the Class 1E bus and the non-Class 1E loads. (See Subsection 8.3.1.1.1 for more details on this feature relative to the FMCRD power circuits.)

Since both the safety systems and their Class 1E power supplies conform to the requirements of IEEE Standard 384 and meet the intent of Regulatory Guide 1.75, an acceptable level of safety exists for both the safety systems and their Class 1E power supplies.

Therefore, this issue is resolved for the ABWR.

References

19B.2.9-1 NUREG-0933, "A Prioritization of Generic Safety Issues (with Supplements 1-15)", U.S. NRC, April 1993.

19B.2.9-2 IEEE Standard 384, "Criteria for Separation of Class 1E Equipment and Circuits", The Institute of Electrical and Electronics Engineers, Inc.

19B.2.9-3 Regulatory Guide 1.75, "Physical Independence of Electric Systems", U.S. NRC.

19B.2.9-4 IEEE Standard 603, "Standard Criteria for Safety Systems for Nuclear Power Generating Stations", The Institute of Electrical and Electronics Engineers, Inc.

19B.2.9-5 Not Used.

19B.2.9-6 IEEE Standard 308, "Criteria for Class 1E Electric Systems for Nuclear Power Generating Stations", The Institute of Electrical and Electronic Engineers, Inc.

19B.2.10 A-31: Residual Heat Removal (RHR) Shutdown Requirements

Issue

Unresolved Safety Issue (USI) A-31 in NUREG-0933 (Reference 19B.2.10-1), addresses the safe shutdown of the reactor, following an accident or abnormal condition other than a Loss of Coolant Accident (LOCA), from a hot standby condition (i.e., the primary system is at or near normal operating temperature and pressure) to a cold shutdown condition. Considerable emphasis has been placed on long-term cooling which is typically achieved by the residual heat removal system which starts to operate when the reactor coolant pressure and temperature are substantially lower than the hot-standby values.

Even though it may generally be considered safe to maintain a reactor in a hot-standby condition for a long time, experience has shown that there have been abnormal occurrences that required long-term cooling until the reactor coolant system was cold enough to perform inspection and repairs. For this reason, the ability to transfer heat from the reactor to the environment, after a shutdown resulting from an accident or abnormal occurrence, is an important safety function. It is essential that a power plant be able to go from hot-standby to cold-shutdown conditions subsequent to any accident or abnormal occurrence condition.

Acceptance Criteria

The acceptance criterion for the resolution of USI A-31 is that the RHR system shall be designed so that the reactor can be brought from a "Hot Standby" to a "Cold Shutdown" condition as described in SRP Section 5.4.7, (Reference 19B.2.10-2).

Specifically, the RHR system shall meet the intent of the following functional requirements with respect to cooldown:

(1) The design shall be such that the reactor can be taken from normal operating conditions to cold shutdown using only safety-grade systems. These systems shall satisfy 10 CFR 50 Appendix A (Reference 19B.2.10-3) General Design Criteria (GDC) 1 through 5, and 34.

(2) The system(s) shall have suitable redundancy in components and features, and suitable interconnections, leak connection, and isolation capabilities to assure that for onsite electrical power system operation (assuming offsite power is not available) the system function can be accomplished assuming a single failure.

(3) The system shall be capable of being operated from the control room with either onsite or offsite power available. In demonstrating that the system can perform its function assuming a single failure, limited operator action outside of the control room would be considered acceptable, if suitably justified.

(4) The system(s) shall be capable of bringing the reactor to a cold shutdown condition, with either offsite or onsite power available, within a reasonable period of time following a shutdown, assuming the most limiting single failure.

In addition to the functional requirements listed above, there are certain additional requirements for the RHR system including, pressure relief, pump protection, test and operation.

Resolution

The Residual Heat Removal (RHR) system is composed of three electrically and mechanically independent divisions, except for the outboard containment isolation valves, which are in different electrical divisions than the inboard valves, designated as A, B, and C with each division containing the necessary piping, pumps, valves, and heat exchangers (Subsection 5.4.7).

One of the basic design functions of the RHR system is shutdown. Shutdown cooling to remove decay and sensible heat from the reactor, which also includes the safety-related requirements that the reactor must be brought to a cold shutdown condition using safety grade equipment (Subsection 5.4.7.1.1.7).

The design basis for the RHR Shutdown Cooling subsystem is that it is manually activated by the operator from the control room following insertion of the control rods and normal blowdown to the main condenser (Subsection 5.4.7.1.1.7).

For emergency operations where one of the RHR loops has failed, the RHR system is capable of bringing the reactor to the cold shutdown condition of 373 K (100°C) within 36 hours following reactor shutdown with any two of the three divisions. The subsystem can maintain or reduce this temperature further so that the reactor can be refueled and serviced (Subsection 5.4.7.1.1).

The RHR system is part of the Emergency Core Cooling (ECCS) System, and therefore is required to be designed with redundancy, piping protection, power separation, and other safeguards as required of such systems (Section 6.3).

Shutdown suction and discharge valves are required to be powered from both offsite and standby emergency power for purposes of isolation and shutdown following a loss of offsite power (Subsection 9A.5.5.14).

The RHR system is designed to meet General Design Criteria (GDC) 1, 2, 3, 4, and 5 for quality assurance, protection against natural phenomenon, environmental and internally generated missiles, pipe breaks, seismic effects, and fires (Subsection 5.4.7.1.6).

The RHR Shutdown Cooling System is designed to meet the intent of SRP Section 5.4.7, Revision 3, with respect to providing a means of bringing the reactor plant from hot standby to cold shutdown under all accident or abnormal occurrence conditions, as described above.

Therefore, this issue is resolved for the ABWR (Subsection 5.4.7.1.1.7).

References

19B.2.10-1 NUREG-0933, “A Prioritization of Generic Safety Issues (with Supplements 1-15)”, U.S. NRC, April 1993.

19B.2.10-2 NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants—LWR Edition”, U.S. NRC.

19B.2.10-3 10 CFR 50 Appendix A, “General Design Criteria for Nuclear Power Plants”, Code of Federal Regulations, Office of the Federal Register, National Archives and Records Administration.

19B.2.11 A-35: Adequacy of Offsite Power System

Issue

Issue A-35 in NUREG-0933 (Reference 19B.2.11-1) concerns the protection of safety-related equipment from the effects of a sustained undervoltage condition or a rapid rate of decay of the frequency of the offsite power source as well as interaction effects between offsite and onsite power sources. Associated testing requirements are also addressed.

The plant operator historically has performed transient and steady-state stability analyses of the offsite power system which were documented in the Safety Analysis Report (SAR). However, abnormal occurrences at several operating plants indicated that a sustained undervoltage condition of the offsite power source not detected by the existing loss of voltage relays could result in a failure of redundant safety-related equipment.

The NRC therefore evaluated the power systems of operating plants to determine the susceptibility of safety-related electrical equipment to:

(1) A sustained undervoltage condition on the offsite power source.

(2) A rapid rate of decay of the offsite power source frequency.

(3) Interaction for the offsite and onsite power sources.

An additional factor evaluated was

(4) The adequacy of testing requirements.

New criteria relative to factors (1), (3) and (4) above were issued in Branch Technical Position (BTP) PSB-1 “Adequacy of Station Electric Distribution System Voltages”, which was incorporated in SRP Section 8.3.1, Appendix A (Reference 19B.2.11-2). Frequency decay [factor (2)] was found not to be a significant safety issue.

Acceptance Criteria

The acceptance criteria for the resolution of USI A-35 is that the design and capability for test and calibration of the undervoltage protection schemes for the Class 1E buses of the onsite power system (while connected to the offsite power source) shall conform to the guidance of BTP PSB-1 in Appendix A of SRP Section 8.3.1.

Specifically, a second level of voltage protection shall be provided for Class 1E equipment in addition to the existing protection based on detecting the complete loss of offsite power to the Class 1E buses. The second level shall have two separate time delays before alerting the control room operator and automatically separating the Class 1E buses from the offsite power source, respectively. Duration of the time delays shall ensure protection from sustained low voltage while avoiding disconnection from the offsite source due to short term transients such as motor starting. The undervoltage protection scheme shall have the capability of being tested and calibrated during power operation. Voltage levels at the safety related buses shall be optimized for the maximum and minimum load conditions that are expected, throughout the anticipated range of offsite power source voltage variation. Technical Specifications are to include limiting conditions of operation, surveillance requirements, and protection equipment setpoints.

Resolution

The conceptual design of an offsite power system and station switchyard(s) for the ABWR design is given in Section 8.2. The interface requirements will ensure that the switchyard(s) provide redundant offsite power feed capability to the nuclear unit, consisting of two preferred power circuits, each capable of supplying the necessary safety loads and other equipment.

The ABWR onsite power systems are described in Section 8.3, and include three redundant and independent 4.16kV Class 1E safety buses. The incoming source breakers trip upon loss of normal power, and emergency power is provided to each Class 1E bus by separate and independent diesel generator (DG) units. A combustion turbine generator automatically assumes the plant investment protection loads, but can be used to manually provide back-up power for any Class 1E bus, should a DG fail or be out of service.

The Class 1E AC Power Systems are described in Subsection 8.3.1.1. Protection against degraded voltage is specifically addressed in Subsection 8.3.1.1.7(8). The protection schemes are designed according to the recommendations of IEEE Standard 741 (Reference 19B.2.11-3), which is consistent with the guidance of BTP PSB-1.

The ABWR Standard Plant Class 1E auxiliary power system is designed in compliance with General Design Criterion (GDC) 18 (Reference 19B.2.11-4) so that inspection, maintenance, calibration and testing can be carried out with a minimum of interference with operation of the nuclear unit, as described in Subsection 8.3.1.1.5.3. On-line testing is greatly enhanced by the design, which utilizes three independent Class 1E divisions. Indication of the system unavailability is provided in the control room.

A Technical Specification establishes limiting conditions for operations, surveillance requirements, trip setpoints with minimum and maximum limits, and allowable values for the undervoltage protection sensors and associated time delay devices.

Protection of the Class 1E power supplies to safety-related equipment from the effects of an undervoltage condition of the offsite power source thus conforms to the guidance of BTP PSB-1, and this issue is, therefore, resolved for the ABWR Standard Plant design.

References

19B.2.11-1 NUREG-0933, “A Prioritization of Generic Safety Issues (with Supplements 1-15)”, U.S. NRC, April 1993.

19B.2.11-2 NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants—LWR Edition”, U.S. NRC.

19B.2.11-3 ANSI/IEEE 741, “Criteria for the Protection of Class 1E Power Systems and Equipment in Nuclear Power Generating Stations”, Institute of Electrical and Electronics Engineers, Inc.

19B.2.11-4 10 CFR 50 Appendix A, “General Design Criteria for Nuclear Power Plants”, Code of Federal Regulations, Office of the Federal Register, National Archives and Records Administration.

19B.2.12 A-36: Control of Heavy Loads Near Spent Fuel

Issue

Issue A-36 in NUREG-0933 (Reference 19B.2.12-1), addresses the consequence of dropping heavy loads on spent fuel. Overhead cranes are used to lift heavy objects in the vicinity of spent fuel. If the heavy object, such as a spent-fuel shipping cask or shielding block, were to fall on to spent fuel, there could be a release of radioactivity to the environment that could exceed 10 CFR 100 guidelines. This issue was resolved by the NRC with the publication of NUREG-0612 (Reference 19B.2.12-2) and SRP Section 9.1.5 (Reference 19B.2.12-3).

Acceptance Criteria

The acceptance criteria for the resolution of Issue A-36 is that the overhead 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

The ABWR design addresses the above criteria as follows:

(1) A transportation routing study will be made of all planned heavy load handling moves to evaluate and minimize safety risks. The study will require the COL to establish the heavy load handling safe load paths and routing plans (Subsections 9.1.5.5, 9.1.5.8 and 9.1.6.6).

(2) The major heavy load handling equipment components (i.e., cranes, hoists, etc.) will be provided with an operating instruction and maintenance manual for reference and utilization by operations and maintenance personnel for use in operating procedures,

maintenance procedures and operator training programs. The handling equipment operating procedures will comply with the requirements of NUREG-0612, Subsection 5.1.1(2) (Subsections 9.1.5.4, 9.1.5.8 and 9.1.6.6).

(3) Crane inspections and testing will comply with the requirements of ANSI B30.2 and NUREG-0612, Subsection 5.1.1(6). The COL applicant will provide the heavy load handling system and equipment inspection and test plans (Subsections 9.1.5.6, 9.1.5.8 and 9.1.6.6).

(4) The equipment handling components, including the reactor building crane and the refueling machine crane, used over the fuel pool are designed to meet the single failure proof criteria of NUREG-0554 (Reference 19B.2.12-8). Redundant safety interlocks and limit switches are provided to prevent transporting heavy loads other than spent fuel by the refueling machine crane, over any spent fuel that is stored in the spent-fuel storage pool (Subsections 9.1.5.2.1 and 9.1.5.5).

(5) The reactor vessel head lifting strongback and the dryer/separator lifting strongback are designed in accordance with the acceptable factors of safety. This is in accordance with ANSI-N14.6 (Reference 19B.2.12-5) and in accordance with NUREG-0612 (Subsection 9.1.4.2.5).

(6) The heavy load handling system is designed in accordance with relevant requirements of GDC 2, 4, and 61 and the guidance of References 19B.2.12-2, and 19B.2.12-5 through 19B.2.12-7. The ABWR design is for a single unit; therefore, GDC 5 is not applicable (Subsection 9.1.5.1 and Section 3.1).

The acceptance criteria for this safety issue are met and, therefore, the issue is resolved for the ABWR design.

References

19B.2.12-1 NUREG-0933, “A Prioritization of Generic Safety Issues (with Supplements 1-15)”, U.S. NRC, April 1993.

19B.2.12-2 NUREG-0612, “Control of Heavy Loads at Nuclear Power Plants”, U.S. NRC.

19B.2.12-3 NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants—LWR Edition”, U.S. NRC.

19B.2.12-4 10 CFR 50 Appendix A, “General Design Criteria for Nuclear Power Plants”, Code of Federal Regulations, Office of the Federal Register, National Archives and Records Administration.

19B.2.12-5 ANSI-N14.6, “American National Standard for Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 KG) or More for Nuclear Materials”, American National Standards Institute.

19B.2.12-6 ANSI/ANS-57.2, “Design Requirements for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Plants”, American Nuclear Society.

19B.2.12-7 ANSI/ANS-57.1, “Design Requirements for Light Water Reactors Fuel Handling Systems”, American Nuclear Society.

19B.2.12-8 NUREG-0554, “Single Failure Proof Cranes for Nuclear Power Plants”, U.S. NRC.

19B.2.13 A-39: Determination of Safety Relief Valve Pool Dynamic Loads and Temperature Limits

Issue

Operation of BWR primary system pressure relief valves can result in hydrodynamic loads on the suppression pool retaining structures located within the pool. These loads result from initial vent clearing of relief valve piping and steam quenching due to high local pool temperatures. This issue addresses GE MARK I, II, and III containments.

Acceptance Criteria

The acceptance criteria set forth for quencher discharge loads are applicable only to the cross-quencher configuration described in Attachment A to Appendix 3B of General Electric Standard Safety Analysis Report II (GESSAR II), Revision 1. Deviation from this configuration shall be reviewed on a plant-unique basis. And acceptability of suppression pool temperature limit(s) shall be based on conformance with the resolution of the issue specified in Section 5 of NUREG-0783 (Reference 19B.2.13-1).

Resolution

Safety/Relief Valves (SRVs) are utilized in a BWR pressure suppression system to provide pressure relief during certain reactor transients. SRV steam flow is routed through discharge lines into the pressure suppression pool, where it is condensed. Each discharge pipe is fitted at the end with a device called a quencher to promote heat transfer during SRV actuation between the high temperature compressed air and steam mixture and the cooler water in the suppression pool. This enhances heat transfer while providing a low amplitude oscillating pressure in the pool and eliminates concern over operation at a high suppression pool temperature. For ABWR plants, the discharge device is an X-quencher such as has been used in prior plants (Section 3B.2.1).

Following the actuation of a SRV, water contained initially in the discharge line is rapidly discharged through the X-Quencher discharge device attached at the end of the SRV discharge line. A highly localized water jet is formed around the X-Quencher arms. The hydrodynamic load induced outside a sphere circumscribed around the quencher arms by the quencher water jet is not significant. This is the first phase of loading on the suppression pool boundary due to the SRV blowdown. There are no submerged structures located within the sphere mentioned above in the ABWR arrangement. The induced load for submerged structures located outside the circumscribed sphere by the quencher arm is negligible and is ignored (Section 3B.5.4).

After the water discharge, the air initially contained in the discharge line is forced into the suppression pool under high pressure. The air bubbles formed interact with the surrounding water and produce oscillating pressure and velocity fields in the suppression pool. This pool disturbance (air-clearing) gives rise to hydrodynamic loads which are the second phase of SRV blowdown loading on submerged structures in the pool and on the pool boundary (Section 3B.5.4).

The final stage of SRV blowdown is the steady steam flow phase. Submerged structure and pool boundary loading is from condensing steam jet oscillations at the quencher (Reference 19B.2.13-1).

This issue was resolved with the issuance of SRP Section 6.2.1.1.C (Reference 19B.2.13-2). NUREG-0763 (Reference 19B.2.13-3), NUREG-0783 (Reference 19B.2.13-1), and NUREG-0802 (Reference 19B.2.13-4) were also issued for Mark I, II, and III containments, respectively, and the load definition for dynamic loads on submerged structures was developed on the basis of cumulative information that was described in Issues A-7, A-8, and B-10 (Subsections 19B.2.3, 19B.2.4, and 19B.2.19, respectively). The load definition methodology for defining the SRV air bubble loads on submerged structures will be consistent with that used for prior plants.

Therefore, this issue is resolved for the ABWR.

References

19B.2.13-1 NUREG-0783, “Suppression Pool Temperature Limits for BWR Containments,” U.S. NRC, November 1981.

19B.2.13-2 NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants—LWR Edition”, U.S. NRC.

19B.2.13-3 NUREG-0763, “Guidelines for Confirmatory In-plant Tests of Safety Relief Valve Discharges for BWR Plants”, U.S. NRC, May 1981.

19B.2.13-4 NUREG-0802, “Safety/Relief Valve Quencher Loads: Evaluation for BWR Mark II and III Containments”, U.S. NRC, October 1982.

19B.2.14 A-40: Seismic Design Criteria Short-Term Program

Issue

Issue A-40 in NUREG-0933 (Reference 19B.2.14-1) addresses short-term improvements in seismic design criteria.

The seismic design sequence for recently designed plants included many conservative factors. Although it is believed that the overall sequence was adequately conservative, certain aspects may not have been conservative for all plant sites. The objective of A-40 was to investigate selected areas of the seismic design sequence to determine their conservatism for all types of sites, to investigate alternative approaches where desirable, to quantify the overall conservatism of the design sequence, and to modify the NRC criteria in the Standard Review Plan (Reference 19B.2.14-2), where justified.

Studies were conducted, and the results were documented in NUREG/CR-1161 (Reference 19B.2.14-3) with specific recommendations for changes in seismic design requirements. In addition, an NRC/Industry workshop was held to discuss the complex and controversial subject of soil-structure interaction (SSI) analysis. The adequacy of seismic design of large, above ground, vertical, safety-related tanks was also of concern to the NRC.

Standard Review Plan (SRP) sections were then revised (Revision 2) with the following principal areas of change: Section 2.5.2, updated to reflect the current NRC staff review practice; Section 3.7.1, design time history criteria; Section 3.7.2, development of floor response criteria, damping values, SSI uncertainties, and combination of modal responses; and Section 3.7.3, seismic analysis of above ground tanks, and Category 1 buried piping.

The NRC concluded in NUREG-1233 (Reference 19B.2.14-4) that these revisions would reflect the current state-of-the-art in seismic design in the licensing process. Implementation of the SRP revisions is expected to contribute to a more uniform and consistent licensing process and is not expected to have significant impact on recently designed plants.

Acceptance Criteria

The acceptance criterion for the resolution of A-40 is that future nuclear power plants shall conform to the seismic design acceptance criteria and guidance of Revision 2 to SRP Sections 2.5.2, Vibratory Ground Motion; 3.7.1, Seismic Design Parameters; 3.7.2, Seismic System Analysis; and 3.7.3, Seismic Subsystem Analysis.

Specifically, these SRP Sections respectively cover review of the site characteristics and earthquake potential, the parameters to be used in seismic design, methods to be used in seismic analysis of the overall plant, and methods to be used in seismic analysis of individual systems or components.

Resolution

The design ground motions, site envelope soil parameters, and system and subsystem analyses criteria and methods described in Sections 2.3.2.22, 3.7.1, 3.7.2 and 3.7.3 meet the intent of Revision 2 of the corresponding SRP sections, except that the OBE is not a design requirement for the ABWR. Elimination of the OBE from the design in advanced reactors is consistent with Policy Issue SECY-93-087 (Reference 19B.2.14-5).

This issue is, therefore, resolved for the ABWR Standard Plant design.

References

19B.2.14-1 NUREG-0933, “A Prioritization of Generic Safety Issues (with Supplements 1-15)”, U.S. NRC, April 1993.

19B.2.14-2 NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants—LWR Edition", U.S. NRC.

19B.2.14-3 NUREG/CR-1161, "Recommended Revisions to Nuclear Regulatory Commission Seismic Design Criteria", U.S. NRC, May 1980.

19B.2.14-4 NUREG-1233, Regulatory Analysis for USI A-40, "Seismic Design Criteria", U.S. NRC, September 1989.

19B.2.14-5 Policy Issue SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs," April 1993.

19B.2.15 A-42: Pipe Cracks in Boiling Water Reactors

Issue

Issue A-42 in NUREG-0933 (Reference 19B.2.15-1), addresses the past occurrences of intergranular stress corrosion cracking (IGSCC) in BWR austenitic steel components. Safe ends, short transition pieces between vessel nozzles and the piping, that have been highly sensitized by furnace heat treatment while attached to vessels during fabrication, were in the late 1960's found to be susceptible to IGSCC.

Acceptance Criteria

The acceptance criteria for the resolution of A-42 are that IGSCC resistant materials and fabrication techniques to minimize sensitization shall be used. In addition, the ABWR water shall be maintained at the lowest practically achievable impurity levels. Furthermore, the material and fabrication techniques shall comply with the guidelines of NUREG-0313 (Reference 19B.2.15-2).

Resolution

For the ABWR, IGSCC resistance is achieved through the use of Type 316 stainless steel and compliance with the guidelines of NUREG-0313. All materials are supplied in the solution heat treated condition. During fabrication, any heating operations (except welding) between 700 and 1255 K (427 and 982°C) are avoided, unless followed by solution heat treatment. The ABWR water is maintained at the lowest practically achievable impurity levels to minimize its corrosion potential.

In summary, only stainless steel type 316 material is used and all austenitic steel components are fabricated in accordance with NUREG-0313.

Therefore, this issue is resolved for the ABWR Standard Plant design.

References

19B.2.15-1 NUREG-0933, "A Prioritization of Generic Safety Issues (with Supplements 1-15)", U.S. NRC, April 1993.

19B.2.15-2 NUREG-0313, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping," US. NRC, July 1977, (Revision 1) July 1980, (Revision 2) January 1988.

19B.2.16 A-44: Station Blackout

Issue

The total loss of AC power (that is, the loss of AC power from both the off-site and on-site sources) is referred to as a station blackout. In the event of a station blackout, the capability to cool the reactor core is dependent on the availability of systems that do not require AC power and on the ability to restore offsite or onsite AC power before other means of cooling the core are lost. The concern is that a prolonged station blackout might result in a core damage accident (Reference 19B.2.16-1).

Acceptance Criteria

The acceptance criteria for the resolution of this issue for evolutionary ALWRS is compliance with:

- SECY-90-016–Evolutionary LWR Certification Requirement (Reference 19B.2.16-1)
- NRC Commissioner Policy Statement Certification Requirement (Reference 19B.2.16-2)
- 10 CFR 50.63, "Loss of all Alternating Current Power" (Reference 19B.2.16-3)
- Regulatory Guide 1.155, "Station Blackout" (Reference 19B.2.16-4)
- NUMARC-87-00 Guidelines and Technical Basis for Resolution of SBO (Reference 19B.2.16-5)
- EPRI-URD–Utility Requirements for Evolutionary LWRS (Reference 19B.2.16-6)

Resolution

The ABWR design satisfies the acceptance criteria by demonstrating (in Appendix 19E.2.1.2.2) that the ABWR can withstand a station blackout without core damage or loss of containment integrity through the use of the combustion turbine generator as an alternate AC source. Station blackout is also addressed in Appendix 1C to show compliance with applicable regulations.

Therefore, this issue is resolved for the ABWR.

References

19B.2.16-1 SECY-90-016, "Evolutionary LWR Certification Issues and Their Relationship to Current Regulatory Requirements," January 12, 1990.

19B.2.16-2 Letter J. Taylor to S. Chilk, "Evolutionary LWR Certification Issues and Their Relationship to Current Regulatory Requirements", June 26, 1990.

19B.2.16-3 10 CFR 50.63, “Loss All Alternating Current Power (Station Blackout–SBO)”, July 21, 1988.

19B.2.16-4 Regulatory Guide 1.155, “Station Blackout.”

19B.2.16-5 NUMARC-87-00, “Guidelines and Technical Basis for NUMARC Initiation Addressing Station Blackout at LWR’s”, Plus Supplemental Q/A, January 4, 1990.

19B.2.16-6 “Advanced Light Water Reactor Utility Requirements Document, Volume II; EPRI”, July 1990.

19B.2.17 A-47; Safety Implications of Control Systems

Issue

This issue, A-47, concerns the potential for accidents or transients (e.g., overpressure, overfilling, reactivity events) being made more severe as a result of control system failures including control and instrumentation power support faults. These failures or malfunctions may occur independently or as a result of an accident or transient and would be in addition to any control system failure that may have initiated the event. Although it is generally believed that control system failures are not likely to result in loss of safety functions which could lead to serious events or result in conditions that safety systems are not able to cope with, in-depth studies have not been performed. The NRC evaluated the effects of control system failures on three operating plants and identified the concern of steam generator overfill by feedwater. Subsequently, GL89-19 (Reference 19B.2.17-3) was issued to require all operating plants and plants under construction to provide automatic feedwater overfill protection.

Acceptance Criteria

The acceptance criteria for resolution is that the plant shall provide automatic reactor vessel overfill protection, and that plant procedures and technical specifications shall include provisions to verify periodically the operability of the overfill protection to assure that automatic overfill protection is available to mitigate main feedwater overfeed events during reactor power operation. Also, the system design and setpoints shall be selected with the objective of minimizing inadvertent trips of the main feedwater system during plant startup, normal operation, and protection system surveillance.

Resolution

The reactor vessel overfill protection is described in Subsection 7.7.1.4(9) and Figure 7.7-8. Plant procedures will be developed by the COL applicant. As a matter of good design practice for maximum availability, the feedwater system design and setpoints will be selected to minimize inadvertent trips for all modes of operation. This system, with fault tolerant digital controllers and self-test and on-line diagnostics, is described in Subsection 7.7.1.4.

Feedwater control (FDWC) system uses the non-Class 1E level transmitter signals in providing Level-8 trip instrumentation and part of the diverse ATWS trip logic. For this reason, reliance on the Class 1E level transmitters by FDWC would partially defeat the diversity of the ATWS design. In addition, three channels of reactor vessel water level high instrumentation are provided as input to a two-out-of-three logic to minimize inadvertent trips of the main feedwater system.

The LCO of the feedwater trip instrumentation is identified in Technical Specifications 3.3.4.2. Operation restrictions are provided in technical specifications to assure the overfill protection availability. Incorporated in these restrictions is a periodic evaluation of the feedwater trip instrumentation condition which considers such availability items as correct channel sensor operation and proper channel function, calibration and logic system operation. Additionally, the COL applicant will be required to incorporate these requirements into plant operating procedures.

Plant operating procedures and operator training will ensure that the operators can mitigate reactor vessel overfill events that may occur by injection via the condensate pumps and condensate booster pumps during reduced pressure CFS operation. Plant procedures and training will be developed by the COL applicant.

Therefore, this issue is resolved for the ABWR.

References

19B.2.17-1 NUREG-0933, “A Prioritization of Generic Safety Issues (with Supplements 1-15)”, U.S. NRC, April 1993.

19B.2.17-2 NUREG-1217, “Evaluation of Safety Implications of Control Systems in LWR Nuclear Power Plants”, June 1989.

19B.2.17-3 Generic Letter 89-19, “Request for Action Related to Resolution of USI A-47, Pursuant to 10CFR50.54(f)”, U.S. NRC, September 20, 1989.

19B.2.18 A-48: Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment

Issue

In the unlikely event of a degraded core accident (or following a LOCA) in a light water reactor plant, it is postulated that the result is the release of large quantities of combustible gases, principally hydrogen, that may accumulate inside the primary reactor containment. These gases would be the result of:

(1) A metal-water reaction involving the fuel element cladding. Hydrogen in significant quantity can be formed as a result of the reaction of zirconium fuel cladding at high temperature with steam.

(2) The radiolytic decomposition of the water in the reactor core and the containment sump.

(3) The corrosion of certain construction materials by the spray solution.

(4) Any synergistic chemical, thermal, and radiolytic effects of post-accident environmental conditions on containment protective coating systems and electric cable insulation.

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), requires 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 is acceptable as a hydrogen control measure.

Resolution

The issue of a large amount of hydrogen being generated and burned within containment was resolved as stated in the NRC document SECY 89-122 dated April 19, 1989 (Reference 19B.2.18-3). This issue covers hydrogen control measures for recoverable degraded core accidents for all BWRs. Extensive research in this area has led to significant revision of the Commission’s hydrogen control regulations, given in 10 CFR 50.44, published September 16, 2003.

The ABWR containment is inerted and per 10 CFR 50.34 (f)(2)(ix) can withstand the pressure and energy addition from a 100% fuel-clad metal-water reaction. However, in the ABWR, there are no design-basis events that result in core uncovery or core heatup sufficient to cause significant metal-water reaction. Section 6.2.5.3 states that this is equivalent to the reaction of the active clad to a depth of 5.842E-3 mm (0.00023 inches) or 0.72% of the active clad.

Therefore, this issue is resolved for the ABWR.

References

19B.2.18-1 10 CFR 50.44, “Standards for Combustible Gas Control System in Light-Water-Cooled Power Reactors”, Office of the Federal Register, National Archives Records Administration.

19B.2.18-2 10 CFR 50 Appendix A, “General Design Criteria for Nuclear Power Plants.”

19B.2.18-3 SECY-89-122, "Resolution of Unresolved Safety Issue A-48, Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment", April 1989.

19B.2.18-4 10 CFR 50.34, "Contents of Applications; Technical Information", Office of the Federal Register, National Archives Records Administration.

19B.2.19 B-10: Behavior of BWR Mark III Containments

Issue

Evaluation and approval is required of various aspects of the MARK III containment design which differs from the previously reviewed MARK I and MARK II designs. The task involves the completion of the staff evaluation of the MARK III containment and documentation of the method used to validate the analytical models and assumptions needed to predict the containment pressures in the event of a LOCA (Reference 19B.2.19-1).

Following a postulated LOCA, escaping steam forces the steam-air mixture out of the drywell into the wetwell. This action results in pool swell and loads from vent clearing, jets, chugging, impact of water, impact from froth impingement, pool fallback, condensation, and containment pressure.

The concern is that these loadings may damage structures and components located within the wetwell. Although many of these structures (e.g., walkways) are by themselves not related to safety, the various ECCS systems take suction from the wetwell and, therefore, damage in the wetwell may affect the performance of the ECCS (Reference 19B.2.19-2).

Acceptance Criteria

On the basis of certain large-scale tests conducted between 1973 and 1979, the General Electric Company developed LOCA-related hydrodynamic load definitions for use in the design of the standard Mark III containment. The NRC staff and its consultants have reviewed these load definitions and their bases and conclude that, with a few specified changes (Reference 19B.2.19-2), the proposed load definitions provide conservative loading conditions.

The staff approved acceptance criteria for LOCA-related hydrodynamic loads are provided in NUREG-0978, Appendix C (Reference 19B.2.19-2). The following describes how the acceptance criteria are applied. The staff will review each applicant's use of the NRC acceptance criteria for applicability to their plant design. Mark III applicants for a construction permit (CP) need only furnish a commitment to use the staff's acceptance criteria in the design of their containment. Mark III applicants for an operating license (OL) will be required to show how the NRC acceptance criteria were applied and to justify any deviations taken. For both CP and OL applicants, the information required shall be submitted in a timely manner to allow for the evaluation to be included in the plant's Safety Evaluation Report, or supplements thereto (Reference 19B.2.19-2).

The ABWR horizontal vent confirmatory test program was performed to obtain data which could be used to determine condensation oscillation and chugging loads for design evaluation of containment structures. The test matrix included tests at conditions which produce bounding loads and additional tests to examine the sensitivity of the loads to system parameters. The test specifically documents work performed, including general evaluation of the test data and the specification of procedures which can be used to define containment loads.

Resolution

The ABWR design utilizes a horizontal vent system, which is similar to the Mark III design, but includes some ABWR design features. These unique features include pressurization of the wetwell airspace, the presence of a lower drywell, the smaller number of horizontal vents (76.2 cm ABWR vs. 304.8 cm Mark III), extension of horizontal vents into the pool, vent submergence, and suppression pool width, as described in Subsection 3B.1.2.

The ABWR horizontal vent test (HVT) program produced test data to confirm and define condensation oscillation (CO) and chugging (CH) loads for design application. The test demonstrated that a blowdown test facility can be constructed to be very rigid and thereby eliminate fluid-structure interaction effects. It was also shown that a scaled test facility can be used to obtain condensation data for full-scale design application. Most important, an extensive data base which can be used for confirmation of ABWR CO and CH loads was obtained

A spectrum of postulated loss-of-accidents (LOCAs) is considered in assessing the design adequacy of the ABWR containment system. Each of the accident conditions is described in Subsection 3B.2.2. The load definition methodology for defining the LOCA induced loads on submerged structures is consistent with the methodology used for prior plants, as described in Section 3B.5. The ABWR is designed to meet the NRC acceptance for Mark III LOCA-related pool dynamic loads.

Therefore, this issue is resolved for the ABWR plants.

References

19B.2.19-1 NUREG-0471, “Generic Task Problem Descriptions (Categories B, C, and D)”, US. NRC, June 1978.

19B.2.19-2 NUREG-0978, “Mark III LOCA-Related Hydrodynamic Load Definition”, U.S. NRC, February 1984.

19B.2.19-3 NUREG-0933, “A Prioritization of Generic Safety Issues (with Supplements 1-15)”, U.S. NRC, April 1993.

19B.2.19-4 NEDC-31393, “Containment Horizontal Vent Confirmatory Test Part 1 – Final Report”, March 1987.

19B.2.20 B-17: Criteria for Safety-Related Operator Actions

Issue

This issue, B-17, involves developing criteria for safety-related operator action (SROA) during the response to or recovery from transients and accidents. The criteria would include a determination of actions that shall be automated in lieu of operator action and the development of a time criterion for SROA. Specifically, to be determined for PWRs, is whether or not to require an automatic switchover from the injection mode to the recirculation mode following a LOCA. This issue has been resolved with no new or revised requirements.
(Reference 19B.2.20-1)

Acceptance Criteria

The acceptance criteria for the resolution of Issue B-17 is that the plant transient response time (i.e., time required for safety systems or operator to act) shall be increased over current plants to improve operability, and that the plant design shall permit increased operator response time, including a determination on the need for automatic actuation. Required time before the operator must act shall be not less than 20 minutes with a target of 30 minutes, assuming a single failure. Best estimate methodology shall be used for analysis to show safety limits are not exceeded. Operational inputs should be obtained from experienced operators.

Resolution

The ABWR design satisfies the Acceptance Criteria concerning automation of safety-related operator actions and operator response times. The ABWR resolution is the same as the ALWR resolution. For example, the ABWR design requires no operator action earlier than thirty minutes for any design basis accident as described in responses to questions 420.81, 420.82, 420.83, 430.26 in Subsections 20.3.8, 20.3.2 on operator performance under the range of Loss-of-Coolant accidents. The ABWR design by incorporating the RHR heat exchanger in the ECCS injection loop has eliminated the need for operator actions for several accidents/transients (Subsection 5.4.7.1.1.1). In fact, even in the long term, operator action is only required for one situation—initiation of containment cooling (Subsection 5.4.7.1.1.6). This is a relatively simple action and some delay in this action should have no adverse consequences, thus eliminating the need to automate this function. In addition, advance Cathode Ray Tubes (CRTs) in the control room shall be utilized for monitoring and alarm functions for safety-related and non-safety-related systems (References 19B.2.20-2, 19B.2.20-3, and 19B.2.20-4). To achieve this goal, information displays, controls and other interface devices in the control room and other plant areas are designed and implemented with good human factors engineering and in compliance with pertinent regulations regarding separation and independence (Section 18.2).

Therefore, this issue is resolved for the ABWR.

References

19B.2.20-1 NUREG-0933, "Resolution of Generic Safety Issues (with Supplements 1-33)", U.S. NRC, August 2010.

19B.2.20-2 EPRI NP-4361, “Power Plant Alarm Systems: A Survey and Recommended Approach For Evaluating Improvements”, December 1985.

19B.2.20-3 EPRI NP-5693P, “Evaluation of Alternative Power Plant Alarm Presentations.”

19B.2.20-4 EPRI NP-3448, “A Procedure For Reviewing and Improving Power Plant Alarm Systems”, April 1984.

19B.2.20-5 ANSI/ANS 58.8, “Time Response Design Criteria for Nuclear Safety Related Operator Actions”, American Nuclear Society.

19B.2.21 B-36: Develop Design, Testing, and Maintenance Criteria for Atmosphere Cleanup System Air Filtration and Adsorption Units for Engineered Safety Features Systems and for Normal Ventilation Systems

Issue

Issue B-36 in NUREG-0471 (Reference 19B.2.21-1) addresses technical advances in design, testing and maintenance criteria for atmosphere cleanup system air filtration and adsorption unit for engineered safety features systems and for normal ventilation systems.

Any technological advances leading to better methods and/or standards for these design, testing and maintenance criteria for these systems in light-water-cooled nuclear power plants need to be documented for NRC staff guidance and technical positions.

However no new design requirements have been established by the Nuclear Regulatory Commission (NRC) other than those of Revision 1 of Regulatory Guide (RG) 1.140 (Reference 19B.2.21-3) and Revision 2 of Regulatory Guide 1.52 (Reference 19B.2.21-2).

Acceptance Criteria

The acceptance criteria for the resolution of Issue B-36 are contained in RG 1.140, Revision 1, and RG 1.52, Revision 2. Both deal with design, testing, and maintenance criteria for air filtration and adsorption units of light-water-cooled nuclear power plants. RG 1.140 specifically applies to the non-safety-related normal ventilation exhaust system. RG 1.52 covers the criteria for post-accident engineered safety features.

Resolution

The filter systems required to perform safety-related functions following a design basis accident are the standby gas treatment system (SGTS) and the control room habitability system as described in Sections 6.4 (Habitability Systems) and 9.4.1.1 (Control Room Habitability Area HVAC), and Subsection 6.5.1 (Engineered Safety Features Filter Systems). The SGTS consists of two parallel and redundant filter trains. Each filter train is designed to have a HEPA filter installed at both inlet and outlet sides of the charcoal adsorber. The CRHA HVAC system is provided with redundant divisions. Each division consists of an emergency filtration unit. A HEPA filter is also provided before and after the charcoal adsorber of each emergency filtration unit. The HEPA filters of these systems will be tested periodically with DOP using the installed instrumentation in conformance with the guidance of SRP Table 6.5.1-1 and as described in Appendix 6B, for SGTS, and Appendix 9D, for CRHA HVAC systems and test connections as required by RG 1.52. Additionally, both of these systems address RG 1.52 as described in Subsection 6.5.1.3.5, Appendix 6A (compliance with RG 1.52), Subsection 9.4.1.1.7 (RG 1.52 Compliance Status), and Appendix 9C.

Air filtration and adsorption units are not required for normal ventilation on ABWR, since there are no requirements for safety-related adsorption units in normal operations, except for the incinerator off-gas exhaust which is directed to a separate monitor vent (Subsection 9.4.6.5.3). Therefore, RG 1.140 is not applicable.

Thus, Issue B-36 is resolved for ABWR.

References

19B.2.21-1 NUREG-0471, "Generic Task Problem Descriptions (Categories B, C, and D)", U.S. NRC, June 1978.

19B.2.21-2 Regulatory Guide 1.52, "Design, Testing, and Maintenance Criteria for Post-Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants", U.S. NRC, March 1978.

19B.2.21-3 Regulatory Guide 1.140, "Design, Testing, and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants", U.S. NRC.

19B.2.22 B-55: Improved Reliability of Target Rock Safety/Relief Valves

Issue

Many of the valves in BWR main steam pressure relief systems are Target Rock safety/relief valves, and a significant number of failures of these valves have occurred. Failures include valves:

(1) Failing to open properly on demand.

(2) Opening spuriously and then failing to reseat properly.

(3) Opening properly and then failing to reseat properly.

The failure of a pressure relief system valve to open on demand results in a decrease in the total available pressure-relieving capacity of the system. Spurious openings of pressure relief system valves, or failures of valves to properly reseat after opening, can result in inadvertent reactor coolant system blowdown with unnecessary thermal transients on the reactor vessel and the vessel internals, unnecessary hydrodynamic loading of the containment systems' pressure-suppression chamber and its internal components, and potential increases in the release of radioactivity to the environs. In addition, if the valve also serves as part of the ADS, a degradation of the capability of the ADS to perform its emergency core cooling function could result.

Acceptance Criteria

In the late 1970s, the NRC staff concluded that the inadvertent blowdown events that had occurred as a result of malfunctions of pressure relief system valves had neither significantly affected the structural integrity or capability of the reactor vessel or its internals or the pressure-suppression containment system, nor resulted in any significant radiation releases to the environment. Even if such events were to occur more frequently, there would not likely be any significant effects. Issue B-55 in NUREG-0933 (Reference 19B.2.22-1) requires that the performance of these valves be under continual surveillance and the consequences of their failures be subject to review.

Resolution

Main steam safety/relief valves for ABWR service will be similar to the dual-function direct-acting SRV types currently in service in GE BWR/5 and BWR/6 plants. These SRV types have demonstrated improved in-service performance and reliability as compared to pilot-operated safety/relief valves used on earlier BWR models.

The B-55 issue is not applicable to the ABWR. The ABWR uses a direct acting safety/relief valve design described in Subsection 5.2.2.4.1. This design does not have a pilot stage such as that present in the Target Rock pilot operated safety/relief valve. Therefore, the typical mechanisms which cause the pilot valve to open spuriously and to fail to open properly are not applicable to the ABWR design. It is these mechanisms which have caused the most serious concerns with the Target Rock safety/relief valve performance. By adopting a direct acting safety/relief valve design, these most serious concerns are eliminated in the ABWR.

The B-55 issue is only applicable to those operating BWRs with Target Rock pilot-operated safety/relief valves. Compliance with existing regulations, such as 10 CFR 50 Appendix B, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants,” and 10 CFR 50.65, “Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants,” are sufficient for the NRC staff to pursue additional improvements on a plant-specific basis, if needed. Thus, this issue was resolved with no new or revised requirements.

References

19B.2.22-1 NUREG-0933, “A Prioritization of Generic Safety Issues (with Supplements 1-15)”, U.S. NRC, April 1993.

19B.2.22-2 Memo from Robert Kirkwood to Robert L. Baer, Engineering Issues Branch, Division of Safety Issue Resolution, Office of Nuclear Regulatory Research, dated on September 2, 1992.

19B.2.23 B-56: Diesel Reliability

Issue

Issue B-56 in NUREG-0933 (Reference 19B.2.23-1), addresses emergency diesel generator reliability. The reliability goal identified in NSAC-108, (Reference 19B.2.23-2) for emergency diesel generator startup, is between 0.95 and 0.975 per demand.

Typical onsite electrical distribution systems for plants use diesel generators as an emergency source of power. These emergency power sources supply safety-related equipment, which is used to prevent or mitigate accidents, in the event of a loss of offsite power.

Because of the safety significance of the emergency diesel generators, limiting conditions for operation (LCOs) were developed and placed in the plant technical specifications. These LCOs require periodic testing. Licensee Event Reports (LERs) sent to the NRC document problems encountered during periodic testing of the emergency diesel generators (to demonstrate operability). As discussed in NUREG-0933, a review of the LERs conducted by the NRC revealed that a diesel generator's starting reliability is, on the average, about 0.94 per demand. Thus, the NRC determined that there was a need to upgrade the reliability of emergency diesel generators. A new reliability of between 0.95 and 0.975 per demand for emergency diesel generator design, operation and periodic testing, was established in Regulatory Guide 1.9, Revision 3 (Reference 19B.2.23-3).

The specific emergency diesel generator starting reliability identified in Regulatory Guide 1.155 (Reference 19B.2.23-4) is the same as in Regulatory Guide 1.9, Revision 3 (i.e., it ranges from 0.95 to 0.975 per demand). The resolution of a related Issue A-44, Station Blackout, addresses the plant response to station blackout conditions.

Acceptance Criteria

The acceptance criteria for the resolution of Issue B-56, is that emergency diesel generator design, operation, and periodic testing shall ensure, as a minimum, a starting reliability of 0.95 per demand, as identified in Regulatory Guides 1.9, Revision 3, 1.155 and 1.160.

Resolution

The ABWR Standard Plant design includes an onsite electrical distribution system which employs three redundant and independent Class 1E load group divisions. The Class 1E safety loads are capable of being supplied power, in decreasing priority, from the unit main turbine generator, either of two offsite power sources, the emergency diesel generators (DGs), and the combustion turbine generator (CTG) (Figure 8.3-1).

Each of the three Class 1E divisions can be supplied with emergency standby power from an independent DG. The DG is designed and sized with sufficient capacity to operate all the needed Class 1E loads powered from its respective Class 1E divisional bus. Furthermore, each division can be manually supplied from the non-Class 1E CTG, which is diverse from the DGs. The reliability of the CTG is comparable to that of the DG (Section 9.5.11).

Each DG is specified to start reliably and, with present technology, industry experience has shown that a starting reliability of 0.986 per demand may be achieved as identified in the EPRI ALWR Utility Requirements Document (Reference 19B.2.23-5). The time required for the DG to attain rated voltage and frequency, and to begin accepting load, has been eased from 13 to 20 seconds after receipt of a start signal. This reduces their starting stress and contributes to improved reliability over the life of the units. The extended time is still within the limiting case for opening of the RHR valves [Subsection 8.3.1.1.8.2(4)].

A variety of tests are performed to assure DG reliability and operability. In addition to factory tests, a number of pre-operational and onsite acceptance tests and periodic tests are conducted on each DG system. These tests are identified in Subsection 8.3.1.1.8.2, and in the technical specifications. Also, conditions for operation are imposed to ensure continual reliability.

In summary, the ABWR Standard Plant design utilizes three independent diesel generators as emergency power sources, which are incorporated in the onsite electrical distribution system, and which have a diverse backup (i.e., the CTG).

The onsite electrical distribution system meets the intent of the guidance given in Regulatory Guides 1.9, Revision 3, 1.155 and 1.160.

Therefore, this issue is resolved for the ABWR Standard Plant design.

References

19B.2.23-1 NUREG-0933, "Resolution of Generic Safety Issues (with Supplements 1-33)", U.S. NRC, August 2010.

19B.2.23-2 NSAC-108, "Reliability of Emergency Diesel Generators at U.S. Nuclear Plants", Electric Power Research Institute, September 1986.

19B.2.23-3 Regulatory Guide 1.9, Revision 3, "Selection, Design, Qualification, Testing, and Reliability of Diesel Generator Units Used as Onsite Electrical Power Systems at Nuclear Power Plants", U.S. NRC.

19B.2.23-4 Regulatory Guide 1.155, "Station Blackout", U.S. NRC.

19B.2.23-5 "Advanced Light Water Reactor Utility Requirements Document", Volume II, Chapter 11; EPRI, April 1989.

19B.2.23-6 Regulatory Guide 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants", U.S. NRC.

19B.2.24 B-61: Allowable ECCS Equipment Outage Periods

Issue

Issue B-61 in NUREG-0933 (Reference 19B.2.24-1) addresses the potential for an overall reduction in the core damage frequency of a plant by reducing the frequency of surveillance testing and reducing permissible outage times for safety-related ECCS equipment.

Historically, ECCS equipment outage times and surveillance testing were not established by analysis. Instead, these test requirements were developed using engineering judgment and equipment operating, performance testing, and maintenance histories. After development, these test requirements were incorporated into the plant Technical Specifications as Limiting Conditions for Operation (LCOs).

Studies performed for the NRC on operating reactors indicate that from 30 to 80 percent of the ECCS system unavailability was due to testing, maintenance and allowed outage periods. The NRC is evaluating whether overall ECCS unavailability and resulting core damage frequency can be reduced by extending the intervals between testing and maintenance of equipment. Intervals can be extended within a range in which equipment unavailability due to testing and maintenance is reduced more than the predicted equipment unavailability due to failure is increased. Probabilistic risk assessment (PRA) methods would be used to determine the optimum intervals between ECCS equipment tests. Surveillance intervals optimized in this manner would then be used in LCOs (See Subsection 6.3).

As a part of this program a computer code (References 19B.2.24-2 and 19B.2.24-3) has been developed for the time dependent unavailability analysis. This code, using generic data from the Interim and National Reliability Evaluation Programs (IREP and NREP, respectively), will be used to verify the capability of the code to determine optimum surveillance intervals and resulting overall risk reduction. The costs and benefits can then be assessed.

Because the NRC evaluation of this issue has not yet been completed the initial LCOs for a future plant design may continue to be based on current industry practice without prejudicing later optimization when the methods and requirements have been confirmed. The overall plant PRA should take the initial LCOs into account, to establish a base against which to measure the effects of later optimization.

Acceptance Criteria

The acceptance criterion for the resolution of Issue B-61 for future plant designs is that the Technical Specification LCOs surveillance periods and allowable completion times of ECCS equipment shall be developed in accordance with current industry practice.

The LCOs surveillance periods and completion times shall be accounted for in the overall plant PRA required by 10 CFR 52.47 (Reference 19B.2.24-4). Any subsequent proposed changes to the LCOs' provisions for ECCS surveillance shall be demonstrated to be within the results of an existing PRA (see Section 6.3).

Resolution

The ABWR design incorporates many design enhancements to improve the operation and safety of the plant, and the most significant advances are in the area of engineered safety features. The ECCS conforms to all licensing requirements and good design practices of isolation, separation and common mode failure considerations.

In order to meet the above requirements, the ECCS network has built-in redundancy so that adequate core cooling can be provided, even in the event of specific failures. Each system of ECCS, including flow rate and sensing networks, is capable of being tested during plant operation, including logic required to automatically initiate component action. Provisions for testing the ECCS network components (electrical, mechanical, hydraulic and pneumatic, as applicable) are installed in such a manner that they are an integral part of the design.

The PRA uses a system fault tree approach to quantify system accident sequences which result in severe core damage. Data related to the engineered safety features used in the quantification includes:

(1) Component failure rates

(2) Component repair times and maintenance frequencies

(3) Component inspection and test times and frequencies

(4) Allowable equipment outage times

The data is used in accordance with the guidance in NUREG/CR-2815 (Reference 19B.2.24-5), and basic failure rate data used in the original PRA was obtained from the ERPI ALWR Requirements Document (Reference 19B.2.24-6) supplemented with other nuclear sources. The generic data that was updated is discussed in Subsection 19D.3.2. Maintenance and repair times are calculated as outlined in NUREG/CR-2815. The inspection and test times and frequencies are as specified in ABWR STS Section 3.5. Several LCO completion times for the ABWR were determined based on relative comparisons of PRA estimated frequencies for conditional core damage.

The PRA demonstrates that the ABWR design meets the industry goal of 1.0 x E-5 core damage frequency per reactor year (Reference 19B.2.24-6) and indicates that the initial LCOs are consistent with this goal. The owner-operator may refine the LCOs to achieve further risk reduction or increased operational flexibility provided that the resulting overall risk is shown to be within the results of the PRA.

This issue is, therefore, resolved for the ABWR.

References

19B.2.24-1 NUREG-0933, "A Prioritization of Generic Safety Issues (with Supplements 1-15)", U.S. NRC, April 1993.

19B.2.24-2 NUREG-0193, “FRANTIC – A Computer Code for Time Dependent Unavailability Analysis”, U.S. NRC, October 1977.

19B.2.24-3 NUREG/CR-1924, “FRANTIC II – A Computer Code for Time Dependent Unavailability Analysis”, U.S. NRC, April 1981.

19B.2.24-4 10 CFR 52, “Early Site Permits; Standard Design Certification; and Combined Licenses for Nuclear Power Reactors”, Office of the Federal Register, National Archives and Records Administration.

19B.2.24-5 NUREG/CR-2815, “Probabilistic Safety Analysis Procedures Guide”, Brookhaven National Laboratory, January 1984.

19B.2.24-6 “Advanced Light Water Reactor Utility Requirements Document – Volume II, Chapter 1: Overall Requirements, Appendix A: PRA Key Assumptions and Groundrules”, Electric Power Research Institute, Draft, April 1987.

19B.2.25 B-63: Isolation of Low Pressure Systems Connected to the Reactor Coolant Pressure Boundary

Issue

Issue B-63 in NUREG-0933 (Reference 19B.2.25-1) addresses the need to ensure the integrity (i.e., leak-tightness) of boundary valves installed between high pressure (HP) (i.e., the Reactor Coolant System pressure boundary) and low pressure (LP) safety-related systems, during plant operation by performing periodic in-service testing.

The ASME Code, Section III, (Reference 19B.2.25-3) controls the design, fabrication, and initial testing of boundary and relief valves. During operation, the ASME Code, Section XI, specifies boundary and relief valve testing requirements to assure continued valve integrity.

Because of the importance of the HP to LP interface for safety-related systems, the NRC reviewed and updated SRP Section 3.9.6 by issuing Revision 2 (Reference 19B.2.25-2). This SRP references and endorses the ASME Code, Section XI (for the in-service testing of the boundary valves).

(A related issue, which also discusses the integrity of the HP to LP interface between safety-related systems, is Issue 105, “Interfacing Systems LOCA.”)

Acceptance Criteria

The acceptance criteria for the resolution of Issue B-63 is that the periodic inservice testing of the HP to LP system boundary valves shall meet the intent of SRP Section 3.9.6, Revision 2. Because SRP 3.9.6, Revision 2, endorses the requirements of the ASME Code, Section XI, periodic testing of these valves shall be performed in accordance with the code.

Specifically, these boundary valves shall comply with the requirements of the applicable IWV subarticles identified within Section XI of the ASME Code. This compliance shall include the appropriate classification and/or categorization of safety-related valves and the development of the proper test procedures for pre-operational and periodic inservice valve testing.

Resolution

All pressure containing components including all high pressure to low pressure safety-related system boundary valves used in the Advanced Boiling Water Reactor (ABWR) Standard Plant design are identified as Safety Class 1, 2, or 3, and are designed, manufactured, and tested in accordance with the guidelines of the ASME Code, Section III. (See Subsections 3.2.1, 3.2.2, and 3.2.3 for Seismic Classification, Quality Group Classifications, and Safety Classifications, respectively. Table 3.2-1 provides a cross-reference between safety and code classifications.)

Boundary valves will be periodically inservice tested in accordance with the provisions of ASME Code, Section XI, to assure operational integrity as well as to Subsection IWV requirements for each valve category. Code Class 1, 2, and 3 valves will be categorized according to Subarticle IWV-2100. Valve test requirements and valve performance testing frequency are listed in the Subsections 3.9.6, 3.9.6.2, 3.9.6.2.1, 3.9.6.2.2, and 3.9.6.2.3.

In summary, the High Pressure and Low Pressure system boundary interface valves are designed, manufactured, pre-operational tested, and in-service tested according to the guidelines of the ASME Code and satisfy the intent of SRP Section 3.9.6, Revision 2.

Therefore, Generic Safety Issue B-63 is resolved for the ABWR design.

References

19B.2.25-1 NUREG-0933, “A Prioritization of Generic Safety Issues (with Supplements 1-15)”, U.S. NRC, April 1993.

19B.2.25-2 NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants—LWR Edition”, U.S. NRC.

19B.2.25-3 “ASME Boiler and Pressure Vessel Code”, Sections III (Nuclear) and XI, American Society of Mechanical Engineers.

19B.2.26 B-66: Control Room Infiltration Measurements

Issue

Issue B-66 in NUREG-0933 (Reference 19B.2.26-1) addresses maintenance of the control room in a safe habitable condition under accident conditions by providing adequate protection for the plant operators against airborne radiation and toxic gases.

The rate of air infiltration into the control room is a significant factor in maintaining habitability, and the NRC measured air exchange rates in selected operating reactor plant control rooms to improve the data base for evaluating its effects.

No new design requirements were established by the NRC as a result of this and other work related to control room habitability in an accident. However, more specific review procedures were incorporated in SRP Sections 6.4.1, 9.4.1 and 15.6.5.5 (Reference 19B.2.26-2), including the habitability review provisions of TMI Action Plan Item III.D.3.4 (Reference 19B.2.26-1) regarding analyses of toxic gas concentrations and operator exposures from airborne radioactive material and direct radiation, to ensure more effective implementation of existing requirements.

Acceptance Criteria

The acceptance criteria for the resolution of Issue B-66 is that the control room ventilation and air-conditioning systems be designed to maintain the room's environment within acceptable limits for the operation, testing and maintenance of the unit controls and for uninterrupted safe occupancy during normal and accident conditions. Specifically, these systems shall be designed to meet the intent of the guidance given in SRP, Sections 6.4, 6.5.1, 9.4.1 and 15.6.5.5.

Resolution

The ABWR main control area envelope is heated, cooled, ventilated and pressurized with respect to the atmosphere and adjacent areas are maintained at positive pressure with respect to the atmosphere by a system mixing recirculated air with filtered outdoor air. It is designed to ensure that the operators can remain in the main control area envelope and take actions to operate the plant safely under normal conditions and maintain it in a safe condition during and following an accident. There are two air intakes on the top floor side walls of the control building, one on each end. Redundant radiation monitoring sensors in each air intake warn operators of airborne contamination, and cause the CRHA HVAC system to switch automatically to an emergency system employing HEPA and charcoal filters for cleanup.

This control room habitability area heating, ventilating and air-conditioning (CRHA HVAC) system is designed:

- With redundancy to ensure operation in an emergency with a single, active failure;
- For radiation exposure limits not exceeding the guidelines of 10 CFR 50, Appendix A, General Design Criterion 19 (Reference 19B.2.26-3), for any of the Chapter 15 DBAs;
- With provisions to detect and remove smoke and airborne radioactive material;
- To provide a controlled temperature and pressurized environment for continued operation of safety-related equipment under accident conditions;
- Protection from toxic chemical and chlorine releases.

In addition, the safety-related components of the CRHA HVAC system are operable during loss of offsite power conditions using divisional onsite power from the diesel generators and safety-related batteries. Provisions are also made for periodic tests of the emergency filtration unit fans and filters. The high-efficiency particulate air (HEPA) filters of the CRHA HVAC system will be tested periodically with dioctyl phthalate (DOP) smoke. The charcoal filters will be periodically tested with an acceptable gas for bypasses. The system ductwork and housings, which are of welded construction, will be periodically tested for unfiltered inleakage in accordance with ASME N510.

This ABWR CRHA HVAC and its design bases are described in Section 6.4, 6.5.1 and Subsection 9.4.1.

Since the control room is monitored, pressurized and filtered by the above described systems, and since the NRC requirements and the guidance for their design are met, the issue of air infiltration is resolved for the ABWR.

References

19B.2.26-1 NUREG-0933, “A Prioritization of Generic Safety Issues (with Supplements 1-15)”, U.S. NRC, April 1993.

19B.2.26-2 NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants—LWR Edition”, U.S.NRC.

19B.2.26-3 10 CFR 50 Appendix A, “General Design Criteria for Nuclear Power Plants”, Office of the Federal Register, National Archives Records Administration.

19B.2.27 C-1: Assurance of Continuous Long-Term Capability of Hermetic Seals on Instrumentation and Electrical Equipment

Issue

Item C-1 in NUREG-0933 (Reference 19B.2.27-1), addresses concerns regarding the long-term capability of hermetically-sealed instruments and equipment which must function in post-accident environments. NUREG-0471 (Reference 19B.2.27-2) was developed because of these concerns.

Certain classes of instrumentation incorporate seals. When safety-related components within containment must function during post-LOCA accident conditions, their operability is sensitive to the ingress of steam or water. If the seals should become defective as a result of personnel errors in the maintenance of such equipment, such errors could lead to the loss of effective seals and the resultant loss of equipment operability. The establishment of a basis for confidence that sensitive equipment has a seal during the lifetime of the plant is needed.

Acceptance Criteria

The NRC has undertaken a program to reevaluate the qualification of all safety-related electrical equipment at all operating reactors. As part of this program, more definitive criteria for environmental qualification of safety-related electrical equipment have been developed by the staff. The Division of Operating Reactors’ “Guidelines for Evaluating Environmental Qualification of Class 1E Electrical Equipment in Operating Reactors” (DOR Guidelines) were completed in November 1979. The Guidelines are intended as a screening device to catch those pieces of equipment which might have qualification problems. In addition, for reactors under licensing review, the staff has issued NUREG-0588 (Reference 19B.2.27-3). The staff intends to evaluate the qualification of all electrical safety equipment in operating plants pursuant to the Guidelines. If problems arise, the staff shall resolve them using NUREG-0588 as a guide for their judgment.

On May 27, 1980 the NRC issued Commission Memorandum and Order CLI-80-21 (Reference 19B.2.27-4) ordering that the above two documents form the requirements which licensees and applicants must meet in order to satisfy those aspects of 10 CFR 50, Appendix A, General Design Criterion (GDC) 4, (Reference 19B.2.27-6) which relate to the environmental qualification of safety-related electrical equipment. The order established an implementation schedule which set a goal that all safety-related electrical equipment in all operating plants be qualified to the DOR Guidelines or NUREG-0588 by no later than June 30, 1982.

Resolution

Environmental qualification of safety-related equipment is described in Section 3.11.

Safety-related equipment located in a harsh environment must perform its proper safety function during normal, abnormal, test, design basis accident and post-accident environments as applicable. A list of all safety-related electrical and mechanical equipment that is located in a harsh environment area will be included in the Environmental Qualification Document (EQD) to be prepared as indicated in Subsection 3.11.6.1.

Environmental conditions for the zones where safety-related equipment is located are calculated for normal, abnormal, test, accident and post-accident conditions and are documented in Appendix 3I. Environmental conditions are tabulated by zones, contained in the referenced building arrangements.

Safety-related electrical equipment that is located in a harsh environment is qualified by test or other methods as described in IEEE 323 (Reference 19B.2.27-5) and permitted by 10 CFR 50.49(f), “Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants “(Reference 19B.2.27-6).

The qualification methodology is described in detail in the NRC approved Licensing Topical Report on GE’s environmental qualification program (Reference 19B.2.27-7). This report also addresses compliance with the applicable portions of the General Design Criteria of 10 CFR 50, Appendix A, and the Quality Assurance Criteria of 10 CFR 50, Appendix B. Additionally, the report describes conformance to NUREG-0588, and Regulatory Guides and IEEE Standards referenced in Section 3.11 of NUREG-0800 (Reference 19B.2.27-8). The COL applicant will address issues identified in the Generic Safety Issue C-1 and provide a list of impacted safety-related components whose operabilities are sensitive to the ingress of steam or water in a harsh environment within containment. The COL applicant will also evaluate the provision of design features for assuring long-term capabilities of these components in post-accident environments based on NUREG-0588, as required in Subsection 3.11.6.

In summary, the safety-related electrical equipment is qualified in accordance with NRC Guidance, including NUREG-0588; and, therefore, this item is resolved for the ABWR Standard Plant design.

References

19B.2.27-1 NUREG-0933, "A Prioritization of Generic Safety Issues (with Supplements 1-15)", U.S. NRC, April 1993.

19B.2.27-2 NUREG-0471, "Generic Task Problem Descriptions (Categories B, C, and D)", U.S. NRC, June 1978.

19B.2.27-3 NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment", U.S. NRC, July 1981.

19B.2.27-4 NRC Memorandum and Order CLI-80-21, docketed May 27, 1980.

19B.2.27-5 IEEE Standard 323, "Qualifying Class 1E Equipment for Nuclear Power Generating Stations", The Institute of Electrical and Electronic Engineers, Inc.

19B.2.27-6 10 CFR 50, "Domestic Licensing of Production and Utilization Facilities", Code of Federal Regulations, Office of the Federal Register, National Archives and Records Administration.

19B.2.27-7 NEDE-24326-1-P, "General Electric Environmental Qualification Program," Proprietary Document, January 1983.

19B.2.27-8 NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants—LWR Edition", U.S.NRC.

19B.2.28 C-10: Effective Operation of Containment Sprays in a LOCA

Issue

Issue C-10 in NUREG-0933 (Reference 19B.2.28-1) is concerned with the effectiveness of various containment spray solutions in removing airborne radioactive materials present in the containment after a loss-of-coolant accident (LOCA). Also of concern is the possible damage to equipment in the containment caused by the solutions in an inadvertent actuation of the spray system.

After the TMI accident it became evident that previous regulatory assumptions as to the forms and timing of the release of radioactive iodine in an accident causing fuel damage were probably unduly conservative. The NRC and industry therefore reviewed experimental data and industry practice with regard to controlling the pH of spray solutions, which have to be borated to prevent boron dilution of reactor coolant, so as to ensure removal of radioactive iodine and particulates from the containment atmosphere and also to minimize corrosion in the safeguards systems during subsequent long term cooling. Some additives commonly used for pH control also have the potential to damage containment equipment if the spray system is unintentionally actuated, and make the resulting cleanup effort more difficult.

It was concluded that during the initial stage of an accident the removal efficiency of containment spray containing no dissolved iodine is essentially independent of the pH (for pH values less than 6.5) of the spray solution, but that while recirculating containment spray after the initial stage of the accident it is desirable to maintain the pH of the containment sump solution high enough to prevent re-release of absorbed iodine. Also at this time, as previously discussed in Branch Technical Position (BTP) MTEB 6-1 attached to Revision 2 of SRP Section 6.1.1 (Reference 19B.2.28-2), the pH should be high enough to preclude stress corrosion cracking of austenitic stainless steel materials used in emergency safeguards systems. The NRC therefore issued Revision 2 of SRP Section 6.5.2 (Reference 19B.2.28-2). This revision endorses the industry standard ANSI/ANS 56.5-1979, "PWR and BWR Containment Spray System Design Criteria" (Reference 19B.2.28-3), with the proviso that the standard's requirements for spray solution pH control need not be followed.

Acceptance Criteria

The acceptance criteria for the resolution of Issue C -10 is that the containment spray system shall be designed to meet the requirements of General Design Criteria 41, 42 and 43 (Reference 19B.2.28-4) related to fission product removal, periodic inspection, and functional testing, respectively, by conforming to the guidance of SRP Section 6.5.2, Revision 2. Specifically, the system design shall consider the appropriate criteria of ANSI/ANS 56.5-1979 except that the requirements of this standard for any spray additive or other pH control system need not be followed. The design shall minimize the probability of inadvertent actuation of the system and of consequent damage to equipment in the containment. The aqueous solution collected in the containment sump after completion of ECCS injection shall be maintained at an equilibrium pH of no less than 7.0 for long-term iodine retention and the protection of austenitic stainless steel materials from stress corrosion cracking in accordance with the guidance of BTP MTEB 6-1. Pre-operational tests of the containment spray system shall be specified to demonstrate that it meets the design requirements for an effective fission product scrubbing function, and technical specifications shall specify appropriate limiting conditions of operation.

Resolution

The Residual Heat Removal (RHR) system provides two independent containment spray cooling systems (on loops B and C) each having a common header in the wetwell and a common spray header in the drywell and sufficient capacity for containment depressurization by removing heat and condensing steam in both the drywell and wetwell air volumes following a LOCA. The drywell sprays also function to provide removal of fission products released during a LOCA as well as in the event of failure of the drywell head. The RHR system pumps water from the suppression pool, through the RHR heat exchangers into the wetwell and drywell spray spargers in the primary containment.

The drywell spray mode is initiated by operator action post-LOCA in the presence of high drywell pressure, and is terminated by operator action. Also, drywell spray is terminated automatically as the RHR injection valve starts to open, (which results from a LOCA and reactor depressurization). The wetwell spray mode is initiated by operator action, and is terminated automatically by a LOCA or terminated by operator action.

The water in the 304L stainless-steel-lined suppression pool is maintained at high purity (low corrosion attack) by the Suppression Pool Cleanup (SPCU) System. In the event of a LOCA, the SPCU function is automatically terminated to accomplish containment isolation. The pH range (5.3-8.9) is maintained to minimize any corrosive attack on the pool liner (304L SS) over the life of the plant. The post-LOCA aqueous phase pH in all areas of containment will have a flat time history (i.e., the liquid coolant will remain at its design basis pH throughout the event). The use of organic coatings within the containment has been kept to a minimum. The major use of such coatings is on the carbon steel containment liner, internal steel structures and equipment inside the drywell and wetwell. The epoxy coatings are specified to meet the requirements of Regulatory Guide 1.54 and are qualified using the standard ANSI tests, including ANSI N101.4. All safety-related equipment in the containment is environmentally qualified, and protected against spray actuation (Section 3.11).

The system design adheres to the appropriate criteria guidelines of ANSI/ANS 56.5-1979. Application of accepted human factors principles and methodologies to the RHR System instrumentation and controls design minimizes the possibility of inadvertent actuation as a result of operator error (Subsection 18.3.1). Pre-operational testing for operability is performed on the RHR Containment Spray Subsystem (Subsection 14.2.12.1.8). Technical Specifications/Limiting Conditions for Operation (LCOs) of the RHR Containment Spray Subsystem and the Primary Containment System are given in Chapter 16, Section 3.6.

It should be noted that credit is not taken for any fission product removal provided by the drywell and wetwell spray portions of the RHR system. The quantity of fission products released into the environment following postulated accidents is controlled by the standby gas treatment system (SGTS) that has the redundancy and capability to filter the gaseous effluent from the primary and the secondary containment.

The ABWR Design fulfills the requirements of General Design Criteria 41, 42, and 43 relating to fission product removal, periodic inspection, and functional testing by conforming to the criteria guidelines of SRP Section 6.5.2, Revision 2 (Subsections 3.1.2.4.12.2, 3.1.2.4.13.2, and 3.1.2.4.14.2).

In summary, the ABWR design meets the intent of the criteria guidelines of SRP Section 6.5.2, Revision 2, and BTP MTEB 6-1 in order to fulfill the function of reducing the concentration of radioactive iodine and particulates in the containment atmosphere during and after a LOCA, while also minimizing the probability of initiating stress corrosion cracking of stainless steel in the safeguard systems. Design features also minimize the probability of inadvertent actuation of the RHR Containment Spray subsystem or the SGTS, thus minimizing possible damage to safety-related equipment in the containment. Technical Specifications/LCOs are also provided.

Issue C-10 in NUREG-0933 is, therefore, resolved for the ABWR Standard Plant design.

References

19B.2.28-1 NUREG-0933, "A Prioritization of Generic Safety Issues (with Supplements 1-15)", U.S. NRC, April 1993.

19B.2.28-2 NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants—LWR Edition", U.S. NRC.

19B.2.28-3 ANSI/ANS 56.5, "PWR and BWR Containment Spray System Design Criteria", American National Standards Institute.

19B.2.28-4 10 CFR 50 Appendix A, "General Design Criteria for Nuclear Power Plants", Office of the Federal Register, National Archives and Records Administration.

19B.2.29 C-17: Interim Acceptance Criteria for Solidification Agents for Radioactive Solid Wastes

Issue

NUREG-0471 Item C-17 (Reference 19B.2.29-1) discusses the Interim Acceptance Criteria for Solidification agents for radioactive solid wastes.

Acceptance Criteria

The acceptance criteria for the resolution of C-17 is under development. This NUREG-0471 (Reference 19B.2.29-1) task involves the development of criteria for acceptability of radwaste solidification agents to properly implement a process control program for the packaging of diverse plant wastes for shallow land burial.

Resolution

10 CFR Part 61 was published in the Federal Register on December 27, 1982 (47 FR 57446) and includes Section 61.56 which addresses waste characteristic (Reference 19B.2.29-2). BTP ETSB 11-3 on waste form has been developed under TMI Action Plan Item IV.C.1. The ABWR is committed to meeting the requirements in 10 CFR Part 61, Reference 19B.2.29-3 (Subsection 11.4.1.2).

The COL applicant shall demonstrate that the wet waste solidification processes and the spent resin and sludge dewatering processes will result in products that comply with 10 CFR 61.56. A process control program (PCP) shall be provided for the processes employed.

This procedure will encourage the development and use of additional acceptable methods of solidifying radioactive waste solids in the future.

Thus, this item has been resolved for the ABWR.

References

19B.2.29-1 NUREG-0471, "Generic Task Problem Descriptions (Categories B, C, and D)", U.S. NRC, June 1978.

19B.2.29-2 Memorandum for T. Speis from J. Funches, "Prioritization of Generic Issues – Environmental and Licensing Improvements", February 24, 1983.

19B.2.29-3 10 CFR 61.56, "Licensing Requirements for Land Disposal of Radioactive Waste."

19B.2.30 15: Radiation Effects on Reactor Vessel Supports

Issue

Issue 15 in NUREG-0933 (Reference 19B.2.30-1), addresses the potential for failure of the reactor vessel support structure (RVSS) due to a combination of low temperature and neutron flux irradiation embrittlement.

Neutron irradiation of structural materials used in the RVSS causes embrittlement that may increase the potential for propagation of pre-existing cracks or flaws within these materials. The potential for brittle fracture of these materials is typically measured in terms of their nil ductility transition temperature (NDTT). As long as the operating environment in which a material is used has a temperature that is significantly higher than the NDTT of the material, no failure by brittle fracture would be expected. Many materials, when subjected to neutron irradiation, experience an upward shift in the NDTT, i.e., they become more susceptible to brittle fracture. This effect must be accounted for in the design and fabrication of the RVSS.

During 1988, new data was developed for the RVSS materials at Oak Ridge National Laboratory (ORNL) (References 19B.2.30-2 and 19B.2.30-3). This data indicated that neutron flux at low temperatures caused greater embrittlement of the materials used in the RVSS than previously anticipated. This increased material embrittlement or "upward shift" in NDTT reduces the fracture toughness of these materials and, under certain specific and conservative transient conditions such as an earthquake or large-break Loss of Coolant Accident, could conceivably result in the failure of the supports thus permitting the reactor vessel to move.

As a result of the ORNL work, the NRC re-prioritized this issue and reviewed its regulatory position relative to low temperature and neutron flux radiation embrittlement.

Acceptance Criteria

The acceptance criteria for the resolution of GSI 15 is that the material integrity for the RVSS shall be maintained for the design lifetime of the plant.

Specifically, the design of the reactor vessel supports shall address irradiation effects (including low temperature and neutron flux) and the attendant material embrittlement, and be designed to restrain the reactor vessel under the combined Safe Shutdown Earthquake and branch line pipe break loadings in accordance with the stress and deflection limits established in Section III of the ASME Code (Reference 19B.2.30-4).

Resolution

The RVSS for the ABWR is described in Subsections 5.3.3.1.4.1 and 3.9.1.4.2 and shown in Figure 5.3-2. The RVSS consists of a support skirt bolted to the support pedestal. The skirt is located below the core beltline and slightly below the core support plate. As such, the skirt is in a region of low neutron flux which is further reduced since the ABWR water flow region between the vessel shroud and vessel wall is almost 40cm wider than previous BWRs. Therefore, neutron embrittlement of the skirt is well below any current or potential future limitations. A bounding analysis of neutron flux in these regions is given in Subsection 5.3.3.1.4.7. The value in this analysis of 6×10^{17} neutron/cm^2 can be compared to the bounding expected value for the skirt welds of 3×10^{14} neutron/cm^2 for a 60 year exposure.

Reference 19B.2.30-1 concluded that the results of further industry analysis virtually eliminated the concern for both radiation embrittlement and significant structural damage from a postulated RPV failure. This issue was resolved with no new requirements.

References

19B.2.30-1 NUREG-0933, "Resolution of Generic Safety Issues (with Supplements 1-33)", U.S. NRC, August 2010.

19B.2.30-2 ORNL/TM-10444, "Evaluation of HFIR Pressure Vessel Integrity Considering Radiation Embrittlement", Oak Ridge National Laboratory, 1988.

19B.2.30-3 ORNL/TM-10966, "Impact of Radiation Embrittlement on the Integrity of Pressure Vessel Supports for Two PWR Plants", Oak Ridge National Laboratory, 1989.

19B.2.30-4 "ASME Boiler and Pressure Vessel Code", Section III (Nuclear), American Society of Mechanical Engineers.

19B.2.31 23: Reactor Coolant Pump Seal Failures

Issue

This issue deals with the high rate of Reactor Coolant Pump (RCP) seal failures that challenge the makeup capacity of the ECCS in PWRs. However, operating experience indicates that the leak test for major RCP seal failures in BWRs is smaller. The smaller leak rate, larger RCIC, HPCI, and feedwater makeup capabilities, and isolation valves on the RCP loops negate the potential problem in BWRs.

Acceptance Criteria

Not applicable. Issue does not apply to BWRs.

Resolution

The ABWR wet motor Reactor Internal Pumps (RIPs) as described in the ABWR Subsection 5.4.1 do not include seals. This feature is further described in ABWR Subsection 1A.2.30.

Therefore, Issue 23 is resolved for ABWR.

References

19B.2.31-1 NUREG-0933, “Resolution of Generic Safety Issues (with Supplements 1-33)”, U.S. NRC, August 2010.

19B.2.32 25: Automatic air Header Dump on BWR Scram System

Issue

This issue concerns the slow loss of control air pressure in the scram system of BWRs. Air pressure dropping at a certain rate will first allow some of the Control Rod Drive (CRD) scram outlet valves to open slightly, thus filling the scram discharge volume with water but allowing little or no control rod movement. Eventually, the rods will try to scram but the scram will be impaired in a manner similar to what happened at Browns Ferry Unit 3 on June 28, 1980 (Reference 19B.2.32-1). Meanwhile, the dropping air pressure can cause a transient (e.g., via feedwater controller lockup) which would normally call for a scram.

Acceptance Criteria

The acceptance criteria for this issue is specific to the scram discharge volume design and is not applicable to the ABWR. See the resolution discussion that follows.

Resolution

For the ABWR fine motion control rod drive (FMCRD) design, scram water is discharged through the drive directly into the reactor vessel. There is no scram discharge volume as used in previous BWR designs employing the locking piston control rod drive (LPCRD). Consequently, the common mode failure or impairment of scram associated with loss of control air pressure and filling of the scram discharge volume is not applicable to the ABWR.

The analogous concern for the ABWR design is that the slow loss of control air pressure in the scram air header can allow some of the scram accumulators to leak to the reactor. This action could deplete the accumulators’ charge and impair or prevent their capability to scram the connected control rods, unless specific design features are provided to prevent or mitigate its occurrence. The ABWR design does provide protection against this event by incorporating the following features:

(1) A scram air header low pressure alarm to alert the operator of a low pressure condition in the header (Figure 4.6-8, Sheet 2)—The setpoint value is chosen to be greater than the pressure at which the scram valves could start to open in order to allow the operator the opportunity to take corrective action.

(2) A rod block and alarm initiated by low pressure and a scram initiated by low-low pressure in the common header supplying the charging water to the scram accumulators—All the accumulators will have sufficient water volume to scram their associated control rods as long as the CRD System pump maintains the pressure in the charging header above the minimum required accumulator charging pressure, even if multiple scram valves are leaking. The pressure in the header will drop only if the total scram valve leakage flow is greater than the capability of the charging pump to provide make-up and maintain system pressure. If this should occur, instrumentation located in the charging header will sense the loss of pressure and signal the RCIS to initiate a rod block and alarm at a predetermined low pressure setpoint. If pressure degrades even further, it will signal the RPS to initiate an immediate scram at a predetermined low-low pressure setpoint. The low-low pressure setpoint value is based on the minimum accumulator charging pressure. This automatic feature protects the capability to safely shut down the plant by assuring that scram occurs while adequate accumulator charge is still available (Subsection 4.6.1.2.4.3).

In summary, the ABWR incorporates design features to prevent the loss or impairment of the scram function due to a slow loss of control air in the scram system. The first is a low pressure alarm to alert the operator to trouble in the scram air header; the second is an accumulator charging header low pressure scram to automatically shut down the plant before the accumulators are depleted.

Therefore, this issue is resolved for the ABWR design.

References

19B.2.32-1 "Report on the Browns Ferry 3 Partial Failure to Scram Event on June 28, 1980", U.S. NRC, July 30, 1980.

19B.2.33 40: Safety Concerns Associated with Pipe breaks in the BWR Scram System

Issue

If a break or leak exists or develops in the scram discharge volume (SDV) piping during a reactor scram, this would result in the release of water and steam at 373.15 K (212°F) into the reactor building at a maximum flow rate of 3.469E-2 m^3/s (550 gpm) and is postulated to result in 100% relative humidity in the reactor building. The principal means of isolating this break would be to close the scram exhaust valves which are located on the hydraulic control units; however, this is dependent upon the ability to reset scram, which cannot be absolutely ensured immediately following the scram. Therefore, a rupture of the SDV could result in the unisolable break outside of primary containment, which is postulated to threaten emergency core cooling equipment by flooding areas in which this equipment is located and by causing ambient temperature and relative humidity conditions for which this equipment is not qualified.

Acceptance Criteria

NUREG-0803 (Reference 19B.2.33-1) provides guidance to ensure SDV pipe integrity, detection capability, mitigation capability and qualification of the emergency equipment to the expected environment.

Resolution

For the ABWR fine motion control rod drive (FMCRD) design, scram water is discharged through the drive directly into the reactor vessel. There are no CRD withdraw lines or SDV as used in previous BWR designs employing the locking piston control rod drive (LPCRD). Consequently, the issue of SDV isolation provisions as addressed in NUREG-0803 (Reference 19B.2.33-1) is not applicable to the ABWR design.

In addition, for protection against a scram insert line break, the ABWR FMCRD design incorporates a ball-check valve located in the FMCRD flange housing at the point of connection of the insert line with the drive scram port. In the event of a rupture of the insert line, the ball-check valve will close to prevent reactor vessel flow out of the break. This feature is the same as used by the LPCRD in previous BWR designs.

For these reasons, this issue is resolved for the ABWR design.

References

19B.2.33-1 NUREG-0803, "Generic Safety Evaluation Report Regarding Integrity of BWR Scram System Piping", U.S. NRC, August 1981.

19B.2.34 45: Inoperability of Instrumentation Due to Extreme Cold Weather

Issue

Generic Safety Issue (GSI) 45 in NUREG-0933 (Reference 19B.2.34-1), addresses the potential for safety-related equipment instrument lines to become inoperable as a result of freezing or reaching the precipitation (i.e., solidification) point of the sensing fluids.

Typical safety-related systems employ pressure and level sensors which use small bore instrumentation lines. Most operating plants contain safety-related equipment and systems, parts of which are exposed to the ambient environment. These lines generally contain liquid (e.g., borated water) which is susceptible to freezing. Where systems or components and their associated instrumentation are exposed to sub-freezing temperatures, heat tracing and/or insulation is used to minimize the effects of cold temperatures.

These sensing and instrumentation lines are of concern because, should they freeze, they may prevent a safety-related system or component from performing its safety function. For example, an incident occurred at a plant wherein the heat tracing system surrounding sensing lines and level transmitters for the Refueling Water Storage Tank (RWST) failed during sub-freezing weather. The failure of the heat tracing systems resulted in freezing of the sensing lines and associated level transmitters causing a loss of all four RWST instrumentation channels, which could have resulted in the failure of the Emergency Core Cooling System, thus jeopardizing plant safety.

Because of the possibility of a safety-related system failure, the NRC issued additional guidance given in Regulatory Guide 1.151 (Reference 19B.2.34-2) to supplement the existing guidance and requirements which include the Standard Review Plan (SRP) Section 7.1, 10 CFR 50, Appendix A, and industry standard ISA-S67.02 (References 19B.2.3-2, 19B.2.34-4, and 19B.2.34-5, respectively). Regulatory Guide 1.151 specifically addresses the prevention of safety-related instrument sensing line freezing and includes design issues such as diversity, independence, monitoring and alarms.

Acceptance Criteria

The acceptance criterion for the resolution of GSI 45 is that the fluid in safety-related equipment instrument sensing lines shall be protected from freezing and maintained above the precipitation point.

The protection of safety-related equipment instrument sensing lines from freezing can be accomplished by providing environmental control systems which meet the requirements of 10 CFR 50, Appendix A (GDCs); industry standard ISA-S67.02; the intent of Regulatory Guide 1.151; and SRP Sections 7.1 (Revision 3), 7.1, Appendix A (Revision 1), 7.5 (Revision 3), and 7.7 (Revision 3).

Also, should environmental control prove to be limited, alternative forms of sensing line protection such as heat tracing and/or insulation can be used. (The use of heat tracing and/or insulation is not anticipated for the ABWR Standard Plant design; however, it is an acceptable alternate to environmental control.)

Resolution

The ABWR Standard Plant incorporates instrument sensing lines in safety-related systems and components. All safety-related systems and components used in the ABWR Standard Plant design, including instrument sensing lines, are located in temperature controlled environments which are maintained above the freezing (or precipitation) point of the contained fluid. The temperatures of these environments are not expected to be less than 283 K (10°C), as shown in Appendix 3I. In addition, the ABWR is committed to meet the requirements of Regulatory Guide 1.151 (Table 1.8-20), which endorses and augments ISA-S67.02.

Therefore, this issue is resolved for the ABWR Standard Plant design.

References

19B.2.34-1 NUREG-0933, "A Prioritization of Generic Safety Issues (with Supplements 1-15)", U.S. NRC, April 1993.

19B.2.34-2 Regulatory Guide 1.151, "Instrument Sensing Lines", U.S. NRC.

19B.2.34-3 NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants—LWR Edition", U.S. NRC.

19B.2.34-4 10 CFR 50 Appendix A, "General Design Criteria for Nuclear Power Plants", Code of Federal Regulations, Office of the Federal Register, National Archives and Records Administration.

19B.2.34-5 ISA-S67.02, "Nuclear-Safety-Related Instrument Sensing Line Piping and Tubing Standards for Use in Nuclear Power Plants", Instrument Society of America.

19B.2.35 51: Proposed Requirements for Improving the Reliability of Open Cycle Service Water Systems

Issue

Issue 51 in NUREG-0933 (Reference 19B.2.35-1), identifies the susceptibility of the Station Service Water System (SSWS) to fouling which leads to plant shutdowns and reduced power operation for repairs.

The SSWS cools the Component Cooling Water System (CCWS) through the Component Cooling Water Heat Exchangers and rejects the heat to the ultimate heat sink (UHS) during normal, transient, and accident conditions. The CCWS in turn provides cooling water to those safety-related components necessary to achieve a safe reactor shutdown, as well as to various non-safety reactor auxiliary components.

Acceptance Criteria

Elimination of the possible effects of fouling of the service water system and ultimate heat sinks is a design goal of the ABWR. The COL applicant is given specific requirements and guidance on achieving this goal, including instruction to consider designs and new requirements which further mitigate the fouling effects. Additionally, the COL applicant is directed to investigate the problem with ice as a flow blockage mechanism and to dispose of and/or dissolve such ice as required.

Resolution

A review of operating plant experience shows that the most prevalent problems with plant cooling water systems are due to the corrosion and fouling caused by poor quality service water. In spite of a variety of water treatment schemes and use of expensive material, the wide range of harsh chemistry, silt and biological content result in a need for continuous maintenance of equipment. In order to make a significant operational improvement in this area, the ABWR requirements for plant cooling water systems will include the following (Reference 19B.2.35-2).

- Direct service water will not be used for component cooling. A closed loop component cooling water system will be utilized to transfer heat from the component heat loads via a heat exchanger to the service water system and ultimate heat sink. This minimizes the number of pieces of equipment which are in contact with the problem-causing service water and focuses the problem on the component cooling water heat exchanger.

- The COL applicant shall treat raw service water as needed to reduce the effect of mud, silt, or organisms.
- The COL applicant shall provide materials for piping, pumps, and heat exchangers that offer greater resistance to the range of probable water chemistry conditions.
- The COL applicant shall make provisions to facilitate the inspection of service water piping and replace sections of piping during plant life.

The COL applicant shall provide sufficient redundancy of makeup pumps for the ultimate heat sink so that makeup capabilities are not unduly reduced when one pump malfunctions. The need for a safety grade makeup shall be established in conjunction with establishing UHS water volume, as specified in Regulatory Guide 1.27 (Reference 19B.2.35-3).

Implementation of a baseline fouling program was issued to licensees in Generic Letter 89-13 (Reference 19B.2.35-4).

The COL applicant shall provide the safety related portions of these systems to meet the design bases during a loss of offsite power. These systems shall be designed to perform their cooling function assuming a single active failure in any mechanical or electrical system.

Therefore, this issue, 51, is resolved for ABWR.

References

19B.2.35-1 NUREG-0933, "A Prioritization of Generic Safety Issues (with Supplements 1-15)", U.S. NRC, April 1993.

19B.2.35-2 "Advanced Light Water Reactor Utility Requirement Document" (Volume II), EPRI.

19B.2.35-3 Regulatory Guide 1.27, "Ultimate Heat Sink for Nuclear Power Plants."

19B.2.35-4 Enclosure 1, Generic Letter 89-13, "Service Water System Problems Affecting Safety-Related Equipment."

19B.2.36 57: Effects of Fire Protection Systems Actuation on Safety-Related Equipment

Issue

Generic Safety Issue (GSI) 57 in NUREG-0933 (Reference 19B.2.36-1), addresses the potential for safety-related equipment to become inoperable because of water spray from the fire protection system. IE Information Notice 83-41 (Reference 19B.2.36-2) identified experiences in which actuation of fire suppression systems caused damage to safety-related equipment.

Acceptance Criteria

The acceptance criteria for the resolution of GSI 57, is that the fire protection system be designed to preclude damaging safety-related equipment and rendering the equipment inoperable. In addition, the fire protection system shall be designed to meet 10 CFR 50, Appendix A (GDC 3) (Reference 19B.2.36-3), which states in part:

> "Fire detection and fighting systems of appropriate capacity and capability shall be provided and designed to minimize the adverse effects of fires on structures, systems, and components important to safety. Fire fighting systems shall be designed to assure that their rupture or inadvertent operation does not significantly impair the safety capability of those structures, systems, and components."

Resolution

The ABWR is designed to prevent the inadvertent actuation of fire protection systems and to limit the effects of water spray onto safety-related equipment. The only safety-related equipment located in areas protected by automatic fire suppression systems are the emergency diesel generators and their associated auxiliary equipment. The automatic fire suppression systems protecting the safety-related equipment are of a highly reliable pre-action automatic sprinkler type. Actuation of these sprinklers requires the detection of a fire by infra-red and/or rate of heat detectors, and the opening of the fusible link sprinkler heads. Furthermore, each division has its own dedicated detection and actuation equipment for the control of the fire sprinklers in that divisional area. Two actuation signals are required to initiate the fire suppression system, the first of which will annunciate an alarm to alert the operator to any potential problems. In addition the operator has the capability of terminating the flow of fire suppressant locally by manual isolation valves.

In order to prevent damage to other equipment due to flooding from the discharge of a sprinkler system, equipment is elevated and floor drains are provided.

The basic physical layout of the ABWR and the selection of systems is such to enhance the tolerance of the ABWR plant to fire. The systems are designed and located such that there are three independent and physically safety-related divisions, any one of which is capable of bringing the plant to a safe shutdown in the event of a fire. For design purposes it is assumed that a fire in a division results in the immediate loss of function of the entire division. Even with this conservative assumption, the two remaining independent safety-related divisions are available for full utilization by the operator.

The ABWR is designed in accordance with 10 CFR 50 Appendix A (GDC 3), Reference 19B.2.36-3, to minimize the adverse effects of fire. Since the automatic fire protection systems are designed to preclude inadvertent actuation and in the event of an improbable inadvertent actuation the effects are limited to a single division, this issue is resolved for the ABWR (Section 9.5 and Appendix 9A).

References

19B.2.36-1 NUREG-0933, "Resolution of Generic Safety Issues (with Supplements 1-33)", U.S. NRC, August 2010.

19B.2.36-2 IE Information Notice 83-41, "Actuation of Fire Suppression System Causing Inoperability of Safety-Related Equipment", June 22, 1983.

19B.2.36-3 10 CFR 50 Appendix A, "General Design Criteria", Office of the Federal Register, National Archives and Records Administration.

19B.2.37 67.3.3: Improved Accident Monitoring

Issue

This Generic Safety Issue addresses the weaknesses in the accident monitoring of the type observed at the Ginna steam generator event (steam generator isolation, reactor coolant pumps trip, thermal shock from cold high pressure injection water). The weaknesses identified were:

(1) non-redundant monitoring of RCS pressure;

(2) failure of the position indication for the steam generator relief and safety valves; and

(3) the limited range of the charging pump flow indicator for monitoring charging flow during accidents.

These conditions make it more difficult for correct operator action in response to such events. Subsequently, the NRC Staff prepared and issued Regulatory Guide (RG) 1.97, Revision 2, which was implemented at Ginna (Reference 19B.2.37-1).

Acceptance Criteria

The acceptance criteria for the resolution of this item is based on the full implementation of the post-accident monitoring requirements of RG 1.97 (Reference 19B.2.37-3) and NUREG-0737 TMI Action Plans into the design of the ABWR.

Resolution

The ABWR has implemented into its basic design RG 1.97 requirements and the TMI action plan requirements of NUREG-0737 and NUREG-0737, Supplement 1 (Reference 19B.2.37-1). Refer to Subsections 7.5.1.1 and 18.2, and Table 7.5-2. The ABWR design is in full compliance with the latest issue of RG 1.97; and, this Issue, 67.3.3, is resolved for ABWR.

References

19B.2.37-1 NUREG-0737, "Clarification of TMI Action Plan Requirements", U.S. NRC, November 1980.

19B.2.37-2 NUREG-0933, "A Prioritization of Generic Safety Issues (with Supplements 1-15)", U.S. NRC, April 1993.

19B.2.37-3 Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident", U.S. NRC.

19B.2.38 75: Generic Implications of ATWS Events at Salem Nuclear Plant

Issue

On two occasions, Salem Unit 1 failed to scram automatically due to failure of both reactor trip breakers to open on receipt of an actuation signal. In both cases the unit was successfully tripped by manual action. The failure of the breakers has been attributed to excessive wear due to improper maintenance of the undervoltage relays which receive the trip signal from the protection system and cause mechanical action to open the breakers.

Failure to scram (also commonly referred to as anticipated transient without scram, ATWS) could result in unacceptable consequences (Reference 19B.2.38-1).

Acceptance Criteria

The acceptance criteria for the resolution of this issue is that:

- The plant must have a program for a post-trip review of unscheduled reactor shutdowns.
- The plant must have a program for safety-related equipment classification and vendor interface.
- The plant must have a program for post-maintenance operability testing.
- The plant must have a program to control vendor-related modifications, preventative maintenance and surveillance for reactor trip breakers.

These acceptance criteria are described in Generic Letter 83-28 (Reference 19B.2.38-2) and NUREG-1000 (Reference 19B.2.38-3).

Resolution

The reactor protection (trip) system (RPS) design provides the capability for the ABWR to satisfy the NRC requirements indicated in Generic Letter 83-28 and in NUREG-1000.

Execution of the programs in the Acceptance Criteria fall primarily into the phase of operations and maintenance that are the responsibility of the COL applicant. However, Section 3.2 provides the safety-related classification of principal components for the second criterion of the Acceptance Criteria.

Therefore, this issue, 75, is resolved for ABWR.

References

19B.2.38-1 NUREG-0933, "A Prioritization of Generic Safety Issues (with Supplements 1-15)", U.S. NRC, April 1993.

19B.2.38-2 Generic Letter No. 83-28, “Required Actions Based on Generic Implication of Salem ATWS Events”, July 8, 1983.

19B.2.38-3 NUREG-1000, “Generic Implications of ATWS Events at the Salem Nuclear Power Plant”, Volumes 1 & 2, April 1983, August 1983.

19B.2.39 78: Monitoring of Fatigue Transient Limits for Reactor Coolant System

Issue

Generic Safety Issue (GSI) 78 in NUREG-0933 (Reference 19B.2.39-1), addresses the concern that for a number of older Operating Plants, there are no Technical Specification (TS) requirements for monitoring the actual number of transient occurrences. In addition, environmental effects were not taken into account in the design bases for Reactor Coolant Pressure Boundary (RCPB) components. Environmental effects on fatigue resistance of material are not explicitly addressed in the ASME Code, Section III, (Reference 19B.2.39-2) Design Fatigue curves. Therefore, an assessment of the increase in Core Damage Frequency (CDF) due to environmental effects on fatigue resistance of material should be performed.

Acceptance Criteria

The acceptance criteria for the resolution of GSI 78 are that operating plants implement TS to monitor plant transients, and environmental effects on the fatigue life of ASME Code, Section III, Class 1 carbon steel piping be considered in accordance with Section 3.9.3.1.1.7.

Resolution

For the ABWR, Technical Specification 5.5.2.5 requires the monitoring of plant transients to ensure that RCPB components are maintained within their design limits. Environmental effects are included in the design bases for ABWR RCPB components. The calculated CDF includes the environmental effects on fatigue resistance of materials.

Therefore, this issue is resolved for the ABWR Standard Plant design.

References

19B.2.39-1 NUREG-0933, “Resolution of Generic Safety Issues (with Supplements 1-33)”, U.S. NRC, August 2010.

19B.2.39-2 “ASME Boiler and Pressure Vessel Code”, Section III (Nuclear), American Society of Mechanical Engineers.

19B.2.40 83: Control Room Habitability

Issue

Safety Issue 83 in NUREG-0933 (Reference 19B.2.40-1) is a concern over the loss of control room habitability following an accident release of external airborne toxic or radioactive material or smoke. Such a loss could impair or cause loss of the control room operators’ capability to safely control the reactor and could lead to a core damaging accident.

Acceptance Criteria

The acceptance criteria for the resolution of Issue 83 is to verify that the control room design is adequate to prevent the loss of habitability of the control room during an accident. The design must meet the guidance given in Standard Review Plan (SRP) Sections 6.4, 9.4.1, and 15.6.5.5 (Reference 19B.2.40-2). The design must be in accordance with 10 CFR 50, Appendix A, General Design Criteria (GDC) 2, 4, and 19 (Reference 19B.2.40-3); and ASME AG-1 and AG-1a (Reference 19B.2.40-5).

Resolution

The ABWR main control room habitability system is described in Subsection 9.4.1 and Section 6.4. The control room is a structure which is important to safety and is designed to withstand the effects of natural phenomena, missiles and postulated accidents in accordance with GDC 2 and 4. The design of the control room [and its heating, ventilation and air conditioning (HVAC) system] permits safe occupancy during abnormal conditions. Radiation exposure of control room habitability area personnel through the duration of any one of the postulated design basis accidents does not exceed the guidelines set by GDC 19, i.e., 50 mSv whole body radiation exposure. Smoke and toxic gas protection is provided as described in Subsection 6.4.4.2 by the use of non-combustible materials, purging by the HVAC, individual respirators, and site-specific considerations of potential chemical releases. The control room Engineered Safety Feature filter trains shall be designed, tested and maintained to comply with the applicable provisions of Regulatory Guide 1.52 (Reference 19B.2.40-4), as described in Subsection 9.4.1.1.7. Fire protection is provided by alarm systems, fire hoses and portable fire extinguishers (Subsections 9.5.1 and 9A.4.2). Testing and inspection requirements are identified in Subsection 6.4.5.

Since the control room design prevents the loss of control room habitability during accident conditions, and since all of the NRC requirements and guidance are met, this issue is resolved for the ABWR.

References

19B.2.40-1 NUREG-0933, "Resolution of Generic Safety Issues (with Supplements 1-33)", U.S. NRC, August 2010.

19B.2.40-2 NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants—LWR Edition”, U.S. NRC.

19B.2.40-3 10 CFR 50 Appendix A, “General Design Criteria for Nuclear Power Plants”, Office of the Federal Register, National Archives and Records Administration.

19B.2.40-4 Regulatory Guide 1.52, “Design, Testing, and Maintenance Criteria for Post-Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants.”

19B.2.40-5 ASME AG-1 “Code on Nuclear Air and Gas Treatment and ASME AG-1a” Addenda.

19B.2.41 86: Long Range Plan for Dealing with Stress Corrosion Cracking in BWR Piping

Issue

Issue 86 in NUREG-0933 (Reference 19B.2.41-1), addresses the past occurrences of intergranular stress corrosion cracking (IGSCC) in BWR recirculation loop piping and its impact on the integrity of the reactor coolant pressure boundary.

Cracking in large-diameter piping resulting from IGSCC could result in a loss-of-coolant accident.

Acceptance Criteria

The acceptance criteria for the resolution of Issue 86 are that IGSCC resistant materials and fabrication techniques to minimize sensitization shall be used. In addition, the ABWR water shall be maintained at the lowest practically achievable impurity levels. Furthermore, the material and fabrication techniques shall comply with the guidelines of NUREG-0313 (Reference 19B.2.41-2).

Resolution

For the ABWR, IGSCC resistance is achieved through the use of Type 316 stainless steel and compliance with the guidelines of NUREG-0313. All materials are supplied in the solution heat treated condition. During fabrication, any heating operations (except welding) between 800 K (427°C) and 1255 K (982°C) are avoided, unless followed by solution heat treatment. The ABWR water is maintained at the lowest practically achievable impurity levels to minimize its corrosion potential.

In summary, only stainless steel Type 316 material is used and the piping is fabricated, tested and installed in accordance with ASME Code, Section III, (Reference 19B.2.41-3) and NUREG-0313. Also, the owner-operator is required to comply with ASME Code, Section XI, (Reference 19B.2.41-3) for the performance of inservice inspection.

Therefore, this issue is resolved for the ABWR Standard Plant design.

References

19B.2.41-1 NUREG-0933, “A Prioritization of Generic Safety Issues (with Supplements 1-15)”, U.S. NRC, April 1993.

19B.2.41-2 NUREG-0313, “Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping”, U.S. NRC, July 1977, (Revision 1) July 1980, (Revision 2) January 1988.

19B.2.41-3 “ASME Boiler and Pressure Vessel Code”, Section III (Nuclear) and Section XI, American Society of Mechanical Engineers.

19B.2.42 87: Failure of HPCI Steam Line Without Isolation

Issue

This issue concerns a postulated break in the High Pressure Coolant Injection (HPCI) System steam supply line and the uncertainty regarding the operability of the HPCI steam supply line isolation valves under the postulated conditions (Reference 19B.2.42-1). A similar situation can occur in the Reactor Water Cleanup (CUW) System.

The HPCI steam supply line has two containment isolation valves (MOVs) in series: one on the inside and one on the outside of the containment. Both are normally open in order for the HPCI system to perform its function. The CUW also has two normally open containment isolation valves (MOVs) which must remain open if the system is to perform its function.

The operation of the valves is tested periodically without steam. Also, due to flow limitations at the valve manufacturer's facilities, only the opening characteristics are tested under operating conditions. Therefore, according to the NRC, the capability of the valves to close when exposed to the forces created by the flow resulting from a break downstream has not been demonstrated.

Furthermore, NRC sponsored testing has increased the concern over whether MOVs can reliably be expected to operate under design basis (i.e., pipe break) conditions.

Under a contract from the NRC, Idaho National Engineering Laboratory (INEL) conducted tests on six MOVs. The tests showed that all six valves required more force to open and close at the line break flow rates than was predicted. Two of the conditions tested were full guillotine breaks in the CUW and HPCI systems. These test results were reported at an NRC sponsored meeting on April 18, 1990 which prompted the NRC to issue Generic Letter 89-10 (Reference 19B.2.42-2).

Acceptance Criteria

The acceptance criteria for the resolution of Issue 87 is defined in Generic Letter 89-10 which requires adequately sized actuators for MOVs, verification of correct thrust and torque settings, and a program for testing, inspection and maintenance of MOVs under differential pressure, temperature and flow conditions so as to provide assurance that they will function when subjected to design basis conditions.

Resolution

The ABWR does not have an HPCI system. It does have an CUW system and a Reactor Core Isolation Cooling (RCIC) system which may fall under this issue.

The ABWR addresses the concerns and issues identified in GL 89-10 (specifically the method of assessment of the loads, the method of sizing the actuators, and the setting of the torque and limit switches) in Subsections 3.9.3.2 and 3.9.6.2.2.

Thus, compliance with GL 89-10 resolves Issue 87 for the ABWR design.

References

19B.2.42-1 NUREG-0933, "A Prioritization of Generic Safety Issues (with Supplements 1-15)", U.S. NRC, April 1993.

19B.2.42-2 Generic Letter No. 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance (with Supplements 1-4)", February 12, 1992.

19B.2.43 89: Stiff Pipe Clamps

Issue

Issue 89 in NUREG-0933 (Reference 19B.2.43-1), addresses the concern that for operating plants, the effects of stiff pipe clamps were assumed to be negligible and were not explicitly considered in the piping design. For some applications, there is a concern that certain piping system conditions coupled with specific stiff pipe clamp design requirements could result in interaction effects that should be evaluated in order to determine the significance of the induced pipe stresses.

The ASME Code, Section III, (Reference 19B.2.43-2) requires that the effects of attachments in producing thermal stresses, stress concentrations and restraints on pressure retaining members be taken into account in checking for compliance with stress criteria. Attachments to piping are generally categorized as integral and non-integral attachments. Lugs welded to the pipe wall are an example of integral attachments. Clamps used for attaching hangers, struts and snubbers to the pipe by bolting are non-integral attachments. Piping design reports specifically address local stresses at integral attachments, such as lugs. Any additional stresses induced in the pipe by non-integral, clamp bolted attachments, are not included in the piping design report.

Acceptance Criteria

The acceptance criteria for the resolution of Issue 89 is that the effect of stiff pipe clamps on piping stresses should be considered in the piping system design. For stiff pipe clamps installed on straight runs of pipe or on bends with a radius of at least five pipe diameters, the pipe clamp induced stresses can be considered negligible and explicit consideration is not required. This acceptance criteria is based upon analysis performed by GE.

In the 1980's, GE performed calculations for typical stiff pipe clamps used on BWR Main Steam and Recirculation piping systems. For each system, the stiff pipe clamps were installed on straight pipe or on bends with a radius of at least five pipe diameters. The purpose of these calculations was to evaluate the additional stresses at clamp locations due to the following:

(1) Differential thermal expansion of the pipe and clamp,

(2) Discontinuity stress in the pipe from internal pressure restraint,

(3) Thermal gradient through the pipe wall in the vicinity of the pipe clamp, and

(4) External loads due to dynamic events such as earthquake.

Maximum incremental primary stresses were less than 25% of the primary stress allowables, and maximum incremental secondary stresses were less than 40% of secondary stress allowables. The stresses at the clamp locations excluding clamp induced stresses were less than 30% of the ASME Code, Section III, allowables. The total primary and secondary stresses, including clamp induced stresses, were less than 70% of allowable stress.

The governing stress locations occurred at piping branch connections, elbows and shear lugs, they did not occur at stiff pipe clamp locations. The stress intensification that occurs at elbows, branch connections and shear lugs is much greater than that which occurs at stiff pipe clamps. Therefore, when the additional clamp induced stresses are included, the peak piping system stresses do not occur at the clamp locations. Based on these calculations, it was concluded that explicit consideration of clamp induced piping stresses is not required when the clamps are installed on straight pipe or on bends with a radius of at least five pipe diameters.

Resolution

For the ABWR, the following stiff pipe clamp parameters will be very similar to those for the BWR stiff pipe clamps evaluated in the calculations summarized above:

- Stiff pipe clamp geometry and material properties
- Pipe schedule and material properties
- Support rated loads less than or equal to 2.26E+05 N
- Piping system operating pressures and temperatures and operating transients
- Piping stresses at branch connections and elbows much greater than at stiff clamp locations

Therefore, it can be concluded that the governing ABWR piping stresses will not occur at stiff pipe clamp locations. For the ABWR, the piping design specifications shall require that stiff pipe clamps be installed on straight runs of pipe or on bends with a radius of at least five pipe diameters. The pipe clamp induced stresses for NSS piping can then be considered negligible and do not warrant explicit consideration. The piping design specifications shall require that if stiff clamps are used on other than NSS piping, the stresses they induce will be considered.

This issue is resolved for the ABWR.

References

19B.2.43-1 NUREG-0933, “Resolution of Generic Safety Issues (with Supplements 1-33)”, U.S. NRC, August 2010.

19B.2.43-2 “ASME Boiler and Pressure Vessel Code”, Section III (Nuclear) and Section XI, American Society of Mechanical Engineers.

19B.2.44 103: Design for Probable Maximum Precipitation

Issue

Issue 103 in NUREG-0933 (Reference 19B.2.44-1) addresses the accepted methodology used for determining the design flood level for a particular reactor plant site. Accurate determination of the design flood level for a specific reactor site is necessary in order to ensure adequate protection of safety-related equipment against possible site flooding.

Reactor plant sites are designed to accommodate maximum flood level because flooding could disable safety-related equipment. Historically estimating design flood levels for specific reactor plant sites has been based upon input data for probable maximum flood (PMF) provided by the U.S. Army Corp. of Engineers for the specific site. The guidance identified in the Standard Review Plan (SRP) Sections 2.4.2, Revision 3, and 2.4.3, Revision 3 (Reference 19B.2.44-2); and GL 89-22 (Reference 19B.2.44-7) is used in predicting design flood levels. Furthermore, general requirements are defined in General Design Criterion (GDC) 2 (Reference 19B.2.44-3). The SRPs state that “design basis flood levels” incorporate the most severe historical environmental data with “sufficient margin”. What is considered to be “sufficient margin” and procedures for estimating PMFs are identified in Regulatory Guides 1.59 and 1.102; and ANSI/ANS 2.8 (References 19B.2.44-4, 19B.2.44-5, and 19B.2.44-6).

Acceptance Criteria

The acceptance criteria for the resolution of Issue 103 is that the site chosen for a commercial nuclear generating facility shall be designed to accommodate a maximum expected flood from precipitation without jeopardizing the safe operation of the facility, in accordance with the guidance given in SRP 2.4.2, Revision 3; SRP 2.4.3, Revision 3; and GL 89-22. Also, the facility design, including structures, systems, and components important to safety, shall meet the criteria specified in 10 CFR 50, Appendix A (GDC 2).

Resolution

The ABWR is designed to meet the requirements of GDC 2 as described in Subsection 3.1.2. This ABWR design is based upon a set of assumed site-related parameters. These parameters were selected to envelope most potential nuclear power plant sites in the United States. A summary of the assumed site design parameters, including maximum flood level, is given in Section 2.0, Table 2.0-1, and Section 3.4.

Detailed site characteristics based upon historical site specific environmental data will be provided by the site owner-operator for any specific application. The site owner-operator will review and evaluate these characteristics to ensure compliance with the enveloping assumptions of Tables 2.0-1 and 3.4.1.

Since the ABWR is designed in accordance with GDC 2 for the most severe expected environment conditions, including flooding, tornado, hurricane etc. and meets the intent of SRP Section 2.4.2, Revision 3; SRP Section 2.4.3, Revision 3; and GL 89-22 with respect to plant design, this issue is resolved for the ABWR design.

References

19B.2.44-1 NUREG-0933, "A Prioritization of Generic Safety Issues (with Supplements 1-15)", U.S. NRC, April 1993.

19B.2.44-2 NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants—LWR Edition", U.S. NRC.

19B.2.44-3 10 CFR 50, Appendix A, "General Design Criteria for Nuclear Power Plants", Code of Federal Regulations, Office of the Federal Register, National Archives and Records Administration.

19B.2.44-4 Regulatory Guide 1.59, "Design Basis Floods for Nuclear Power Plants", U.S. NRC.

19B.2.44-5 Regulatory Guide 1.102, "Flood Protection for Nuclear Power Plants", U.S. Nuclear Regulatory Commission.

19B.2.44-6 ANSI/ANS 2.8, "Standard for Determining Design Basis Flooding at Power Reactor Site", American Nuclear Society.

19B.2.44-7 GL 89-22, "Potential for Increased Roof Loads and Plant Area Flood Runoff Depth at Licensed Nuclear Plants Due to Recent Change in Probable Maximum Precipitation Criteria", National Weather Service, October 19, 1989.

19B.2.45 105: Interfacing Systems LOCA at BWRs

Issue

In all currently operating light water reactors, there are a number of high/low pressure interfaces between the reactor coolant pressure boundary (RCPB) and connected systems. Thus there are systems in BWRs that are designed for a pressure lower than that of the primary system. For example, the BWR primary system operates at about 7.0 MPa, while the Residual Heat Removal (RHR) System can operate at pressures up to 3.53 MPa. Isolation valves (at least two) and piping to the primary system are designed for about 8.73 MPa. The discharge of the BWR RHR System, which also functions as a low pressure injection system, passes through testable check valves prior to returning to the reactor coolant system.

The common concern in the above issue is that either tests that require valve actuation, valve leakage, or multiple valve failures could result in a system pressure that exceeds the design pressure of the low pressure emergency cooling or decay heat removal systems causing them to fail from overpressure.

Risk calculations on existing plants suggest there may be a need for improved protection against the potential for overpressurization of some emergency cooling and decay heat removal systems (Reference 19B.2.45-1).

Acceptance Criteria

Reference 19B.2.45-2 indicated that future ALWR designs like the ABWR should reduce the possibility of a LOCA outside containment by designing (to the extent practicable) all systems and subsystems connected to the reactor coolant system (RCS) to an ultimate rupture strength (URS) at least equal to full RCS pressure.

Reference 19B.2.45-3 found that for the ABWR the design pressure for the low-pressure piping systems that interface with the RCPB should have the following criteria to achieve satisfactory retention of the full 7.17 MPa reactor pressure on an ultimate rupture strength basis.

(1) The design pressure for the low-pressure piping systems that interface with the RCPB pressure boundary should be equal to 0.4 times the normal operating RCPB pressure of 7.17 MPa.

(2) The minimum wall thickness of the low-pressure piping should be no less than that of a standard weight pipe.

(3) The remaining components in the low-pressure systems should also be designed to a design pressure of 0.4 times the normal operating reactor pressure [i.e., 2.93 MPa]. This is accomplished for the ABWR by the revised boundary symbols of the P&IDs to the 2.91 MPa design pressure, which includes all the piping and components associated with the boundary symbols.

(4) A Class 300 valve is adequate for ensuring the pressure of the low-pressure piping system under full reactor pressure.

(5) The design is to be in accordance with the ASME Code, Section III, Subarticle NC/ND-3600.

(6) Periodic surveillance and leak rate testing are required of the pressure isolation valves per Technical Specification requirements as a part of the In-Service Inspection (ISI) program.

Resolution

The ABWR design was evaluated and upgraded to comply with the above criteria. Criteria numbered 1 through 4 were accepted and implemented in Tier 2 documentation primarily by indicating the design pressure and design features on the system P&IDs (Piping and Instrument Diagrams). Criteria 5 and 6 were originally part of the ABWR design, and no upgrade was required to comply.

The increased design pressure was extended, forming an URS region extending outward from the RCPB, to the extent practicable. 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 ABWR:

- It is impractical to design large tank structures to the URS design pressure that are vented to atmosphere and have a low design pressure. Tanks included in this category are:
  - Condensate storage tank,
  - Standby Liquid Control System (SLCS) main tank,
  - Low Conductivity Waste (LCW) receiving tank,
  - High Conductivity Waste (HCW) receiving tank,
  - Fuel Pool Cooling (FPC) System skimmer surge tank, and
  - Fuel Pool Cooling (FPC) System spent-fuel storage pool and cask pit.

  These are termed low pressure sinks for the purposes of this discussion. The suppression pool is also a low pressure sink that is impractical to upgrade its pressure since it is part of the containment structure, which is designed to contain the most severe LOCA.

- It is impractical to consider a disruptive open flow path from reactor pressure to a low pressure sink. As a consequence, the furthest downstream valve in such a path is assumed closed (with nominal leakage) so that essentially all of the static reactor pressure is contained by the URS upgrade.

- It is impractical to design piping systems (that are connected to a low pressure sink) to URS design pressure when the piping is always locked open to a low pressure sink by locked open valves. Nominal leakage past the last closed valve is the only pressure source that could pressurize the line, and that line is locked open to the low pressure sink vented to atmosphere.

As implied above, boundary limits of the URS design are established assuming slow rates of leakage between high and low pressure regions. A key assumption to understanding the establishment of the boundary limits from the above practicality basis is that only static pressure conditions are considered. Static conditions result by assuming that the last valve in the URS region adjacent to a low pressure sink remains closed, yet considering a slow leak rate that would not over pressurize the down stream piping and components. Thus, the dynamic pressurization effects, violent high flow transients, and temperature escalations are precluded that would occur if the full RCPB pressure was connected directly to the low pressure sink. This means only static pressurization with small leak flow needs to be considered, and low pressure sinks do not pressurize because the path to the sink is open.

The following twelve systems, interfacing directly or indirectly with the RCPB, were evaluated and upgraded.

(1) Residual Heat Removal (RHR) System

(2) High Pressure Core Flooder (HPCF) System

(3) Reactor Core Isolation Cooling (RCIC) System

(4) Control Rod Drive (CRD) System

(5) Standby Liquid Control System (SLCS)

(6) Reactor Water Cleanup (CUW) System

(7) Fuel Pool Cooling Cleanup (FPC) System

(8) Nuclear Boiler System (NBS)

(9) Reactor Recirculation System (RRS)

(10) Makeup Water Condensate (MUWC) System

(11) Makeup Water Purified (MUWP) System

(12) Radwaste System (LCW Receiving Tank, HCW Receiving Tank).

The detailed system evaluation for ISLOCA is provided in Attachment 3MA.

The low pressure piping boundaries were upgraded to URS pressures and extend to the last closed valve connected to piping interfacing a low pressure sink, such as the suppression pool, condensate storage tank or other open configuration (identified pool or tank). The lines from the URS boundary to the low pressure sink do not pressurize because the path to the sink is open. Each interfacing system's piping was reviewed for upgrading. For some systems, with low pressure piping and normally open valves, the valves were changed to lock open valves to ensure an open piping pathway from the last URS boundary to the tank or low pressure sink.

In addition to the above 12 systems, the following two systems were identified as requiring an Interfacing System LOCA (ISLOCA) evaluation.

(1) Condensate, Feedwater and Condensate Air Extraction (F, FDW, AO) System

(2) Sampling (SAM System)

However, these two systems are not in sufficient detail to perform an ISLOCA evaluation. Therefore, an ISLOCA evaluation for these two systems is cited in Tier 2 as requirements for the COL applicant.

The periodic surveillance testing of the ECCS injection valves that interface with the reactor coolant system might lead to ISLOCA conditions if their associated testable check valve was stuck open. To avoid this occurrence, the RHR, HPCF, and RCIC motor-operated injection valves will only be tested during low pressure shutdown operation. This practice follows from the guidance given by Reference 19B.2.45-4.

Although the following is not a new design feature, the RHR shutdown cooling suction line containment isolation valves are also only tested during shutdown operation. These valves are interlocked against opening for reactor pressure greater than the shutdown cooling setpoint approximately 1.03 MPaG.

In summary, based on the NRC staff's new guidance cited in References 19B.2.45-2 through 19B.2.45-5, the ABWR is in full compliance. For ISLOCA considerations, a design pressure of 1.91 MPa and pipe having a minimum wall thickness equal to standard grade has been provided as an adequate margin with respect to the full reactor operating pressure of 7.17 MPa by applying the guidance recommended by Reference 19B.2.45-2. This design pressure was applied to the low pressure piping at their boundary symbols on the P&IDs; and, therefore, impose the requirement on the associated piping, valves, pumps, tanks, instrumentation and all other equipment shown between boundary symbols. A note was added to each URS upgraded P&ID requiring pipe to have a minimum wall thickness equal to standard grade. Upgrading revisions were made to 12 systems.

References

19B.2.45-1 NUREG-0933, "Resolution of Generic Safety Issues (with Supplements 1-33)", U.S. NRC, August 2010.

19B.2.45-2 Dino Scaletti, NRC, to Patrick Marriott, GE, "Identification of New Issues for the General Electric Company Advanced Boiling Water Reactor Review", September 6, 1991.

19B.2.45-3 Chester Poslusny, NRC, to Patrick Marriott, GE, "Preliminary Evaluation of the Resolution of the Intersystem Loss-of-Coolant Accident (ISLOCA) Issue for the Advanced Boiling Water Reactor (ABWR) – Design Pressure for Low-Pressure Systems", December 2, 1992, Docket No. 52-001.

19B.2.45-4 James M. Taylor, NRC to The Commissioners, SECY-90-016, "Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements", January 12, 1990, page 8, paragraph 7.

19B.2.45-5 James M. Taylor, NRC, to The Commissioners, SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs", April 2, 1993.

19B.2.46 106: Piping and the Use of Highly Combustible Gases in Vital Areas

Issue

Generic Safety Issue (GSI) 106 in NUREG-0933 (Reference 19B.2.46-1) was initiated to address the issue of combustible or explosive mixtures of gases accumulating in buildings containing safety-related equipment at a nuclear power plant and the potential disablement of these safety-related equipment if the accumulated gas mixtures are inadvertently ignited.

Except for hydrogen, most combustible gases are used in limited quantities and for relatively short periods of time at a nuclear power plant. Hydrogen is used as a coolant for electric generators and reactor water chemistry in both BWRs and PWRs, and is usually stored in high pressure storage vessels, and supplied to various systems through standard piping. The concern is the potential for leakage, accumulation of combustible or explosive mixtures of gases, and the inadvertent ignition of the gas. The ensuing combustion or explosion could damage or cause failure of safety-related equipment, thereby contributing to a possibly significant increase in the core-melt probability of the plant.

Generic Letter 93-06 (Reference 19B.2.46-2) discusses Issue 106 and risks from failures of hydrogen system lines and components.

Acceptance Criteria

The acceptance criteria for the resolution of GSI 106 are that systems for delivery of hydrogen or other combustible gases be designed to preclude:

(1) large release and accumulation of combustible or explosive mixtures of gases, and

(2) combustion and explosions which could damage or cause failure of safety-related equipment.

This can be accomplished either by designing the piping systems to preclude failure, providing means to detect and limit the amount of hydrogen leakage and accumulation in the event of a piping system break or large leak, or locating safety-related equipment in areas that are not susceptible to damage from combustion and explosions.

Furthermore, the designer shall follow the guidance described in SRP Section 9.5-1 (Reference 19B.2.46-3) and modified BTP CMEB 9.5-1, Part C.5.d(5).

Resolution

The ABWR design uses hydrogen for the Hydrogen Water Chemistry (HWC) System and the main generator bulk hydrogen supply system. These systems are non-nuclear, non-safety-related and are located in the turbine building which is a non-safety-related structure in a non-vital area.There is no significant amount of hydrogen or other highly combustible gases in any vital area for ABWR design.

The arrangement of buildings at the facility, and location of building doors and the bulk hydrogen storage tanks is such that damage to buildings containing safety-related equipment due to combustion of hydrogen or an explosion is unlikely (Subsection 10.2.2.2).

The prevention and mitigation of hydrogen combustion and explosions are discussed in Subsections 9.3.9 and 10.2.2. The wind and tornado loading and missile protection for buildings containing safety-related equipment are presented in Sections 3.3 and 3.5, respectively.

Therefore, this issue is resolved for the ABWR.

References

19B.2.46-1 NUREG-0933, “Resolution of Generic Safety Issues (with Supplements 1-33)”, U.S. NRC, August 2010.

19B.2.46-2 Generic Letter 93-06, Research Results of Generic Safety Issue 106, “Piping and the Use of Highly Combustible Gases in Vital Areas”, October 25, 1993.

19B.2.46-3 NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants—LWR Edition”, U.S. NRC.

19B.2.47 Not Used

19B.2.48 118: Tendon Anchorage Failure

Issue

Generic Safety Issue (GSI) 118 in NUREG-0933 (Reference 19B.2.48-1), addresses the failure of lower vertical tendon anchor heads in a PWR prestressed concrete containment structure. The failures appear to have been caused by hydrogen stress cracking. The hydrogen is liberated by zinc in the presence of water. Quantities of water ranging from a few cubic centimeters (a few ounces) to about 5.7E+3 cm^3 (1.5 gallons) were found in the grease caps.

Acceptance Criteria

For the ABWR design, the primary containment structure consists of a reinforced concrete design. Since the prestressed concrete containment design is not used in the ABWR Standard Plant design, the tendon anchorage failure issue is not applicable; therefore, no acceptance criteria are needed.

Resolution

For the ABWR design, the primary containment structure is of a reinforced concrete design; therefore, Issue 118 is not applicable.

References

19B.2.48-1 NUREG-0933, “A Prioritization of Generic Safety Issues (with Supplements 1-15)”, U.S. NRC, April 1993.

19B.2.49 120: On-Line Testability of Protection Systems

Issue

Issue 120 was established to examine the on-line (at-power) testability of protection systems and the possibility that some plants might not provide complete testing capability. Protection systems consist of the reactor protection system (RPS) and the engineered safety features actuation system (ESFAS) (Reference 19B.2.49-1).

Acceptance Criteria

The acceptance criteria for the resolution of Issue 120 is compliance with General Design Criterion (GDC) 21, “Protection System Reliability and Testability”, of Appendix A to 10 CFR 50 (Reference 19B.2.49-5). Supplementary guidance is provided in Regulatory Guides 1.22 and 1.118 (References 19B.2.49-2 and 19B.2.49-3), and IEEE Standard 338 (Reference 19B.2.49-4) to ensure that protection systems (including logic, actuation devices, and associated actuated equipment) will be designed to permit testing while the plant is operating without adversely affecting the plant’s operation. These requirements apply to both the RPS and the ESFAS. Existing Standard Technical Specification indicate that it is desirable to test all protection systems every six months.

Resolution

In the ABWR design the RTIS and ELCS can be tested during reactor operation. The first five tests are primarily manual tests and, although each individually is a partial test, when combined with the sixth test they constitute a complete system test. The sixth test is the test of the safety system logic and control which tests the complete system, excluding sensors and actuators. Online testability of protection systems is explained in Subsection 7.1.2.1.6. Periodic surveillance testing is required by Chapter 16, LCO 3.3.1.1 for the SSLC sensor instrumentation, LCO 3.3.1.2 for the RPS and the MSIV actuation, and LCO 3.3.1.4 for the ESF actuation.

On line testability of protection systems may fall into the Operational Reliability Assurance Program (O-RAP). The COL applicant will specify the policy and implementation procedures for the O-RAP, as described in Subsection 17.3.9.

Therefore, this issue, 120, is resolved for ABWR.

References

19B.2.49-1 NUREG-0933, “A Prioritization of Generic Safety Issues (with Supplements 1-15)”, U.S. NRC, April 1993.

19B.2.49-2 Regulatory Guide 1.22, “Periodic Testing of Protection System Actuation Functions.”

19B.2.49-3 Regulatory Guide 1.118, “Periodic Testing of Electric Power and Protection Systems.”

19B.2.49-4 IEEE Standards 338, “Criteria for the Periodic Surveillance Testing of Nuclear Power Generating Station Safety Systems.”

19B.2.49-5 10 CFR 50, Appendix A, "General Design Criteria for Nuclear Power Plants," Office of the Federal Register, National Archives and Records Administration.

19B.2.50 121: Hydrogen Control for Large, Dry PWR Containments

Issue

This issue, 121, concerns the control of hydrogen concentrations in large, dry PWR containments during and after a degraded core accident. In December 1984, the staff recommended that rulemaking with regard to this issue could be safety deferred due to the greater inherent capability of these containments to accommodate large quantities of hydrogen. Ongoing NRC experimental and analytical programs are intended to provide data to support a final recommendation on whether safe shutdown equipment is likely to survive a hydrogen burn (Reference 19B.2.50-1).

Acceptance Criteria

The acceptance criteria for the resolution of Issue 121 is that the control of hydrogen generated in the containment in a degraded core accident shall meet the requirements of 10 CFR 50.34(f) (Reference 19B.2.50-2) on limiting the distributed hydrogen concentration to 10%, on limiting combustible concentrations, and on maintaining safe shutdown equipment and containment integrity.

Resolution

This issue does not apply to BWRs and pressure suppression containment. Also, the ABWR primary containment is inerted and is, therefore, protected from hydrogen combustion regardless of the amount or rate of hydrogen generation.

References

19B.2.50-1 NUREG-0933, "A Prioritization of Generic Safety Issues (with Supplements 1-15)", U.S. NRC, April 1993.

19B.2.50-2 10 CFR 50, Appendix A, "General Design Criteria for Nuclear Power Plants", Office of the Federal Register, National Archives and Records Administration.

19B.2.51 124: Auxiliary Feedwater System Reliability

Issue

Issue 124 in NUREG-0933 (Reference 19B.2.51-1) addresses Auxiliary Feedwater System reliability and availability and its impact on reducing core-melt frequency in PWRs.

Acceptance Criteria

The acceptance criteria for the resolution of Issue 124 is that the Auxiliary Feedwater System shall be designed for a high degree of reliability (i.e., using reliability analyses the system shall attain 0.0001 to 0.00001 unavailability per demand).

Resolution

This issue, 124, is not applicable to BWRs and is, therefore, resolved for ABWR.

References

19B.2.51-1 NUREG-0933, “A Prioritization of Generic Safety Issues (with Supplements 1-15)”, U.S. NRC, April 1993.

19B.2.52 128: Electrical Power Reliability

Issue

Issue 128 in NUREG-0933 (References 19B.2.52-1, 19B.2.52-6, and 19B.2.52-7), addresses the reliability of on-site electrical systems.

The minimum acceptable DC power system is comprised of two physically independent divisions which supply DC power for control and actuation of redundant safety-related systems. Questions have been raised concerning the position of regulatory staff, including the application of the single failure criterion for assuring a reliable DC power supply. These concerns stem from the dependence on DC power of the decay heat removal systems required for long-term heat removal. Failure of one DC division would generally result in a reactor scram which then would require removal of decay heat. The frequency of reported single DC division failures gives rise to the concern that the second DC division may not be available.

Two of the specific reasons for the concern that safety-related power may be unreliable are also addressed by this issue. One is that some operating nuclear power plants do not have technical specifications or administrative controls governing operational restrictions for Class 1E 120 VAC vital instrument buses and associated inverters. Without such restrictions these power sources could be out of service indefinitely and thereby may place certain safety systems in a situation where they could not meet the single failure criterion. The other is that the design of some plants do not provide interlocks to prevent the inadvertent closure of the single tie breaker between the 4160 V Class 1E buses.

Acceptance Criteria

The acceptance criteria for the resolution of GSI 128 are encompassed in three interrelated issues, i.e., A-30, 48 and 49, which are summarized as follows:

The acceptance criteria for Issue A-30 are:

(1) Non-safety-related loads shall be placed on completely separated DC power supplies.

(2) Class 1E (safety-related) DC power systems shall be divided into physically and electrically independent systems.

(3) Bus tie breakers between DC systems shall not be used.

(4) Operation and maintenance procedures and/or Technical Specifications governing maintenance and out-of-service shall be implemented.

The acceptance criterion for Issue 48 is that administrative controls or Technical Specifications shall be provided to govern operational restrictions for Class 1E 120 VAC vital instrument buses and associated inverters.

The acceptance criterion for Issue 49 is that the bus tie breakers, if provided between Class 1E emergency buses, shall be redundant and physically separated and open as a condition of operability of the redundant Class 1E electrical distribution system.

Resolution

The ABWR safety-related DC power system design as listed below provides reliable DC buses for safety-related electrical loads and meets the acceptance criteria specified by the resolution of Issue A-30. See Subsection 8.3.2 for as discussion of compliance.

(1) Does not supply power to any non-Class 1E loads, with the exception of the associated DC emergency lighting as described in Subsection 9.5.3.2.3.1.

(2) Consists of four separate and independent DC battery systems.

(3) Does not contain any direct bus ties between DC battery systems. However, it does contain two standby battery chargers. Each standby battery charger is capable of supplying one of two divisional DC systems. Redundant key locked breakers are provided to prevent manual paralleling between divisions. No automatic connections are provided between DC divisions.

(4) COL applicants are required to provide administrative controls for standby battery charger operation and Technical Specifications Sections 3.8.4, 3.8.5, 3.8.6 and 3.8.10 are provided for operational restrictions and allowable out of service times.

The ABWR design meets the acceptance criterion specified above for the resolution of Issue 48 by the system design and Technical Specifications. As described in Subsection 8.3.1.1.4.2, the ABWR design consists of four separate and independent Class 1E 120 VAC vital instrument buses with their respective inverters. There are no bus ties between the four divisions. Operational restrictions are provided in Technical Specifications Sections 3.8.7, 3.8.8, 3.8.9, and 3.8.10 to assure the onsite Class 1E AC and DC power distribution system availability and thus an uninterruptable power source for safety-related systems and components. The Technical Specifications include specific requirements regarding a periodic evaluation of the onsite power system bus condition which considers such availability items as correct breaker and the alignment and adequate bus voltage.

The ABWR design meets the acceptance criteria as stated above for the resolution of Issue 49. The ABWR Class 1E diesel generator bus design does not contain bus tie breakers between Class 1E divisions. However, it is possible to manually cross-connect the Class 1E diesel buses through the combustion turbine generator (CTG) connections since the ABWR design does have the capability of providing power to each diesel bus from the CTG. To cross-connect any two diesel buses, at least four circuit breakers must be at close positions and at least one circuit breaker must be racked in prior to closing. Each diesel generator is provided with synchronizing equipment for paralleling with offsite power supplies. The normal and alternate feeder breakers to the diesel buses are interlocked to prevent paralleling offsite circuits. See Subsections 8.2.1.3 and 8.3.1.1.6 for a discussion of compliance. As discussed in the resolution of Issue 48, the Technical Specification Sections 3.8.9 and 3.8.10 will include operational restrictions and periodic evaluation of correct breaker alignment of Class 1E onsite power distribution system.

Additionally, the ABWR Class 1E power distribution system design as described in Subsection 8.3 fully complies with the IEEE 308 (Reference 19B.2.52-5) and 603 (Reference 19B.2.52-4).

In summary, the ABWR design for the electrical power system avoids the problems described in this issue. Each division of the engineered safety systems has emergency onsite sources of AC and DC power, and at least two connections for offsite power, all of which are separate and independent. There are three divisions of decay heat removal, each with its own emergency AC and DC power source.

This issue is considered resolved for the ABWR.

References

19B.2.52-1 NUREG-0933, "A Prioritization of Generic Safety Issues (with Supplements 1-15)", U.S. NRC, April 1993.

19B.2.52-2 NUREG-0305, "Technical Report on DC Power Supplies in Nuclear Power Plants", U.S. NRC, July 1977.

19B.2.52-3 NUREG-0666, "A Probabilistic Safety Analysis of DC Power Supply Requirements for Nuclear Power Plants", U.S. NRC, April 1981.

19B.2.52-4 IEEE Standards 603, "Standard Criteria for Safety Systems for Nuclear Power Generating Stations", The Institute of Electrical and Electronics Engineers, Inc.

19B.2.52-5 IEEE Standard 308, "Criteria for Class 1E Electric Systems for Nuclear Power Generating Stations", The Institute of Electrical and Electronic Engineers, Inc.

19B.2.52-6 NRC Letter to All Holders of Operating Licensees, "Resolution of Generic Issue A-30, 'Adequacy of Safety-Related DC Power Supplies,' Pursuant to 10 CFR 50.54(f) (Generic Letter 91-06)", April 29, 1991.

19B.2.52-7 NRC Letter to All Holders of Operating Licenses, "Resolution of Generic Issues 48, 'LCOs for Class 1E Vital Instrument Buses,' and 49, 'Interlocks and LCOs for Class 1E Tie Breakers' Pursuant to 10 CFR 50.54(f) (Generic Letter 91-11)", July 18, 1991.

19B.2.53 142: Leakage Through Electrical Isolators in Instrumentation Circuits

Issue

Electronic isolators are used to maintain electrical separation between safety and non-safety-related electrical systems in nuclear power plants, preventing malfunctions in the non-safety systems from degrading performance of safety-related circuits. Isolators are primarily used where signals from Class-1E safety-related systems are transmitted to non-Class 1E control or display equipment.

There are a number of devices which may qualify as electrical isolators in a nuclear power plant, including fiber optic and photo-electric couplers, transformer-modulated isolators, current transformers, amplifiers, circuit breakers, and relays. These isolators are designed and tested to prevent the maximum credible fault applied in the transverse mode on the non-Class 1E side of the isolator from degrading the performance of the safety-related circuits (Class-1E side) below an acceptable level.

This issue was identified by the staff in June 1987 and arose from observations made during Safety Parameter Display System (SPDS) evaluation tests that demonstrated, for electrical transients below the maximum credible level, a relatively high level of noise could pass through certain types of isolation devices and be transmitted to safety-related circuitry. In some cases, the amount of energy that can pass through the isolator may be sufficient to damage or seriously degrade the performance of Class 1E components, while, in other cases, electrically-generated noise on the circuit may cause the isolation device to give a false output.

Due to the fact that there are a great number of each type of isolator in the field, this issue would require the staff to determine the extent to which potentially susceptible isolators are used in nuclear power plants and to identify the systems in which they are used. An NRC bulletin to all licensees to provide input on these questions would be necessary.

Acceptance Criteria

Assuming that the staff determines from the licensee responses to the proposed bulletin that a potential problem exists, a research program consisting of two major objectives would have to be initiated to develop the solution to this issue. The first objective would be to develop test procedures and acceptance criteria for isolators that licensees could use to determine the adequacy of installed isolators. The second objective would involve development of appropriate hardware fixes that could resolve the issue.

Therefore, with a reliable data base the final step in the solution to this issue would be the issuance of a generic letter to licensees with the following guidelines for:

(1) inspection and testing of all electrical isolation devices between Class 1E and non-Class 1E systems;

(2) repair/replacement of isolators that fail the tests, including description of acceptable hardware fixes to the isolators; and

(3) implementation of an annual program to inspect and test all electronic isolators between Class 1E and non-Class 1E systems.

Issue 142 must meet the requirements of the Licensing Review Bases (LRBs) Criteria on isolators [the LRBs are contained in a letter from T. Murley of NRC to R. Artigas of G.E., dated August 7, 1987 (Reference 19B.2.53-3)].

Resolution

Fiber optic data links are the only type of isolation device used in the ABWR for electrical isolation of logic level and analog signals between protection divisions and from protection divisions to non-safety-related equipment (Subsection 7A.3). Subsection 7A.3 resolves issues regarding the Licensing Review Basis Criteria on isolators.

Maximum credible electrical faults applied at the outputs of isolation devices do not apply to fiber optic systems. The maximum credible fault is cable breakage causing loss of signal transmission. Faults cannot cause propagation of electrical voltages and currents into other electrical circuitry at the transmitting or receiving ends. Conversely, electrical faults originating at the input to the fiber optic transmitter can only damage the local circuitry and cause loss or corruption of data transmission; damaging voltages and currents will not propagate to the receiving end.

Fiber optic isolation devices are expected to have less difficulty than previous isolation devices in complying with all qualification requirements due to their small size, low mass, and simple electronic interfaces. The basic materials and components, except for the fiber optic cable itself, are the same as those used in existing, qualified isolation devices.

Fiber optic cable is used for Class 1E isolation and does not use any electrical power to accomplish that function.

The isolating devices used for ABWR are similar to the Group 1 types referred to in Reference 19B.2.53-2. They are of the long fiber optic cable design, so transmitting and receiving ends are separated by a significant distance [typically one meter (several feet) to several hundred meters (several hundred feet)]. These types of designs had the best isolating characteristics of the various isolators compared in the NUREG study (Reference 19B.2.53-2).

Typically, the electrical-to-optical interfaces are part of the general logic processing equipment within a channel and do not reside in separate isolator units. The fiber optic interfaces receive the protection from EMI and surge currents designed into the logic equipment (for example, power supply decoupling, shielding, filtering, single-point signal common connection to chassis ground, and chassis ground connection to ground bus). The equipment will undergo EMI and surge testing to the standards identified in the NUREG or equivalent.

The results of the NUREG tests show that the fiber optic type of isolators exhibited no or very little effects from the major fault and lightning surge tests. Only surge and EMI tests applied to the isolator power supplies caused damage to the isolator input side, mainly because of the output and input supplies sharing a common, commercial AC power line. For the ABWR, RPS and ESF functions are supplied from different plant power sources (120 Volt Vital AC and 125 VDC, respectively). The low voltage DC supplies fed from these sources are highly regulated and filtered. Thus, isolator circuits are isolated from most power source transients.

See Subsection 19B.3.2 for COL license information pertaining to testing of isolators.

Therefore, this issue is resolved for the ABWR.

References

19B.2.53-1 Memorandum for B. Morris from B. Sheron (NRC Staff), “Proposed Generic Issue on Leakage Through Electrical Isolators”, June 23, 1987.

19B.2.53-2 NUREG/CR-3453, “Electronic Isolators Used in Safety Systems of U.S. Nuclear Power Plants”, U.S. NRC, March 1986.

19B.2.53-3 A letter from T.E. Murley of NRC to R. Artigas of G.E., “Advanced Boiling Water Reactor Licensing Review Bases”, dated August 7, 1987.

19B.2.53-4 NUREG-0933, “Resolution of Generic Safety Issues (with Supplements 1-33)”, U.S. NRC, August 2010.

19B.2.54 143: Availability of Chilled Water Systems and Room Cooling

Issue

In recent years, several nuclear power plants have experienced problems with safety system components and control systems that were caused by a partial or total loss of heating, ventilating, and air conditioning (HVAC) systems. Many of these problems exist because of the desire to provide increased fire protection and the need to avoid severe temperature changes in equipment control circuits. Since the Browns Ferry fire, considerable effort has been expended to improve the fire protection of equipment required for safe shutdown. Generally, this improvement has been made by enclosing the affected equipment in small, isolated rooms. The result has been a significant increase in the impact of the loss of room cooling. Plant control and safety have improved with the introduction of electronic integrated circuits; however, these circuits are more susceptible to damage from severe changes in temperature caused by the loss of room cooling.

It is believed that failures of air cooling systems for areas housing key components, such as residual heat removal pumps, switchgear, and diesel generators, could contribute significantly to core-melt probability in certain plants. Because corrective measures are often taken at the affected plants once such failures occur, the impact of these failures on the proper functioning of air cooling systems may not have been considered in the final PRA of plants. Thus, plants with similar inherent deficiencies may not be aware of these problems.

Operability of some safety-related components is dependent upon operation of HVAC and chilled water systems to remove heat from the rooms containing the components. If chilled water and HVAC systems are unavailable to remove heat, the ability of the equipment within the rooms to operate as intended cannot be assured (Reference 19B.2.54-1).

Acceptance Criteria

The impact of loss of room cooling is an important design consideration for the ABWR. Under these circumstances, a key design objective is to ensure that ABWR safety-related equipment will still operate reliably during the period of loss of room cooling. The following criteria will establish an acceptable ABWR design.

(1) An evaluation of the dependencies or non-dependencies of safety-related equipment on HVAC cooling shall be performed. This evaluation will include assessments of room heat load and heatup rates, and establish equipment operating conditions. The capability of the equipment to withstand these conditions without loss of function shall be established.

(2) For equipment found to be significantly dependent on HVAC cooling, an assessment of the HVAC system reliability shall be performed. PRA analyses will be carried out to assess plant risk and determine whether any modifications are necessary.

(3) Corrective design measures shall be identified where necessary to reduce plant risk.

Resolution

The ABWR design uses large chilled-water systems to provide essential environmental cooling, which in turn includes cooling of the solid-state electronic components. The performance of chilled-water systems under varying accident loads and during loss-of-offsite-power events, and their ability to operate after a prolonged station blackout are evaluated. The ABWR design features which address the acceptable criteria include the following:

- As part of actions to mitigate station blackout events, COL applicants are required to perform an analysis to confirm that RCIC room temperature will not exceed equipment design temperature without room cooling for at least 8 hours as stated in Subsection 19.9.9.(3).

- Table 3I-13 defines the most severe thermodynamic environment conditions at which other injection systems (HPCF, RHR/LPFL) equipment are qualified in their designs. These temperature extremes are expected to occur at about 6 hours into the post postulated accident periods without room cooling. If power is lost, the HVAC systems which provide room cooling for these plant areas will not be available for 10 minutes until power is recovered by the combustion turbine generator. The capability of these equipment to withstand the temperature environment which will develop during this 10 minute period is assured for the ABWR design.

- Detailed design specifications for ABWR safety-related equipment will specify the room conditions under which equipment must operate without room cooling. Room heat assessments will be performed to establish environmental conditions for equipment specification (Subsection 3I.3.2.1).

- Potential modifications including procedure changes or hardware changes evaluated through PRA analyses to ensure acceptable plant risk.

Despite the few extreme events which would cause loss of room cooling, the ABWR design incorporates several safety-related HVAC systems which provide room cooling under most circumstances. These systems include:

- R/B Safety-Related Equipment HVAC—providing fan-coil units for safety-related equipment rooms, including the 3 divisions of ECCS pump rooms (Subsection 9.4.5.2).
- R/B Safety-Related Electrical Equipment HVAC—3 divisions each with 2-100% supply/exhaust fans, 1 air conditioning unit (Subsection 9.4.5.4).
- R/B Safety-Related Diesel Generator HVAC—2 supply fans per division (Subsection 9.4.5.5).
- HVAC Emergency Cooling Water 3 Divisions—provides chilled water to R/B electrical equipment HVAC, C/B HVAC, and C/R habitability HVAC (Subsection 9.2.13).

The reliability and availability of these safety-related HVAC systems will be specified in detailed design to ensure a controlled environment for operation of safety-related equipment.

References

19B.2.54-1 NUREG-0933, "Resolution of Generic Safety Issues (with Supplements 1-33)", U.S. NRC, August 2010.

19B.2.55 145: ACTIONS TO REDUCE COMMON CAUSE FAILURES

Issue

Issue 145 is concerned that common cause failures can be a major cause of a system failure. The TMI-2 and David Besse incidents were examples of scenarios involving common cause failures (Reference 19B.2.55-1).

Effective maintenance is important to ensure that design assumptions and margins in the original design basis are maintained. In the design of nuclear power plants, an important safety margin is the redundancy of equipment to perform safety functions. This redundancy, however, can be degraded by common cause failures. Therefore, defense against such failures (by root cause analyses and investigations) over the life of the plant is an important part of the licensee's maintenance program.

The NRC has published Regulatory Guide 1.160 (Reference 19B.2.55-3) to implement the maintenance rule, 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants "(Reference 19B.2.55-2).

Acceptance Criteria

The acceptance criteria for the resolution of Issue 145 is to demonstrate compliance with the maintenance rule, 10 CFR 50.65.

Resolution

Compliance with 10 CFR 50.65 will be the responsibility of the COL applicant.

In addition, the ABWR design demonstrates in Chapter 19 its capability to respond to system interactions and common cause failures (Subsection 19.2.3.4).

Actions to reduce common cause failures may fall into the Operational Reliability Assurance Program (O-RAP). The COL applicant will specify the policy and implementation procedures for the O-RAP as described in Subsection 17.3.9.

Therefore, this issue, 145, is resolved for the ABWR.

References

19B.2.55-1 NUREG-0933, "A Prioritization of Generic Safety Issues (with Supplements 1-15)", U.S. NRC, April 1993.

19B.2.55-2 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants", Office of the Federal Register, National Archives and Records Administration.

19B.2.55-3 Regulatory Guide 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants", U.S. NRC.

19B.2.56 151: Reliability of Anticipated Transient Without Scram Recirculation Pump Trip (ATWSRPT) in BWRs

Issue

Issue 151 in NUREG-0933 (Reference 19B.2.56-1) addresses the issue of the reliability of the ATWS RPT in BWRs. Issue 151 specifically identifies a reliability problem with GE's type AKF-25 circuit breaker and trip hardware [actually a type AKF-2-25 breaker, per NRC's IE Notice 87-12 (Reference 19B.2.56-2)].

Acceptance Criteria

The acceptance criterion for the resolution of Issue 151 is the use of reactor recirculation system pump trip hardware or method that is more reliable than the previously used AKF-2-25 breaker hardware or method.

Resolution

The design for the ABWR reactor recirculation system and RPT method and hardware is completely different from the previously designed BWR reactor recirculation systems and RPT trip methods. The design is more diverse and redundantly reliable. Rather than using only two recirculation pumps and the associated single RPT breakers, the ABWR will use ten pumps and multiple pump and RPT trip logic, circuits and hardware. Adjustable speed drive (ASD), recirculation internal pumps (RIPs) are used. The ABWR RPT trip hardware (not yet specifically identified) will be completely different. The ABWR does not use AKF-2-25 circuit breakers in the RPT logic circuits. Instead of using AKF-2-25 breaker switching hardware to provide a RPT, RFC controller switching and ASD gate inverter turn-off circuit hardware provides the RPT [Subsections 7.7.1.3(7) and 7.7.1.3(8)].

Thus, by diversity and redundancy in design, the ABWR addresses and resolves Issue 151.

This issue, 151, may fall into the Operational Reliability Assurance Program (O-RAP). The COL applicant will specify the policy and implementation procedures for O-RAP, as described in Subsection 17.3.9.

References

19B.2.56-1 NUREG-0933, "A Prioritization of Generic Safety Issues (with Supplements 1-15)", U.S. NRC, April 1993.

19B.2.56-2 IE Information Notice 87-12, "Potential Problems with Metal Clad Circuit Breakers, General Electric Type AKF-2-25", U.S. NRC, February 13, 1987.

19B.2.57 153: Loss of Essential Service Water in Light-Water Reactors

Issue

The Essential Service Water (ESW) system at a nuclear power plant supplies cooling water to transfer heat from various safety-related and non-safety-related systems and equipment to the ultimate heat sink of the plant. Under Issue 153, the staff examined all potential causes for ESW system unavailability, except those that are considered to be resolved by implementing the resolutions addressed in (Generic Letter (GL) 89-13 (Reference 19B.2.57-1), such as biofouling, sediment, corrosion, and erosion (Issue 51). The safety concerns of this issue include partial or complete loss of ESW system functions resulting from common causes (such as icing of the intake structure), degradation of the ESW system, design deficiencies, and procedural or maintenance errors. A complete loss of the ESW system could lead to a core-melt accident, posing a significant risk to the public.

The NRC evaluation of this issue has been completed.

Acceptance Criteria

The ESW system is needed in every phase of plant operations and, under accident conditions, supplies adequate cooling water to systems and components that are important to safe shutdown or to mitigate the consequences of the accident. Under normal operating condition, the ESW system provides component and room cooling (mainly via the component cooling water system). During shutdown it also ensures that the residual heat is removed from the reactor core. The ESW system may also supply makeup water to fire protection systems, cooling towers, and treatment systems at a plant.

The design features for the essential service water (ESW) system are summarized as follows:

- Performance Requirements
  - The ESW system will be designed to meet the required heat loads.
  - The ESW system will be provided with two pumps and two heat exchangers per division.
  - The plant designer will provide analyses for all potential operating conditions that properly account for uncertainties.
- System Arrangement
  - The ESW system will be divided into approximately three equal-sized divisions.
  - A division will be made up of independent piping systems, each with pumps, heat exchangers, strainers, controls and instrumentation, power supplies, and associated equipment required for regulating system flow.

In addition, the ESW design shall address partial or complete loss of ESW system functions resulting from common causes, degradation of the ESW system, design deficiencies, and procedural or maintenance errors. The plant designer should provide an assessment of these potential failure modes and their associated contributions to the core damage frequency and should identify dominant accident sequences.

Resolution

The ABWR Reactor Service Water (RSW) system removes heat from the Reactor Building Cooling Water (RCW) system and transfers that heat to the Ultimate Heat Sink (UHS). The RSW system is provided in three divisions. Each division has two pumps which send cooling water to three RCW heat exchangers. Normally one pump and two heat exchangers are operating in each division. When heat removal requirements increase, the remaining pump and heat exchanger are automatically put into operation. If additional heat removal capacity is needed, some of the non-safety-related cooling loads may be taken out of operation.

In case of failure which disables any of the three RSW divisions, the other two divisions meet plant safety shutdown requirements (Subsection 9.2.15 and Table 9.2-5).

The ABWR RSW system divisions are physically and electrically separated from each other. This reduces the potential effects of common causes. Normally, each division is operating at all times with the capability to put into service the remaining pump and heat exchanger at any time. Margin is provided in pump flow capacity (and in RCW heat exchanger heat removal capacity). Periodic testing of these components will be performed and corrective action taken when needed (Subsections 9.2.11.4 and 9.2.15.1.4).

Several potential causes of RSW system degradation are site dependent. The RSW system is designed to prevent this degradation from occurring. Additionally, the COL applicant will provide the following system design features for those portions of the system which are not in the ABWR standard plant scope:

- Adequate NPSH for the pumps at low UHS water levels.
- Low point drains and high point vents.
- Prevention of organic fouling (using methods such as trash racks, biocide treatment or thermal backwashing, as required).
- Component material selection suited to site water conditions.
- Protection against flooding, spraying, steam impingement, pipe whip, jet forces, missiles, fire and the effect of failure of any non-Seismic Category I equipment.

If required, recirculation of warm water through the intake structures will be provided to reduce the likelihood that ice will block cooling water flow (Subsections 9.2.5.4 and 9.2.15.2). Also system degradation is minimized by periodic testing and inspection to insure integrity and functional capability (Subsections 9.2.11.4 and 9.2.15.1.6).

The RSW pumps and pump house will be designed by the COL applicant, who will consider and reduce the effects of procedural and maintenance errors.

When the future plant-specific design is prepared, another assessment will be made of potential failure modes and their associated contributions to the core damage frequency and the dominant accident sequences will be identified.

These issues are resolved for the ABWR through the design features of the RSW system, the system design features, and the Operational Reliability Assurance Activities (Subsection 17.3.9) which will be provided by the COL applicant.

References

19B.2.57-1 Generic Letter 89-13, “Service Water System Problems Affecting Safety-Related Equipment”, July 18, 1989.

19B.2.57-2 NUREG-0933, “Resolution of Generic Safety Issues (with Supplements 1-33)”, U.S. NRC, August 2010.

19B.2.58 155.1: More Realistic Source Term Assumptions

Issue

Current siting regulations (10 CFR Part 100) require that an accidental fission product release from the core into containment be assumed and that its offsite radiological consequences be evaluated against guideline doses given in Part 100. The postulated source term is derived from TID-14844 (Reference 19B.2.58-1) and is contained in Regulatory Guides 1.3 and 1.4. The regulatory guides specify a release into containment of 100% of the core inventory of noble gases and 50% of the iodine fission products. Half of the iodine is assumed to deposit on interior surfaces assuming instantaneous appearance within containment and that the iodine is predominately in elemental form (I_2).

Use of the TID-14844 source term has not been restricted to evaluation of plant mitigation features and site suitability. Regulatory applications of the source term are broad, including use as the basis for

(1) The post-accident environment for which safety-related equipment should be qualified.

(2) Post-accident habitability requirements for the control room.

(3) Post-accident sampling systems and accessibility.

A substantial amount of information has been developed to update knowledge about LWR severe accidents and behavior of fission products that could be released into containment. Studies have confirmed that although the TID-14844 source term is substantial and that its use has resulted in a high level of plant capability, the present recipe can be substantially improved.

In their staff requirements memorandum (SRM) dated January 25, 1991, the Commission approved the plan proposed by the staff to revise Part 100 to delete the source term and dose calculations and to directly specify site criteria; to issue (in parallel) an interim revision to Part 50 to retain the present source term and dose calculation (but not for siting purposes); to update the TID-14844 source term; and, in a second rule-making phase, to incorporate severe accident and revised source term insights for future plants. In their SRM dated April 11, 1991, the Commission requested the staff to make recommendations on the values of releases into containment (to update TID-14844), to provide a discussion of the status of EPRI’s comparable values, and to discuss the use of the updated source term in evaluations of existing and future plants.

Acceptance Criteria

The acceptance criteria for GSI 155.1 is that the plant shall be designed to ensure that the dose commitment to the public in the event of a licensing design basis accident shall be within those limits prescribed by existing regulations based upon the limitations of 10 CFR 100.

Resolution

The ABWR is being designed and analyzed to the existing Regulatory Guides, Standard Review Plans, and General Design Criteria which are based upon TID-14844 (e.g., Regulatory Guide 1.3, Standard Review Plan 15.6.5). The use of revised source terms based upon NUREG-1465 (Reference 19B.2.55-2) is premature for the ABWR based upon the lack of clarification of what is a design basis event under the revised source terms and lacking adequate guidance from the Commission as to acceptable methods and conditions, i.e., revised regulatory guides and standard review plans.

References

19B.2.58-1 DiNunno, J.J. et al, "Calculation of Distance Factors for Power and Test Reactor Sites", Technical Information Document 14844, March 23, 1962.

19B.2.58-2 Soffer, L. et al, "Accident Source Terms for Light-Water Nuclear Power Plants", NUREG-1465, Draft Report for Comment, U.S. NRC, June 1992.

19B.2.59 A-17: Systems Interactions in Nuclear Power Plants

Issue

Unresolved Safety Issue (USI) A-17 in NUREG-0933 (Reference 19B.2.59-1) addresses the concern that inconspicuous or unanticipated interdependencies may exist between systems and may result in a degradation of the predicted capability of safety systems in an accident or transient, in particular from flooding and water intrusion.

In its regulatory analysis in NUREG-1229 (Reference 19B.2.59-4), the NRC concluded that for future plants the existing Standard Review Plans (SRPs) (Reference 19B.2.59-5) in general cover Adverse System Interactions (ASIs) of concern, except for the areas of internal flooding and water intrusion. A flooding event could cause a transient and also disable the equipment needed to mitigate the consequences of the event. NUREG-1174 (Reference 19B.2.59-6) provided guidance in this area and references NRC Information Notices regarding operating plant experiences. The NRC plans to develop an SRP relative to flooding and water intrusion, but otherwise not issue new requirements. In the meantime, the NRC recommends that plant designers keep current on lessons learned from operating experience as reported in LERs, and that the Probabilistic Risk Assessment (PRA) required for a future plant be also considered as a tool to help uncover flooding and water intrusion ASIs.

Acceptance Criteria

The acceptance criterion for the resolution of USI A-17 is that attention shall be paid in the detailed plant design to detecting and minimizing the potential for ASIs. Future plants must encompass the full spectrum of potential system interactions from operating plant experience and new design evaluations. The objective is to preserve the means for reaching and maintaining a safe hot shutdown.

Resolution

The A-17 Systems Interaction issues/concerns have been long recognized as being critically important to safe and reliable plant design, operation and maintenance activities. A significant amount of sometimes splintered and sometimes coordinated efforts have taken place during the last ten years. These include:

(1) NRC Research Studies—A series of NUREG reports were directed at achieving a systematic way to address system interactions (e.g. Diagraphic approaches).

(2) EPRI Research Studies—A series of reports evaluated plant system dependencies and interactions, common mode or cause failure methodologies, operator error profiles, etc.

(3) Office for Analysis and Evaluation of Operational Data (AEOD) Studies—On a continuing basis, the AEOD has searched for common cause high frequency, high consequence, etc. events aspects. They have also addressed man-machine interactions. Their review of Licensing Event Report Evaluation (LERE) and event inspection evaluations have provided the data for their findings and conclusions.

(4) Nuclear Safety Analysis Corporation (NSAC) and Institute of Nuclear Power Operations (INPO) Operating Experience Feedback and Generic Safety Issue Tracking—For over 10 years these organizations have systematically and comprehensively evaluated LERs, significant events, maintenance and operator anomalies and other events of interest. Special emphasis is placed on operator error-plant anomalies aspects by INPO.

(5) NSSS—Vendors have tracked equipment performance and failure aspects for over 25 years. System interactions are an integral part of the root cause analysis in these evaluations.

(6) PRA Evaluations—A full spectrum of PRA analyses now exist for each plant. These analyses can be macro rather than micro in assessing ASIs. However, the recent expanded use and application of PRA for SSFIs, 10 CFR 50.59s, etc. demand detailed inquiries of SI efforts.

(7) NRC-Staff/Region Information—A wide spectrum of operating experience feedback is available in Generic Letters (GLs), Information Notices (INs), etc. In summary, a lot of information is available, distributed and utilized by the designer, the analyst, the safety evaluator and reviewer, and the plant operator. This information supplements and compliments the previous or original design basis insights. Most unique or important operating experience feedback is associated with system interaction elements.

Plant system designers have found a number of critically important prevention—mitigation—accommodation ASIs avoidance attributes. Several attribute groups are mentioned below for example purposes.

(1) Separation Criteria—Physical, Electrical, Mechanical, Environmental

(2) Kind/Type Criteria—Redundant, Diverse, Reliable and Available

(3) Failure Aspects—Fail As-Is, Fail Safe, Fail Recoverable, Single Active/Passive Failures, Common-Cause/Mode Failures, Fail Alarm

(4) Protective Action—Auto vs. Manual, Auto Reset, No Operator Involvement (Errors of Omission), Limited Operator Involvement (Errors of Commission)

(5) Maintenance Aspects—Limited Replacement or Repair and Replacement (R&R), Staggered Testing/Calibration/Inspection, One-on-One Signoff, Diverse Crews, No At-Power Maintenance

(6) Diverse Phenomena and Responses—Built-in Inherencies, Gravity-Driven Responses, Extended/Enhanced Capability, Time-Independent Responses, Non-Electro/Mechanical Response, Self-Powered Capabilities

The designer now has more evaluation tools to work with relative to ASIs. Detailed Failure Modes and Effects Analyses (FMEAs), Fault Tree Analyses (FTAs), Event Tree Analyses (ETAs), System Dependency Charts, Common-Mode Failures (CMFs), Common-Cause Failures (CCFs), Common-Mode Probabilistic Failures (CMPFs), etc. are but a few of the new tools available to use in ASI evaluations. Simulator and Emergency Operating Procedures (EOPs) audit critiques provide insights into man-machine aspects (e.g. operator error patterns, recovery alternatives, etc.). More precise root cause analysis techniques are available and demanded by regulatory requirements [e.g. Kepner Tregoe (KT), Event Sequence Plots, etc.]. PRA and operating experience feedback give the designer a feel for which component, sequence, or operator action is critical, sensitive, difficult, time related, etc.

The ABWR is very much like prior BWRs. Many BWR plant features are designed into the plant explicitly to avoid unwanted, unacceptable or unknown ASIs. The major items include:

- Utilize an Operating Experience Proven Design
- Multiple Fission Product Barriers
- Inherent Shutdown Features and Mechanisms
- Redundant and Diverse Containment Features
- Redundant and Diverse ESF Network

- Redundant and Diverse I&C Protection Network
- Redundant and Diverse Safe Shutdown Capabilities

ABWR unique features (in addition to other plants) to prevent, mitigate and accommodate ASIs include:

- More Redundant, Diverse and Independent Decay Heat Removal Systems (DHRSs) Capabilities (Subsections 5.4.7 and 6.3.2.2.4)
- More Redundant, Diverse and Independent RPV and Containment Make-up and Cooling Capabilities (Subsections 6.3.1.1.1 to 6.3.1.1.4)
- More Redundant, Diverse and Independent Power Sources (Subsections 8.1.2.1 and 8.1.2.2)
- More Redundant, Diverse and Independent Operator Action Capabilities (Subsection 18.4.2)
- More Redundant, Diverse and Independent and Fault-Tolerant I&C Protection Nature (Appendix 7C)
- More Secure and Protected ESF Housing from Fire and Flood Aspects (Appendices 19M and 19R, and Subsection 9.5.1)
- More Secure and yet accessible ESF Housing for Inspection and Maintenance (Subsection 19K.11.1)

The ABWR design utilizes most of the ASI avoidance attributes described and cited above. The ABWR has been extensively evaluated both deterministically and probabilistically. This design is based on over 25 years of successful operating experience. The plant designers had access to the extensive lessons learned and feedback over the ten years prior to initial ABWR certification. The design has been reviewed and evaluated over the 5 years prior to initial ABWR certification by the world's foremost safety experts (ACRS, NRC Staff, GE Staff, Utility Staffs, DOE and Consultants) for a spectrum (both broad and deep) of inquiry. The plant reflects proven technology and accepted design standards and requirements. The plant design addresses system interactions at three levels—prevention, mitigation and accommodation.

Therefore, this issue is resolved for the ABWR.

References

19B.2.59-1 NUREG-0933, “A Prioritization of Generic Safety Issues (with Supplements 1-15)”, U. S. NRC, April 1993.

19B.2.59-2 NUREG/CR-3922, “Survey and Evaluation of System Interaction Events and Sources”, U. S. NRC, January 1985.

19B.2.59-3 NUREG/CR-4261, “Assessment of System Interaction Experience at Nuclear Power Plants”, U. S. NRC, June 1986.

19B.2.59-4 NUREG-1299, “Regulatory Analysis for Resolution of USI A-17”, U. S. NRC, August 1989.

19B.2.59-5 NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants—LWR Edition”, U. S. NRC.

19B.2.59-6 NUREG-1174, “Evaluation of Systems Interaction in Nuclear Power Plants – Technical Findings Related to Unresolved Safety Issue A-17”, U. S. NRC, May 1989.

19B.2.60 A-29: Nuclear Power Plant Design for the Reduction of Vulnerability to Industrial Sabotage

Issue

Issue A-29 in NUREG-0933 (Reference 19B.2.60-1) addresses the susceptibility of nuclear power plants to industrial sabotage, the resulting risk to plant safety, and the countermeasures to assure an acceptable level of protection.

Consideration should be given to sabotage during the design phase of the plant. The goal would be to achieve an acceptable level of protection of a plant to industrial sabotage by emphasizing design features which reduce the likelihood of the plant incurring damage from industrial sabotage, both internal and external.

Acceptance Criteria

The acceptance criteria for the resolution of Issue A-29 is that plants shall be designed to be resistant to the effects of internal and external sabotage through prevention, deterrence and mitigation.

Specifically, plant safety-related systems and components required for the safe operation and shutdown of the plant shall be designed for protection against and mitigation of sabotage.

Resolution

The ABWR design will mitigate the acts of sabotage through physical separations in the plant arrangement of independent, engineered safety systems, and the design and location of barriers to resist threats (refer to Sections 9.5, 3.4, and 3.6).

Appendix 19C describes and analyzes the ABWR design features that reduce the risk from postulated insider sabotage.

In addition, the ABWR design includes various methods of access control to prevent intrusion as well as provide detection during a breach of the system. Specifically, Subsection 13.6.3 requires the COL applicant to describe the physical protection systems and controls for compliance with 10 CFR 73.55 (Reference 19B.2.60-2).

The design of the decay heat removal system provides an inherent resistance to sabotage by its protection against tornado missiles, winds, earthquakes and floods.

The COL applicant should confirm that the provisions of SECY-91-029 dealing with procedures for access will be in effect during cold shutdown.

In summary, the ABWR design is highly resistant to sabotage, because of the features described which protect against internal and external sabotage.

Therefore, this issue is resolved for the ABWR.

References

19B.2.60-1 NUREG-0933, “A Prioritization of Generic Safety Issues (with Supplements 1-15)”, U.S. NRC, April 1993.

19B.2.60-2 10 CFR 73.55, “Requirements for Physical Protection of Licensed Activities in Nuclear Power Reactors Against Radiological Sabotage”, Office of the Federal Register, National Archives Records Administration.

19B.2.61 B-5: Ductility of Two-Way Slabs and Shells and Buckling Behavior of Steel Containments

Issue

Generic Safety Issue (GSI) B-5 in NUREG-0933 (Reference 19B.2.61-1) identifies two concerns relating to containment design. First that sufficient information is not available to predict the behavior of two-way reinforced concrete slabs; and second, that the structural design of a steel containment vessel subjected to unsymmetrical dynamic loadings may be governed by the instability of the shell.

(1) Ductility of Two-Way Slabs and Shells

The first concern was originally identified in NUREG-0471 (Reference 19B.2.61-2) and involved concern over the lack of information related to the behavior of two-way reinforced concrete slabs loaded dynamically in biaxial membrane tension (resulting from in-plane loads), flexure, and shear. If structures (concrete slabs) were to fail (floor collapse or wall collapse) due to loading caused by a loss-of-coolant-accident (LOCA) or high-energy-line break (HELB), there would be a possibility that other portions of the reactor coolant system or safety-related systems could be damaged. Such loads would be caused by very concentrated high-energy sources causing direct impact on the structures of concern. The damage could lead to an accident sequence resulting in the release of radioactivity to the environment.

Because of NRC and industry concern, the American Concrete Institute addressed these dynamic loads by establishing the methodology identified in the Appendix C Commentary to ACI 349 (Reference 19B.2.61-3).

(2) Buckling Behavior of Steel Containments

The second concern, also identified in Reference 19B.2.61-2, involves concern over the lack of a uniform, well-defined approach for design evaluation of steel containments. The structural design of a steel containment vessel subjected to unsymmetrical dynamic pressure loadings may be governed by the instability of the shell. For this type of loading, the current design verification methods, analytical techniques, and the acceptance criteria may not be as comprehensive as they could be. Section III of the ASME Code (Reference 19B.2.61-4) does not provide detailed guidance on the treatment of buckling of steel containment vessels for such loading conditions.

Moreover, this Code does not address the asymmetrical nature of the containment shell due to the presence of equipment hatch openings and other penetrations. Regulatory Guide 1.57 recommends a minimum factor of safety of two against buckling for the worst loading condition provided a detailed rigorous analysis, considering in-elastic behavior, is performed.

On the other hand, the 1977 Summer Addendum of the ASME Code permits three alternate methods, but requires a factor of safety between 2 and 3 against buckling, depending upon applicable service limits.

However, NUREG-0933 states that the issue was resolved and no new requirements were established.

Acceptance Criteria

The acceptance criteria for part 1 of this issue is that the design of safety-related concrete structures shall meet the ductility requirements of ACI 349, as supplemented by RG 1.142 (Reference 19B.2.61-5).

The acceptance criteria for part 2 of this issue is that the buckling design of steel portions of containment vessels (i.e., ABWR reactor closure head) shall meet provisions of NE-3222 or code case N-284 of the ASME code.

Resolution

The design of ABWR safety-related concrete structures (other than containment) is based on ACI 349 as supplemented by RG 1.142.

Part 1 of this issue is thus resolved for the ABWR.

The ABWR containment is a reinforced concrete structure and it is designed according to ASME Code, Section III, Division 2, Subsection CC. The steel components (reactor closure head not backed by concrete) of the containment vessel are designed in accordance with to ASME Code, Section III, Subsection NE, including the buckling provisions as stated in the acceptance criteria above.

Part 2 of this issue is, thus, resolved for the ABWR.

References

19B.2.61-1 NUREG-0933, "A Prioritization of Generic Safety Issues (with Supplements 1-15)", U.S. NRC, April 1993.

19B.2.61-2 NUREG-0471, "Generic Task Problem Descriptions (Categories B, C, and D)", U.S. NRC, June 1978.

19B.2.61-3 ACI 349, "Code Requirements for Nuclear Safety Related Structures", American Concrete Institute.

19B.2.61-4 "ASME Boiler and Pressure Vessel Code", Section III (Nuclear), Division I, Subsection NE, American Society of Mechanical Engineers.

19B.2.61-5 Regulatory Guide 1.142, "Safety Related Concrete Structures for Nuclear Power Plants (Other than Reactor Vessels and Containments)", U.S. NRC.

19B.2.61.1 C-8: Main Steam Line Leakage Control Systems

Issue

Dose calculations indicated that operation of the main steam isolation valve leakage control system (MSIVLCS) required for some BWRs could result in higher offsite accident doses than if the system were not used and the integrity of the steam lines and condenser was maintained. The calculations for accidents with a TID-14844 (Reference 19B.2.61.1-2) source indicated a potential increase in offsite releases of iodine by two to three orders of magnitude for proper operation of a MSIVLCS, when compared to the calculations of releases assuming the steam system intact and MSIV leakage is eventually released through the condenser. Therefore, use of the MSIVLCS recommended in Regulatory Guide 1.96 (Reference 19B.2.61.1-3) could increase the overall risk to the public. After an extensive evaluation of alternative solutions, it was decided that Regulatory Guide 1.96 was acceptable, and the issue was resolved with no new requirements (Reference 19B.2.61.1-1).

Acceptance Criteria

This issue was resolved with no new requirements. However, the requirements of GDC 54 and the guidance of RG 1.96 are applicable to the ABWR. RG 1.96 describes a method of implementing GDC 54 with regard to design of a leakage control system for the MSIVs of

BWRs to ensure that total radiological effects do not exceed guidelines of 10 CFR 100 in the event of a postulated design basis LOCA. RG 1.96 states that the isolation function of the MSIVs should be supplemented by a leakage control system (LCS), or if an alternative method is used it must be approved by the NRC staff. RG 1.96 indicates that a leakage control system would not be required if the main steam line leakage path can be relied on to remain intact and capable of providing significant dose reduction factors for postulated accident conditions.

Resolution

The ABWR main steam line leakage path is designed to remain intact and capable of providing significant dose reduction factors for postulated accident conditions. The design of the ABWR main steam leakage path is described in Subsection 3.2.5.3. The main steam lines and all branch lines 65A (2-1/2 inches) in diameter and larger are designed to withstand the safe shutdown earthquake; the main steam and bypass lines at the turbine that are not safety-related, are analyzed to demonstrate their structural integrity under the safe shutdown earthquake loading. The condenser anchorage is seismically analyzed to demonstrate that it does not fail. The radiation results of the main steamline leakage analysis (Subsection 15.6.5) are given in Tables 15.6-13 and 15.6-14, for offsite and control room dose evaluations, respectively, and are within current regulatory guidelines. The COL applicant will recalculate iodine removal credit on the basis of the specific design characteristics of main steamlines, drains, and main condenser, as outlined in Subsections 15.6.5.5.1.2 and 15.6.5.5.1.3. The ABWR alternative design approach has been approved by the NRC staff.

Therefore, upon approval of the alternative design approach, this issue is resolved.

References

19B.2.61.1-1 NUREG-0933, “A Prioritization of Generic Safety Issues (with Supplements 1-15)”, U.S. NRC, April 1993.

19B.2.61.1-2 TID-14844, “Calculation of Distance Factors for Power and Test Reactor Sites”, U.S. Atomic Energy Commission, March 23, 1962.

19B.2.61.1-3 Regulatory Guide 1.96, “Design of Main Steam Isolation Valve Leakage Control Systems for Boiling Water Reactor Nuclear Power Plants”, U.S. NRC.

19B.2.62 29: Bolting Degradation or Failure in Nuclear Power Plants

Issue

Issue 29 in NUREG-0933 (Reference 19B.2.62-1) addresses bolting degradation within safety-related components and support structures and its impact on the integrity of the reactor coolant pressure boundary.

The most crucial bolting applications are those constituting an integral part of the primary pressure boundary such as closure studs and bolts on reactor vessels and reactor coolant pumps. Degradation of these bolts or studs could result in the loss of reactor coolant. Other bolting applications such as component support and embedment anchor bolts or studs are essential for withstanding transient loads created during abnormal or accident conditions.

Acceptance Criteria

The acceptance criteria for the resolution of Issue 29 are that proven bolting designs, materials, and fabrication techniques shall be employed. Reactor coolant pressure boundary (RCPB) bolting, component support bolts and embedment anchor bolts or studs shall meet the requirements of ASME Code, Section III; NUREG-1339; EPRI NP-5769; and GL 91-17 (References 19B.2.62-2, 19B.2.62-3, 19B.2.62-4 and 19B.2.62-5, respectively). Also, for RCPB bolting the owner-operator shall use established industry practice in developing maintenance, assembly, and disassembly procedures. Furthermore, for RCPB and its support bolting, inservice inspection shall meet the requirements of ASME Code, Section XI (Reference 19B.2.62-2).

Resolution

Bolting degradation of RCPB bolts is primarily an operating plant issue since most of the degraded bolts have resulted from poor maintenance practices. Bolting integrity is assured by the designer through the initial specification of proven bolting materials, installation requirements, and by the owner-operator through the use of acceptable maintenance and inspection practices.

For the ABWR design, only proven materials for the specific application and environment are employed, having been selected after evaluation of the potential for corrosion wastage and intergranular stress corrosion cracking. Also, the RCPB components and their integral bolts, including the reactor vessel, reactor coolant pumps and piping are fabricated, tested, and installed in accordance with ASME Code, Sections III and XI; and NUREG-1339, EPRI NP-5769 and GL 91-17 (References 19B.2.62-3, 19B.2.62-4 and 19B.2.62-5, respectively). Finally, the owner-operator must perform periodic inservice inspection in accordance with ASME Code, Section XI. In addition, for critical pressure boundary applications such as the reactor vessel head closure, redundant seals and leak monitoring further assure the integrity of the RCPB.

Therefore, this issue is resolved for the ABWR Standard Plant design.

References

19B.2.62-1 NUREG-0933, “A Prioritization of Generic Safety Issues (with Supplements 1-15)”, U.S. NRC, April 1993.

19B.2.62-2 “ASME Boiler and Pressure Vessel Code”, Section III (Nuclear) and Section XI, American Society of Mechanical Engineers.

19B.2.62-3 NUREG-1339, “Resolution of Generic Safety Issue 29: Bolting Degradation or Failure in Nuclear Power Plants”, U.S. NRC, June 1990.

19B.2.62-4 EPRI NP-5769, “Degradation and Failure of Bolting in Nuclear Power Plants, Electric Power Research Institute”, April 1988.

19B.2.62-5 Generic Letter 91-17, “Generic Safety Issue 29, ‘Bolting Degradation or Failure in Nuclear Power Plants ‘”, U.S. NRC, October 17, 1991.

19B.2.63 82: Beyond Design Basis Accidents In Spent-Fuel Pools

Issue

Issue 82 in NUREG-0933 (Reference 19B.2.63-1) addresses the potential for a beyond-design-basis accident in which the water is drained out of the spent-fuel pool. In such an event the discharged fuel from the last two refuelings may have sufficient decay heat to melt, ignite the zircaloy cladding and release fission products to the atmosphere.

Acceptance Criteria

The acceptance criteria for the resolution of Issue 82 is that the design of the spent-fuel pool, storage racks, fuel pool cooling and cleanup system and the load handling equipment in the spent-fuel pool area shall meet applicable current requirements, i.e., the guidance of the Standard Review Plan (SRP) Sections 9.1.2 – 9.1.5 (Reference 19B.2.63-2) and Regulatory Guide 1.13 (Reference 19B.2.63-3).

Resolution

The ABWR design includes a spent-fuel storage facility, a fuel pool cooling and cleanup system and a fuel handling system that meet the intent of Regulatory Guide 1.13 and SRP 9.1.2 - 9.1.5 as described in Subsection 9.1. A brief summary of the design features relating to the Regulatory Guide and the SRP follows.

- The spent-fuel pool and storage racks are Seismic Category I structures. The spent-fuel pool is in the Reactor Building which is also Seismic Category I. There are no non-seismic systems, high or moderate energy pipes, or rotating machinery located in the vicinity of the spent-fuel pool or cask loading area on the refueling floor.

- The Reactor Building protects the fuel and spent-fuel pool from tornadic winds and the missiles generated by these winds. The Reactor Building also prevents turbine missiles from effecting the spent-fuel pool.

- Interlocks prevent the movement of heavy loads over the spent-fuel pool. Heavy loads, defined such that if inadvertent operations or equipment malfunction either separately or in combination, could cause:

  (1) a release of radioactivity,

  (2) a criticality accident, or

  (3) the inability to cool fuel within the spent-fuel pool.

- The Standby Gas Treatment System limits the potential release of radioactive iodine and other radioactive materials from the Reactor Building which encloses the spent-fuel pool.
- The travel of the Reactor Building crane which handles heavy loads, including the fuel casks, is limited by interlocks to preclude movement over the spent-fuel storage pool.
- No inlets, outlets or drains are provided that might permit the pool to be drained below a safe shielding level. Lines extending below this level are equipped with siphon, breakers, check valves, or other suitable devices to prevent inadvertent pool drainage.
- A level switch is provided in the spent-fuel pool to alarm locally and in the control room on either high or low level. The Fuel Handling Area Ventilation Exhaust Radiation system monitors the offgas radiation level in the fuel handling area ventilation exhaust duct. A high-high radiation trip results in the initiation of the Standby Gas Treatment System and in the isolation of the secondary containment (including closure of the containment purge and vent valves, and closure of the Reactor Building ventilation exhaust isolation valves).
- The Fuel Pool Cooling and Cleanup (FPC) System provides the primary means of maintaining the water level in the spent-fuel pool utilizing a connection to the Condensate System. The Suppression Pool Cleanup (SPCU) System can be used as a backup. Both the FPC and SPCU Systems are Seismic Category I designs. Additionally, connections from the RHR System to the FPC System provide a Seismic Category I, safety-related makeup capability to the spent-fuel pool. The FPC System from the RHR connections to the spent-fuel pool are Seismic Category I and safety related. The RHR system connections will be protected from the effects of pipe whip, internal flooding, internally generated missiles, and the effects of a moderate pipe rupture. Furthermore, fire water can be used to supply water from the fire protection system to the spent-fuel pool via the RHR system or fire hoses.

Since the acceptance criteria are met for the spent-fuel storage facility, this issue is resolved for the ABWR.

References

19B.2.63-1 NUREG-0933, “A Prioritization of Generic Safety Issues (with Supplements 1-15),” U.S. NRC, April 1993.

19B.2.63-2 NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants—LWR Edition.”

19B.2.63-3 Regulatory Guide 1.13, “Design Objective for Light-Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations.”

19B.2.64 113: Dynamic Qualification Testing of large Bore Hydraulic Snubbers

Issue

Issue 113 in NUREG-0933 (Reference 19B.2.64-1) addresses the need for requirements for dynamic qualification testing of large bore hydraulic snubbers [>2.224E5 Newtons (>50 kips) load rating]. Qualification tests of large bore hydraulic snubbers typically utilize a shutoff valve in place of the snubber control valve. To assure operability of the snubber control valves when subjected to dynamic loads, testing should be performed to determine the operational characteristics of the snubber control valve.

Acceptance Criteria

The acceptance criteria for the resolution of Issue 113 for the ABWR design are the performance of dynamic tests in accordance with Subsection 3.9.3.4.1 (3). The dynamic load tests identified specifically for large bore hydraulic snubbers (LBHS) are to be performed in addition to the dynamic tests required for mechanical and hydraulic snubbers.

Resolution

Mechanical and hydraulic snubbers will only be used for piping systems when dynamic supports are required at locations where large thermal displacements prohibit the use of rigid supports.

Large bore hydraulic snubbers (LBHS) will only be used as piping restraints. Mechanical and hydraulic snubbers including LBHS are tested to insure that they can perform as required during seismic and other dynamic loading events. These tests are described in Subsection 3.9.3.4.1(3). Additional dynamic cyclic load tests are required for LBHS to assure operability of the snubber control valves when subjected to dynamic loads. This requirement is specified in Subsection 3.9.3.4.1(3)(C).

The acceptance criteria for this issue are met, therefore, the issue is resolved for the ABWR design.

References

19B.2.64-1 NUREG-0933, “A Prioritization of Generic Safety Issues (with Supplements 1-15)”, U.S. NRC, April 1993.

19B.2.65 I.D.5(2) Plant Status and Postaccident Monitoring

Issue

The issue addressed is documented in TMI Action Plan, and focuses on the need to improve the ability of nuclear power plant control room operators to prevent, diagnose, and properly respond to accidents by using the full information provided to them (Reference 19B.2.65-1).

Acceptance Criteria

The acceptance criteria for the resolution of Issue I.D.5(2) is that plant status and post-accident monitoring is in compliance with Regulatory Guide (RG) 1.97 (Reference 19B.2.65-2).

Resolution

The ABWR design of its information systems (important to safety) provide information for manual initiation and control of safety systems. These systems provide indication to the control room that plant safety functions are being accomplished and provide information from which appropriate actions can be taken to mitigate the consequences of anticipated operational occurrences and accidents. It is designed to perform as described in Subsection 7.5 and is in compliance with RG 1.97 (Reference 19B.2.65-2).

Therefore, this issue, I.D.5(2), is resolved for the ABWR.

References

19B.2.65-1 NUREG-0660, “NRC Action Plan Developed as a Result of the TMI-2 Accident”, U.S. NRC, May 1980.

19B.2.65-2 Regulatory Guide 1.97, “Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs condition During and Following an Accident”, U.S. NRC.

19B.2.66 I.D.5(3) On-Line Reactor Surveillance System

Issue

NUREG-0933 (Reference 19B.2.66-1), Generic Safety Issue (GSI) Item I.D.5(3) addresses the TMI issue of an “On-Line Reactor Surveillance System.” This issue specifically concerns detecting abnormal reactor core internal's noise associated with on-line reactor operation, e.g., detecting loose internal reactor parts.

Acceptance Criteria

The acceptance criteria for the resolution of GSI-I.D.5(3) is that, based on the on-going generic BWR programs, it is concluded that the technical resolution of this issue has been identified (see Reference 19B.2.66-1).

Resolution

The primary cause of core vibration is high and turbulent reactor water recirculation flow. To detect such vibration, the ABWR design incorporates a reactor vessel loose parts monitoring system (LPMS), as described in Subsection 4.4.4, that complies with NRC's Regulatory Guide 1.133 (Reference 19B.2.66-2) requirements. In addition, with the redesign for the ABWR reactor core internals, i.e., core fuel supports, fuel boxes and instrument channel's etc., problem reoccurrence has essentially been eliminated. The LPMS and other ABWR instrumentation systems will continue to monitor various reactor operational parameters, e.g., reactor core vibration, neutron flux patterns and stability; and thus, any problem recurrence would be quickly detected prior to any adverse core effects which might result. Furthermore, when compared to most other BWR's, the ABWR design incorporates ten small, rather than two large, reactor water recirculation pumps and these are in vessel type pumps. This arrangement is designed to more uniformly distribute core flow, and thus, reduce any flow turbulence that might lead to the loosening of reactor internal core parts.

Therefore, this issue, I.D.5(3), is resolved for the ABWR.

References

19B.2.66-1 NUREG-0933, “Resolution of Generic Safety Issues (with Supplements 1-33),” U.S. NRC, August 2010.

19B.2.66-2 Regulatory Guide 1.133, “Loose Parts Detection Program for the Primary System of Light-Water Cooled Reactors”, U.S. NRC.

19B.2.67 I.G.2: Scope of Test Program

Issue

The major thrust of TMI Action Plan I.G is to use the preoperational and startup test programs as a training exercise for the operating crews.

In contrast to this, Item I.G.2 calls for a more comprehensive test program to search for anomalies in a plant’s response to a transient. The safety significance of this issue lies in the early discovery of anomalies of unanticipated plant behavior. When a plant responds to a transient in an anomalous or unanticipated manner, the result may be an accident caused directly by the new phenomena, or by the surprise or confusion on the part of the operators (Reference 19B.2.67-1).

Acceptance Criteria

The acceptance criteria for the resolution of Issue I.G.2 is compliance with Standard Review Plan (SRP) Chapter 14 (Reference 19B.2.67-2), and Regulatory Guide 1.68 (Reference 19B.2.67-3).

Resolution

The ABWR will have a test program to evaluate and demonstrate, to the extent possible, that the operating group is knowledgeable about the plant and procedures and fully prepared to operate the facility in a safe manner as described in Chapter 14. Subsection 14.2.7 identifies Regulatory Guide 1.68 and other applicable regulatory guides used in the development of test programs.

Therefore, this issue is resolved for ABWR.

References

19B.2.67-1 NUREG-0933, “A Prioritization of Generic Safety Issues (with Supplements 1-15)”, U.S. NRC, April 1993.

19B.2.67-2 NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants—LWR Edition”, U.S. NRC.

19B.2.67-3 Regulatory Guide 1.68, “Initial Test Programs for Water-Cooled Nuclear Power Plants”, U.S. NRC.

19B.2.68 II.E.6.1: Test Adequacy Study

Issue

The purpose of this TMI Action Plan (Reference 19B.2.68-1) is to establish the adequacy of current requirements for safety-related valve testing. It recommends a study which would result in recommendations for alternate means of verifying performance requirements.

Acceptance Criteria

The acceptance criteria for the resolution of Issue II.E.6.1 include the following four parts:

(1) Investigation of in-situ testing of pressure isolation valves (PIVs) under Issue 105

(2) In-situ testing and surveillance of check valves will be performed to ensure their adequacy under design basis and required operating conditions

(3) Compliance with the thermal overload protection provisions of Regulatory Guide 1.106 (Reference 19B.2.68-4) for motor-operated valves (MOVs)

(4) Compliance with the recommendations of GL 89-10 (Reference 19B.2.68-2) for in-situ testing of motor-operated valves

Resolution

In-situ testing of PIVs, including check valves, is addressed in the resolution of Issue 105 (Subsection 19B.2.45). The COL applicant is to perform periodic surveillance and leak rate testing of PIVs per the ABWR Technical Specifications as part of the IST program.

With regard to in-situ testing and surveillance of safety-related check valves, Subsection 3.9.6.2.1 requires the COL applicant to perform in-situ full-flow testing, in addition to the ASME Code, Section XI, in-service testing requirements. Additionally, the COL applicant will use advanced non-intrusive techniques to assess valve degradation and performance. The COL applicant will also develop a program which establishes the frequency and extent of disassembly and inspection of check valves.

As indicated in Table 1.8-20, the ABWR will comply with the guidance of Regulatory Guide 1.106 (Reference 19B.2.68-4) regarding the application of thermal overload protection devices that are integral with the motor starter for electric motors on MOVs.

The COL applicant will need to address the concerns and issues identified in GL 89-10 (Reference 19B.2.68-2) for MOVs prior to plant startup (Subsection 3.9.6.2.2).

Valve performance is critical to the successful functioning of a large number of the plant safety systems. In-service testing of safety-related valves will be performed in accordance with the requirements of ASME/ANSI OMa-1988 Addenda to ASME/ANSI OM-1987, Parts 1, 6 and 10, as described in Subsection 3.9.6. Subsection 3.9.6 lists the in-service testing parameters and frequencies for the safety-related valves. The reason for each code defined testing exception or justification for each code exemption request is noted in the description of the affected valve. Valves having a containment isolation function are also noted in the listing.

Details of the in-service testing program, including test schedules and frequencies, will be reported in the in-service inspection and testing plan which will be provided by the applicant referencing the ABWR design. The plan will integrate the applicable test requirements for safety-related valves including those listed in the technical specifications (Chapter 16) and the containment isolation system. This plan will include baseline pre-service testing to support the periodic in-service testing of the components. Depending on the test results, the plan will provide a commitment to disassemble and inspect the safety-related valves when limits of the OM Code are exceeded. The primary elements of this plan, including the requirements of Generic Letter 89-10 (Reference 19B.2.68-2) for motor operated valves, are delineated in Subsection 3.9.6.

Therefore, this issue is resolved for the ABWR.

References

19B.2.68-1 NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident", U.S. NRC, May 1980, Revision 1, August 1980.

19B.2.68-2 NRC Letter to All Licensees of Operating Power Plants and Holders of Construction Permits for Nuclear Power Plants, "Safety-Related Motor-Operated Valve Testing and Surveillance (Generic Letter No. 89-10) – 10 CFR 50.54(f)", June 28, 1989.

19B.2.68-3 NUREG-0933, "A Prioritization of Generic Safety Issues (with Supplements 1-15)", U.S. NRC, April 1993.

19B.2.68-4 Regulatory Guide 1.106, "Thermal Overload Protection for Electric Motors on Motor-Operated Valves."

19B.2.69 II.K.1(13) Propose Technical Specification Changes Reflecting Implementation of all Bulletin Items

Issue

Issue II.K (Measures to Mitigate Small - Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents) has the objective of improving the capability to mitigate the consequences of small-break accidents and loss-of-feedwater events. Nine Inspection and Enforcement (IE) bulletins were issued to operating plants with twenty-eight requirements [Task II.K and Table C.1, NUREG-0660 (Reference 19B.2.69-1)] for reviews of plant design and operation.

Issue II K.1(13) is one of the twenty-eight requirements of the overall issue. It is directed at implementing the Technical Specification changes that would be required from other changes made to respond to all IE bulletin items.

Acceptance Criteria

The acceptance criteria for the resolution of Issue II.K.1(13) is compliance with 10 CFR 50.36, Technical Specifications (Reference 19B.2.69-2), and the interim "Proposed Policy Statement on Technical Specification Improvements for Nuclear Power" (Reference 19B.2.69-3).

Resolution

The ABWR demonstrates in Chapter 15, Accident Analysis, the capability to respond to the full spectrum of line breaks and loss-of-feedwater accidents without loss of containment or significant core damage. Chapter 16 sets forth the restrictions on plant operation required to control the transients and abnormal events of Chapter 15 to ensure conformance with the NRC rules identified in the Acceptance Criteria for this issue.

Accordingly, the analyses of Chapter 15 and the operational conditions and limitations of Chapter 16 ensure that the ABWR fulfills the intent of Issue II.K.1(13).

References

19B.2.69-1 NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident", U.S. NRC, May 1980.

19B.2.69-2 10 CFR 50.36, "Technical Specifications", Office of the Federal Register, National Archives and Records Administration

19B.2.69-3 Federal Register Notice 52FR3788, "Proposed Policy Statement on Technical Specification Improvements for Nuclear Power", February 1987.

19B.2.70 II.K.3(11): Control Use of PORV Supplied by Control Components, Inc. Until Further Revision Complete

Issue

Issue II.K, "Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents," has the objective of improving the capability to mitigate the consequences of small-break accidents and loss-of-feedwater events. For this issue, the Bulletins and Orders (B&O) Task Force conducted generic reviews of systems reliability, emergency procedures, and operator training as documented in NUREG-0626 (Reference 19B.2.70-2) and the NRC issued some 32 recommendations for the BWR [Task II.K and Table C.3, NUREG-0660 (Reference 19B.2.70-1)] for reviews of plant design and operations.

Issue II.K.3(11) is one of the 32 BWR recommendations of the Bulletins and Orders Task Force. It requires all plants to justify the use of PORVs (Power Operated Relief Valves) supplied by Control Components, Inc. that had failed during testing.

Acceptance Criteria

The acceptance criteria for the resolution of Issue II.K.3(11) is compliance with 10 CFR 50. Appendix A, General Design Criterion 15, “Reactor Coolant System Design”, and the applicable codes and standards governing safety/relief valves (SRV).

Resolution

The ABWR demonstrates in Chapter 15 the capability to respond to the full spectrum of line breaks and loss-of-feedwater accidents without loss of containment or significant core damage.

Section 5.2 describes the overpressure protection provided by the SRVs performing an overpressure relief valve function, an overpressure safety valve function, or an Automatic Depressurization system (ADS) function.

The SRV for the ABWR is not a Power Operated Relief Valve by Control Components, Inc. It is a spring-loaded safety valve for the safety valve function with a pneumatic cylinder/piston for power operation in the ADS and relief function.

Subsection 3.9.3.2.4.2 describes the qualification by type test of the SRVs to IEEE 344, “Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations”(Reference 19B.2.70-3), for operability during a dynamic event.

Therefore, this issue, II.K.3(11), is resolved for the ABWR.

References

19B.2.70-1 NUREG-0660, “NRC Action Plan Developed as a Result of the TMI-2 Accident”, U.S. NRC, May 1980.

19B.2.70-2 NUREG-0626, “Staff Report on the Generic Assessment of Feedwater Transients and Small Break Loss-of-Coolant Accidents in Boiling Water Reactors Designed by the General Electric Company”, U.S. NRC, January 1980.

19B.2.70-3 IEEE Standard 344, “Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations.”

19B.2.71 II.K.3(27): Provide Common Reference Level For Vessel Instrumentation

Issue

Issue II.K., “Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents,” has the objective to perform systems reliability and to effect changes in emergency operating procedures and operator training to improve the capability to mitigate such accidents.

The concern in Issue II.K.3(27) is that different reference points of the various reactor vessel water level instruments could cause operator confusion. Either the bottom of the vessel or the active fuel were considered to be reasonable reference points (Reference 19B.2.71-1).

Acceptance Criteria

The acceptance criteria for the resolution of Issue II.K.3(27) is to confirm that the ABWR design has a common zero reference for all water level indications.

Resolution

The resolution of this issue, II.K.3(27), for the ABWR is accomplished by setting a common reference for the reactor vessel water level at the top of the active fuel as shown on Figure 5.1-3 and as described in Section 7.7.

Therefore, this issue, II.K.3(27), is resolved for the ABWR.

References

19B.2.71-1 NUREG-0933, "A Prioritization of Generic Safety Issues (with Supplements 1-15)", U.S. NRC, April 1993.

19B.2.72 III.D.3.3(1): Issue Letter Requiring Improved Radiation Sampling Instrumentation

Issue

10 CFR Part 20 provides criteria for control of exposures of individuals to radiation in restricted areas, including airborne iodine. Since iodine concentrates in the thyroid gland, airborne concentrations must be known in order to evaluate the potential dose to the thyroid. If the airborne iodine concentration is overestimated, plant personnel may be required to perform operations functions while using respiratory equipment, which sharply limits communication capability and may diminish personnel performance during an accident. The purpose of this issue is to improve the accuracy of measurement of airborne iodine concentrations.

Acceptance Criteria

Airborne iodine concentrations must be accurately determined throughout the plant under accident conditions.

Resolution

Item III.D.3.3(1) which concerns in-plant radiation monitoring is resolved in Subsection 12.3.4 which also references each area detector location on the plant layout drawings for each building (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.

19B.3 COL License Information

19B.3.1 COL Applicant Safety Issues

The COL applicant shall provide resolutions for the issues identified as COL applicant in the Safety Issues Index consistant with the documentation format discussed in Subsection 19B.1.1.

19B.3.2 Testing of Isolators

As established in Section 7A.3, the COL applicant is required to establish a test program for fiber optic-type isolators used between safety-related and non-safety-related systems. If other types of isolators are used (those subject to electrical leakage due to maximum credible electrical faults), the COL applicant shall implement the required testing, inspection, and replacement isolators when needed (See Subsection 19B.2.53).

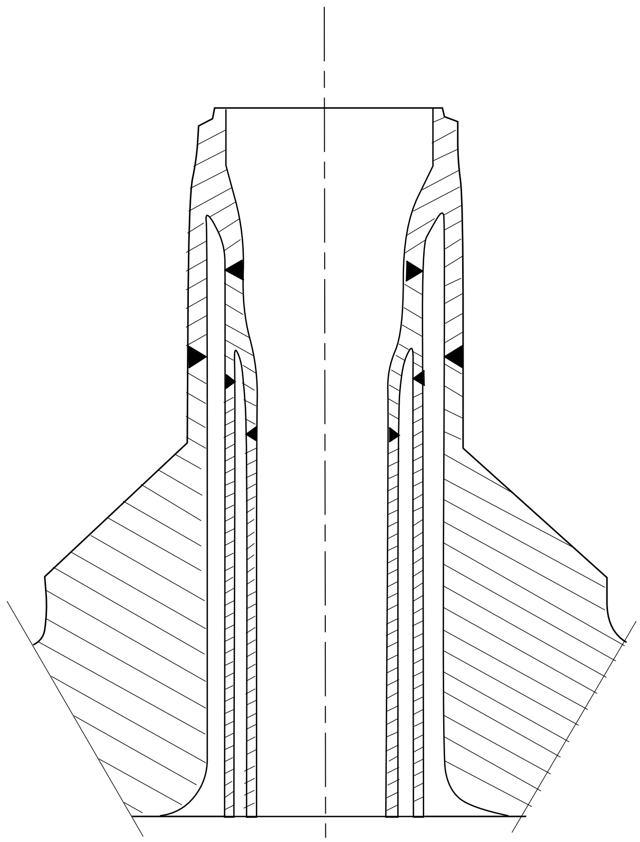

Figure 19B-1 Double Feedwater Nozzle-Thermal Sleeve