GE Nuclear Energy

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July 15, 1993

Docket No. STN 52-001

Chet Poslusny, Senior Project Manager
Standardization Project Directorate
Associate Directorate for Advanced Reactors
and License Renewal
Office of the Nuclear Reactor Regulation

Subject: Submittal Supporting Accelerated ABWR Schedule - **USIs/GSIs**

Dear Chet:

Enclosed are responses to NRC comments on USIs/GSIs including our conference call of July 9, 1993 and the addition of Issue C-8 to the issue group resolved with no new requirements.

Please provide a copy of this transmittal to Melinda Malloy.

Sincerely,

Jack Fox

Jack Fox
Advanced Reactor Programs

cc: Alan Beard (GE)
Norman Fletcher (DOE)
Bernie Genetti (GE)
Carl Szybalski (GE)

JF93-229

Table 19B.1-1

SAFETY ISSUES INDEX (Continued)

Title	NRC Priority	SSAR Subsection
New Generic Issues (Continued)		
45 Inoperability of Instrumentation Due to Extreme Cold Weather	Resolved	19B.2.34
51 Proposed Requirements for Improving the Reliability of Open Cycle Service Water Systems	Resolved	19B.2.35
57 Effects of Fire Protection System Actuation on Safety-Related Equipment	Resolved	19B.2.36
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75 Generic Implications of ATWS Events at the Salem Nuclear Plant	Resolved	19B.2.38
78 Monitoring of Fatigue Transient Limits for Reactor Coolant System	Resolved	19B.2.39
83 Control Room Habitability	Near Res.	19B.2.40
86 Long Range Plan for Dealing with Stress Corrosion Cracking in BWR Piping	Resolved	19B.2.41
87 Failure of HPCI Steam Line Without Isolation	Resolved	19B.2.42
89 Stiff Pipe Clamps	Medium	19B.2.43
103 Design for Probable Maximum Precipitation	Resolved	19B.2.44
105 Interfacing Systems LOCA at BWRs	High	19B.2.45
106 Piping and Use of Highly Combustible Gases in Vital Areas	Medium	19B.2.46
118 Tendon Anchorage Failure	Resolved	19B.2.48
120 On-Line Testability of Protection Systems	Medium	19B.2.49
121 Hydrogen Control for Large, Dry PWR Containments	Resolved	19B.2.50
124 Auxiliary Feedwater System Reliability	Resolved	19B.2.51
128 Electrical Power Reliability	Resolved	19B.2.52
142 Leakage Through Electrical Isolators in Instrumentation Circuits	Medium	19B.2.53
143 Availability of Chilled Water Systems and Room Cooling	High	19B.2.54
145 Actions to Reduce Common Cause Failures in BWRs	Resolved	19B.2.55
151 Reliability of Anticipated Transient without Scram Recirculation Pump Trip	Resolved	19B.2.56
153 Loss of Essential Service Water in LWRs	High	19B.2.57
155.1 More Realistic Source Term Assumptions	Near Res.	19B.2.58
Human Factors Issues		
HF.1.1 Shift Staffing	Resolved	COL App.
HF.4.4 Guidelines for Upgrading Other Procedures	High	COL App.
HF.5.1 Local Control Stations	High	COL App.
HF.5.2 Review Criteria for Human Factors Aspects of Advanced Control and Instrumentation	High	COL App.
Issues Resolved With No New Requirements		
A-17 Systems Interaction	Resolved	19B.2.59
A-29 Nuclear Power Plant Design for Reduction of Vulneralbility to Industrial Sabotage	Resolved	19B.2.60 COL App.

Table 19B.1-1

SAFETY ISSUES INDEX (Continued)

Title	NRC Priority	SSAR Subsection
Issues Resolved With No New Requirements (Continued)		
B-5 Ductility of Two-Way Slabs and Shells and Buckling Behavior of Steel Containments	Resolved	19B.2.61
C-8 Main Steamline Leakage Control Systems	Resolved	19B.2.61.1
29 Bolting Degradation or Failure in Nuclear Power Plants	Resolved	19B.2.62
82 Beyond Design BAses Accidents in Spent Fuel Pools	Resolved	19B.2.63
113 Dynamic Qualification Testing of Large Bore Hydraulic Snubbers	Resolved	19B.2.64
TMI Issues		
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	COL App.
I.A.1.4 Long-Term Upgrading	Resolved	COL App.
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	COL App.
I.A.3.1 Revise Scope of Criteria for Licensing Examinations	Resolved	COL App.
I.A.4.1(2) Interim Changes in Training Simulators	Resolved	COL App.
I.A.4.2(1) Research on Training Simulators	Resolved	19A.2.13
I.A.4.2(2) Upgrade Training Simulator Standards	Resolved	19A.2.13
I.A.4.2(3) Regulatory Guide on Training Simulators	Resolved	19A.2.13
I.A.4.2(4) Review Simulators for Conformance to Criteria	Resolved	19A.2.13
I.C.1(1) Small-Break LOCAs	Resolved	COL App.
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	19.A.2.41
I.C.6 Procedures for Verification of Correct Performance of Operating Activities	Resolved	COL App.
I.C.7 NSSS Vendor Review of Procedures	Resolved	COL App.
I.C.8 Pilot-Monitoring of Selected Emergency Procedures for Near-Term Operating License Applicants	Resolved	COL App.
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	Medium	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	Near Res.	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

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 10 inches to 12 inches. Although most cracks ranges from 1/2 inch to 3/4 inch in depth (including cladding), one crack penetrated the cladding into the base metal for a total depth of approximately 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 a unresolved safety issue (USI) (References 1 and 2).

ACCEPTANCE CRITERIA

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

Bypass leakage past the junction of the thermal sleeve and nozzle safe end is the primary source of cold water impinging upon 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. 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 cooled water that may be shedding from the outside surface of the inner sleeve from impinging on the nozzle bore and the inside nozzle corner.

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 in Table 2 of NUREG 0619. The closest design mentioned in Table 2 may be "Welded, clad removed (spargers have top-mounted elbows)."

No leakage between thermal sleeve and nozzle, no nozzle cladding, double thermal sleeve and excellent field experience with welded thermal sleeves justify a routine ISI program by ultrasonic inspection of the inside nozzle blend radius section.

The frequency of the planned ISI program is as outlined in Section XI of the ASME Code; e.g., 100% over 10 years.

As a design feature, the triple feedwater nozzle sleeve was considered less desirable than the double sleeve arrangement for ABWR because the triple sleeve allows more cold water to impinge on the nozzle by leakage past two sets of cylinder ring seals and there is also greater risk of complete failure of the cylinder ring seals. The greater leakage past these seals causes greater thermal stress cycling of the nozzle.

REFERENCE

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

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

19B.2.14 A-40: SEISMIC DESIGN CRITERIA SHORT-TERM PROGRAM

ISSUE

Issue A-40 in NUREG-0933 (Reference 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 2), where justified.

Studies were conducted, and the results were documented in NUREG/CR-1161 (Reference 3) with specific recommendations for changes in seismic design requirements. In addition, a 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 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 5). This issue is therefore resolved for the ABWR standard design.

REFERENCES

1. NUREG-0933, "A Prioritization of Generic Safety Issues" (with supplements), U.S. NRC, July 1991.

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

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

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

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

19B.2.22 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, and (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 1) requires that the performance of these valves be under continual surveillance and the consequences of their failures be subject to review.

RESOLUTION

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 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 the BWRs with Target Rock pilot operated safety/relief valves. GE has identified the principal cause of the most significant concern with these Target Rock pilot operated safety/relief valve and has developed a modification to greatly improve the performance of this valve model.

REFERENCES

1. NUREG-0933, "A Prioritization of Generic Safety Issues" (with supplements), U.S. NRC, July 1991.

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.2.39 78: MONITORING OF FATIGUE TRANSIENT LIMITS FOR REACTOR COOLANT SYSTEM

ISSUE

Generic Safety Issue (GSI) 78 in NUREG-0933 (Reference 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 Section III (Reference 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 III Class 1 carbon steel piping be considered in accordance with Section 3.9.3.1.1.7.

RESOLUTION

For the ABWR, Technical Specification 5.7.2.9 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 design.

REFERENCES

1. NUREG-0933, "A Prioritization of Generic Safety Issues" (with supplements), U.S. NRC, July 1991.

2. American Society of Mechanical Engineering Boiler and Pressure Vessel Code, Section III

19B.2.43 89: STIFF PIPE CLAMPS

ISSUE

Issue 89 in NUREG-0933 (Reference 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 Section III Code (Reference 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 are that pipe clamps shall only be installed on straight runs of pipe or on bends with a radius of at least five pipe diameters.

In the 1980's, GE performed calculations for typical stiff pipe clamps used on BWR Main Steam and Recirculation piping systems. The purpose of these calculations was to evaluate the additional stresses at clamp locations due to internal pressure, thermal expansion and dynamic loads. These calculations demonstrated that even with the increase in pipe stresses at the clamp locations, the stresses at these locations were still less than the peak piping system stresses. The stress intensification that occurs at elbows, branch connections and lugs is much greater than that which occurs at stiff pipe clamps on essentially straight runs of pipe.

RESOLUTION

For the ABWR, the piping design specifications require that stiff pipe clamps be installed on straight runs of pipe or on bends with radius of at least five pipe diameters. Since the ABWR will utilize stiff pipe clamps similar to those evaluated in the calculations summarized above, the magnitude of the piping stress increases at the clamp locations will be approximately equal to those calculated for BWR Main Steam and Recirculation piping systems. Therefore, the pipe clamp induced stresses can be considered negligible and do not warrant explicit consideration. This issue 89 is resolved for the ABWR.

REFERENCES

1. NUREG-0933, "A Prioritization of Generic Safety Issues" (with supplements), U.S. NRC, July 1991.

2. American Society of Mechanical Engineering Boiler and Pressure Vessel Code, Section III

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 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 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 349-85 (Reference 3).

(2) Buckling Behavior of Steel Containments

The second concern, also identified in Reference 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 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 endorsed by RG 1.142 (Reference 5).

The acceptance criteria for part 2 of this issue is that the buckling design of steel containment vessels 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 the latest edition of ACI 349 (1990). Part 1 of this issue is thus resolved for the ABWR.

19B 2.61 B-5

The ABWR containment is a reinforced concrete structure and it is designed according to ASME-III, Division 2, Subsection CC. The steel components (not backed by concrete) of the containment vessel are designed in accordance with to ASME-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

1. NUREG-0933, "A Prioritization of Generic Safety Issues", (with Supplements) U.S. NRC, July 1991.

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

3. ACI 349-85, "Code Requirements for Nuclear Safety Related Structures", American Concrete Institute, 1985.

4. ASME Boiler and Pressure Vessel Code, Section III, Division I, Subsection NE, American Society of Mechanical Engineers, 1986.

5. Regulatory Guide 1.142, "Safety Related Concrete Structures for Nuclear Power Plants (Other than Reactor Vessels and Containments) U.S. NRC, October 1981, Revision 1.

NRC Comments 07/09

AC1 349-85 is old; use ASME III Div. 2

Be explicit in statement on buckling.

19.B.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 (MSIVCS) 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 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 prescribed by Regulatory Guide 1.96 (Reference 3) could increase the overall risk to the public. After an extensive evaluation of alternative solutions, it was decided to take no action, and the issue was resolved with no new requirements. (Reference 1.)

ACCEPTANCE CRITERIA

The acceptance criteria for the resolution of issue C-8 is to provide a safe path (remains functional under a safe shutdown earthquake) for the main steam leakage through the main steam piping, bypass line, and condenser to hold up and plate out the release of fission products following postulated core damage.

RESOLUTION

The design of the ABWR main steam leakage path is described in Subsection 3.2.5.3, "Main Steam Line Leakage Path." The main steam lines and all branch lines 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. Therefore, this issue C-8 is resolved for the ABWR.

REFERENCES

1. NUREG-0933, "A Prioritization of Generic Safety Issues," (with Supplements) U.S. NRC, July, 1991.

2. TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," U.S. Atomic Energy Commission, March 23, 1962.

3. Regulatory Guide 1.96, "Design of Main Steam Isolation Valve Leakage Control Systems for Boiling Water Reactor Nuclear Power Plants," U.S. NRC, June 1976.

19B.2.62.29: BOLTING DEGRADATION OR FAILURE IN NUCLEAR POWER PLANTS

ISSUE

Issue 29 in NUREG-0933 (Reference 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 (References 2, 3, 4 and 5). 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, Section XI (References 2, 3, 4 and 5).

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 References 3, 4 and 5. Finally, the owner-operator must perform periodic inservice inspection in accordance with ASME Code Section XI and References 3, 4 and 5. 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 Design.

REFERENCES

1. NUREG-0933, "A Prioritization of Generic Safety Issues" (with Supplements), U.S. NRC, July 1991

2. American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section III (Nuclear) and Section XI, American Society of Mechanical Engineers.

3. NUREG-1339, "Resolution of Generic Safety Issue 29: Bolting Degradation or Failure in Nuclear Power Plants," U.S. NRC, June 1990.

4. EPRI NP-5769, "Degradation and Failure of Bolting in Nuclear Power Plants."

5. Generic Letter 91-17, "Generic Safety Issue 29, 'Bolting Degradation or Failure in Nuclear Power Plants,'" U.S. NRC.

19B.2.64 113: DYNAMIC QUALIFICATION TESTING OF LARGE BORE HYDRAULIC SNUBBERS

ISSUE

Issue 113 in NUREG-0933 (Reference 1), addresses the need for requirements for dynamic qualification testing of large bore hydraulic snubbers (>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 SSAR Section 3.9.3.4.1 (3).

RESOLUTION

For the ABWR design, large bore hydraulic snubbers (LBHS) will not be used as extensively as in some operating plants. The ABWR design will have less than one quarter of the number of snubbers at a typical BWR operating plant. LBHS's will only be used for piping systems when dynamic supports are required at locations where large thermal displacements prohibit the use of rigid supports. When LBHS's are used the required dynamic tests will be performed to confirm the operational characteristics of the snubber control valve.

REFERENCES

1. NUREG-0933, "A Prioritization of Generic Safety Issues" (with supplements), U.S. NRC, December 1992.