GE Nuclear Energy

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April 30, 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 Review Schedule - **USIs and GSIs**

Dear Chet:

Enclosed is a draft replacement for Appendix 19B, "Assessment of Applicable Unresolved Safety Issues and Generic Safety Issues," which addresses Open Items 20.1-1 and 20.2-1.

Please provide a copy of this transmittal to Melinda Malloy.

Sincerely,

Jack Fox

Jack Fox
Advanced Reactor Programs

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

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OPEN 20.1-1
OPEN 20.1-2

19B.1 INTRODUCTION

19B.1.1 Purpose

INSERT 19B.1.1

The NRC generic licensing, TMI and new generic safety issues in NUREG-0933 (Reference 1) and associated correspondence were reviewed and evaluated for the ABWR. The unresolved issues for the ABWR were identified from the issues applicable to BWR design, and at the beginning of GFY 1989, the NRC resolution was not fully complete in documentation or distribution. Status of the issues from the NRC (Reference 2) and EPRI were reviewed. Issues superceded, dropped, resolved, regulatory impact or PWR design specific were excluded. The unresolved safety issues requiring ABWR resolution according to the NRC severe accident policy are identified in Table 19B.1-1.

Generic safety issues (GSI) and unresolved safety issues (USI) of concern to the NRC for reviewing ALWR designs are listed in Table 19B.1-2. A set of GSIs and USIs was provided ~~in Enclosure~~ 2 of the Request for Additional Information (RAI), Reference 20.4.17 in the following categories: deleted

(1) High and Medium Priority Unresolved GSIs.

(2) Nearly-Resolved GSIs

(3) Resolved GSIs and USIs for which guidance has been issued in the form of Generic Letters, NUREGs, Reports, Bulletins, etc.

(4) GSIs and USIs for which resolution has resulted in the issuance of a rule, or the development of a standard review plan revision, regulatory guide, or revision to a regulatory guide.

A second set of GSIs and USIs identified in NUREG-0933 Appendix B as resolved was also provided by Reference 20.4.17. A third NRC set of GSIs and USIs, Beyond Appendix B to NUREG-0933, is also listed in Table 19B.1-1. The table includes remarks for reference to the ABWR.

19B.1.2 Summary

The ALWR issue description summary and resolution summary are included herein to maintain BWR consistency in the ABWR. Also repeated are the pertinent requirements in the EPRI-ALWR Requirements Document (Reference 3) identified in the topic papers. Safety issue resolutions from the ALWR requirements are superceded by the NRC resolutions as the latter are developed. The ABWR design has been compared to the NRC or ALWR resolution requirements with the resulting evaluation leading to resolution of the safety issues for the ABWR.

19B.1.3 References

(1) *A Prioritization of Generic Safety Issues*, NUREG-0933, Including Supplement 7 dated March 1988.

(2) *Generic Issue Management Control System* - Fourth Quarter FY-~~90~~ 93 Update, Memorandum for James M. Taylor from E.S. Beckjord dated November 3, 1989 March 30, 1993

(3) *Advanced Light Water Reactor Utility Requirements Document*, Electric Power Institute, Advanced LWR Program.

Insert 19B.1.1

The ABWR has proposed technical resolutions of those Uresolved Safety Issues (USI) and medium and high priority Generic Safety Issues (GSI) which are identified in the version of NUREG-0933 (Reference 1) current on the date six months prior to the ABWR application and which are technically relevant to the ABWR design in accordance with 10 CFR 52.47(a)(iv). NUREG-0933 and associated correspondence (References 2 & 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 satisfying severe accident requirements are addressed in Subsection 19B.2.

The following guidelines were use 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 five NRC selected issues.
(4) Resolved with no new requirements except for six 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 Table 19B.1-1 and are evaluated in the referenced Subsection. The COL applicant will evaluate those issues referencing the COL applicant in accordance with Subsection 19B.3.

The documentation of the issue evaluation is comprised of four sections: ISSUE, ACCEPTANCE CRITERIA, RESOLUTION and REFERENCES. The ISSUE statement is a brief summary description of the issue. The ACCEPTANCE CRITERIA are taken from NUREG-0933 and GIMCS (Reference 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 the SSAR.

Table 19B.1-1

SAFETY ISSUES INDEX

Title	NRC Priority	SSAR Subsection
Generic Issues		
A-1 Water Hammer	Resolved	19B.2.2
A-7 Mark I Long-Term Program	Resolved	19B.2.3
A-8 Mark I Containment Pool Dynamic Loads - Long Term Program	Resolved	19B.2.4
A-9 ATWS	Resolved	19B.2.5
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
A-25 Non-Safety Loads on Class 1E Power Sources	Resolved	19B.2.9
A-31 RHR Shutdown Requirements	Resolved	19B.2.10
A-35 Adequacy of Offsite Power Systems	Resolved	19B.2.11
A-36 Control of Heavy Loads Near Spent Fuel	Resolved	19B.2.12
A-39 Determination of Safety Relief Valve Pool Dynamic Loads and Temperature Limits	Resolved	19B.2.13
A-40 Seismic Design Criteria - Short Term Program	Resolved	19B.2.14
A-42 Pipe Cracks in Boiling Water Reactors	Resolved	19B.2.15
A-44 Station Blackout	Resolved	19B.2.16
A-47 Safety Implications of Control Systems	Resolved	19B.2.17
A-48 Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment	Resolved	19B.2.18
B-10 Behavior of BWR Mark III Containments	Resolved	19B.2.19
B-17 Criteria for Safety-Related Operator Actions	Resolved	COL App.
B-36 Develop Design, Testing and Maintenance Criteria for Atmospheric Cleanup System Air Filtration and Absorption Units for Engineered Safety Feature Systems and for Normal Ventilation Systems	Resolved	19B.2.21
B-55 Improved Reliability of Target Rock Safety Relief Valves	Resolved	19B.2.22
B-56 Diesel Reliability	Resolved	19B.2.23
B-61 Allowable ECCS Equipment Outage Periods	Resolved	19B.2.24
B-63 Isolation of Low Pressure Systems Connected to the Reactor Coolant pressure Boundary	Resolved	19B.2.25
B-66 Control Room Infiltration Measurements	Resolved	19B.2.26
C-1 Assurance of Continuous Long Term Capability of Hermetic Seals on Instrumentation and Electrical Equipment	Resolved	19B.2.27
C-10 Effective Operation of Containment Sprays in a LOCA	Resolved	19B.2.28
C-17 Interim Acceptance Criteria for Solidification Agents for Radioactive Solid Wastes	Resolved	19B.2.29
New Generic Issues		
15 Radiation Effects on Reactor Vessel Supports	High	19B.2.30
23 Reactor Coolant Pump Seal Failures	High	19B.2.31
25 Automatic Air Header Dump on BWR Scram System	Resolved	19B.2.32
40 Safety Concerns Associated with Pipe Breaks in the BWR Scram System	Resolved	19B.2.33

Table 19B.1-1

SAFETY ISSUES INDEX (Continued)

Title	NRC Priority	SSAR Subsection
New Generic Issues <u>(Continued)</u>		
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
67.3.3 Improved Accident Monitoring	Resolved	19B.2.37
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	Near Res.	19B.2.41
87 Failure of HPCI Steam Line Without Isolation	Near Res.	19B.2.42
89 Stiff Pipe Clamps	Medium	19B.2.43
103 Design for Probable Maximum Precipitation	High	19B.2.44
105 Interfacing Systems LOCA at BWRs	Resolved	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	High	19B.2.54
* 145 Improve Surveillance and Startup Testing Programs	Resolved	19B.2.55
151 Reliability of Recirculation Pump Trip During an ATWS	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
<u>Human Factors Issues</u>		
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.
<u>Issues Resolved With No New Requirements</u>		
A-17 Systems Interaction	Resolved	19B.2.59
A-29 Plant Design for Reduction of Vulneralbility to Sabotage	Resolved	19B.2.60
B-5 Ductility of Two-Way Slabs and Shells and Buckling Behavior of Steel Containments	Resolved	19B.2.61

* Awaiting NRC input.

Table 19B.1-1

SAFETY ISSUES INDEX (Continued)

Title	NRC Priority	SSAR Subsection
New Generic Issues (Continued)		
Issues Resolved With No New Requirements (Continued)		
29 Bolting Degradation or Failures in Nuclear 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

19B.2.2 A-1: WATER HAMMER

ISSUE

Unresolved Safety Issue (USI) A-01 in NUREG-0933 (Reference 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 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 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

The acceptance criterion for the resolution of USI A-01 is that safety-related process fluid systems shall be designed to conform to the requirements of 10CFR50 Appendix A, GDC 4 (Reference 3), by implementing the guide lines identified in SRP (Reference 4) Sections 5.4.7, 6.3, 9.2.1, 9.2.2, 10.3, and 10.4.7 (including BTP ASB 10-2).

The following systems shall be designed to include features minimizing the probability of "water hammer" occurrences as well as features to withstand the adverse dynamic loads imposed by "water hammer" events: Condensate and Feedwater System, Main Steam System, Emergency Core Cooling System (ECCS), Residual Heat Removal (RHR) Containment Spray Subsystem, RHR Shutdown Cooling Subsystem, Reactor Building Cooling Water (RCW) System, HVAC Normal Cooling Water (HNCW) System, HVAC Emergency Cooling Water (HECW) System, and the Turbine Building Cooling Water (TCW) System.

Operating and Maintenance procedures shall include adequate precautions to minimize the potential occurrence of "water hammer".

RESOLUTION

The ABWR SSAR Standard Design adequately addresses system dynamic loads such as may result from "water hammer". The entire Condensate and Feedwater System piping is analyzed for water hammer loads that could potentially result from anticipated flow transients (see Section 10.4.7.3).

The main steam, feedwater, and associated drain lines are protected from potential damage due to fluid jets, missiles, reaction forces, pressures, and temperatures resulting from pipe breaks as well as analyzed for dynamic loadings due to fast closure of the turbine stop valves (see Section 5.4.9).

Condensate-induced Water Hammer (CIWH) phenomenon can potentially occur in BWR ECCS piping during transient and/or accident conditions involving reactor depressurization. A General Electric study and analysis was made to review the ABWR RHR System, the High Pressure Core Flooder (HPCF) System, and the Reactor Core Isolation Cooling (RCIC) System piping configuration from the point of view of CIWH. Based upon this analysis, it was determined that since CIWH can only occur in piping used to inject cold water into the reactor, the HPCF, RHR-LPFL (Low Pressure Flooder), RCIC, and Feedwater (FW) injection piping was reviewed. It was concluded that the HPCF and RHR-LPFL mode in the LOCA event may have a potential for water hammer. This was concluded after analyzing the fluid conditions inside the reactor pressure vessel (RPV) at the corresponding nozzle height. However, detailed study of this piping demonstrated that CIWH would not occur. The effect of depressurization transients during a LOCA (flashing of saturated water in completely filled upstream LPFL pipe) was analyzed in order to calculate the quantity of water left in the pipe when the pressure drops to 250 psia. It was found that about 80% of water still remains in the pipe. 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. In the HPCF System, the presence

of two-phase mixture or saturated water at the nozzle height and in the piping inside the RPV avoids the occurrence of high water hammer loads. In the RCIC and FW Systems, the water level in the reactor is above the nozzle level at the time of system initiation, and therefore, there is no CIWH. The overall conclusion is that the proposed ABWR injection piping configuration is not susceptible to CIWH.

Dynamic loads caused by condensation-induced "water hammer" as well as piping arrangements and drainage provision protection against water entrainment are adequately addressed by the ABWR SSAR Main Steam System (see Section 10.3.3).

Dynamic loads, and the provisions of vents and drains (where appropriate) are also addressed in the design of the following systems: RHR Shutdown Cooling System, ECCS, RHR Containment Spray System, RCW System, HNCW System, HECW System, and the TCW System (see Sections 5.4.7, 6.3.1, 5.4.7.1, 9.2.11.1, 9.2.11, 9.2.12, 9.2.13, and 9.2.14 respectively).

Plant Operating and Maintenance Procedures are prepared by the COL applicant (in accordance with General Electric guidelines), and require proper precautions to minimize "water hammer" potential.

Pre-operational Tests are established to be performed by the COL applicant which include the guidelines of BTP ASB 10-2. These tests verify that unacceptable "water hammer" does not occur in the following systems: RHR System, RCIC System, HPCF System, RCW System, HECW System, HNCW System, Condensate and Feedwater System, and the TCW System (see Sections 14.2.12.1.8, 14.2.12.1.9, 14.2.12.1.10, 14.2.12.1.29, 14.2.12.1.32, 14.2.12.1.33, 14.2.12.1.53, and 14.2.12.1.63 respectively).

Since the design and testing of the safety systems potentially subject to "water hammer" meet the intent of the acceptance criteria above, this issue is resolved for the ABWR SSAR Standard Design.

REFERENCES

1. NUREG-0933, "A Status Report on Unresolved Safety Issues", U.S. NRC, April 1989.

2. NUREG-0927, Revision 1, "Evaluation of Water Hammer Occurrences in Nuclear Power Plants", U.S. NRC, April 1984.

3. 10CFR50 Appendix A, "General Design Criteria for Nuclear Power Plants", Code of Federal Regulations, Office of the Federal Register, National Archives and Records Administration.

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

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 7). 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 6) 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 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 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 NRC staff has concluded that the improved load definition submitted by the Mark I Owners Group for downcomer condensation oscillation loads is acceptable. In addition, the staff has concluded that the load specification associated with the confirmatory experimental and analytical programs has been justified.

This USI was RESOLVED (Reference 3) in August 1982 with the issuance of Supplement 1 to NUREG-0661 (Reference 1) and SRP (Reference 2) Section 6.2.1.1C. The load definition methodology used for the ABWR containment design is similar to that used for prior BWR containment designs. Wherever the ABWR unique design features warranted need for additional information for defining ABWR design loads, ABWR unique analyses and tests were conducted to provide an adequate data base for defining the pertinent hydrodynamic loads (Reference 4). Therefore this issue is resolved for the ABWR.

REFERENCES

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

2. NUREG-0800, "Standard Review Plan," U.S. NRC.

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

4. ABWR SSAR Section 3B.1 ABWR Containment Design.

5. NUREG-0933, "A Prioritization of Generic Safety Issues," July 1991 (and Supplements 1-12).

6. Task Action Plans for Generic Activities Category A, 1978.

7. Mark I Containment Short-Term Program, 1978.

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. This item was originally identified in NUREG-0371 (Reference 5) 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 1, Appendix A). 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 load definition methodology for defining the LOCA pool swell loads, LOCA steam condensation oscillation loads, and LOCA chugging loads on submerged structures for the ABWR will be consistent with the methodology used for prior plants (Reference 3). This USI was RESOLVED in August 1981 with the issuance of NUREG-0808 (Reference 1) and SRP Section 6.2.1.1C (Reference 2).

REFERENCES

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

2. NUREG-0800, "Standard Review Plan," U.S. NRC.

3. ABWR SSAR Section 3B.5: Submerged Structure Loads.

4. NUREG-0933, "A Prioritization of Generic Safety Issues," July 1991 (and Supplements 1-12).

5. Task Action Plans for Generic Activities Category A, 1978.

19B.2.5 A-9: ANTICIPATED TRANSIENTS WITHOUT SCRAM (ATWS)

ISSUE

This issue A-9 (Reference 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 (Reference 1). The technical report on ATWS (WASH-1270) (Reference 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, 10CFR50.62 (Reference 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 10CFR50.62 after incurring an anticipated transient and a failure to scram. Specifically, 10CFR50.62 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 10CFR50.62 to resolve this issue for the ABWR.

REFERENCES

1. NUREG-0933, "A Prioritization of Generic Safety Issues" (with Supplements 1-12) U.S. NRC, July 1991.

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

3. 10CFR50.62, "Requirements for Reduction of Risk from Anticipated Transients Without Scram (ATWS) Events for Light-Water-Cooled Nuclear Power Plants."

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 steam separators and dryers and the incoming cold feedwater causes thermal stress cycling of 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 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 impinge on the nozzle bore and the inside nozzle corner.

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.

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.2.7 A-13: SNUBBER OPERABILITY ASSURANCE

ISSUE

Generic Safety Issue (GSI) A-13 in NUREG-0933 (Reference 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; and,
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 (LER's), concerning snubber operability, were issued by utilities. A review of these LER's 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 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 3) as described in Section 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 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

1. NUREG-0933, "A Prioritization of Generic Safety Issues" (with Supplements 1-12), U.S. NRC.

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

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 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 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 3). In a subsequent rulemaking, 10 CFR 50.49 (Reference 4) established the requirement for an environmental qualification program for Class 1E electrical equipment together with rules for its content. References 2 and 3 comprise the bases for the rules. Regulatory Guide 1.89 was then revised (Reference 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 6).

ACCEPTANCE CRITERIA

The acceptance criteria for the resolution of issue A-24 are 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

Environmental design and qualification are described in Section 3.11. The Class 1E electrical equipment (including pump and valve motors and electrical accessories) of the ABWR is environmentally qualified by the methods documented in the NRC-approved report NEDE-24326-1-P (Reference 7). These methods are in accordance with the guidance of IEEE Std 323-1974 (Reference 8), NUREG-0588, Regulatory Guide 1.89 Revision 1, and the generic requirements of 10 CFR 50.49 as described in Section 3.11.

Equipment required to mitigate the consequences of a design basis accident (DBA) or to attain a safe shutdown of the reactor is identified in Section 6.3 for emergency core cooling function and in Appendix 3I.3.2 for typical normal and accident environments in that location including integrated radiation doses.

Typical environment conditions (temperature, pressure, humidity, integrated radiation dose, and exposure to chemicals) are given in Appendix 3I.3.2 and cover the design lifetime. Conditions are tabulated for normal operation in and outside of containment, and for loss-of-coolant accident (LOCA) and high energy line break (HELB) inside containment.

Environmental qualification tests and analyses are addressed in Section 3.11.2. The safety-related equipment in the areas of Appendix 3I.3.2 is required to remain functional in the environmental conditions expected at the equipment location during and after the limiting DBA. Qualification tests and analyses of electrical equipment for the effects of aging, radiation, temperature, humidity, chemical spray, submergence, and power supply variation, as applicable, are performed and the results documented in accordance with NEDE-24326-1-P.

Dynamic and seismic qualification testing and analysis of the electrical equipment in SSAR Appendix 3I.3.2 are addressed in Section 3.10 except for pump motors and valve motor operators, which are addressed in Section 3.9. The tests and analyses are performed in accordance with IEEE Std 344-1987 (Reference 9), which is endorsed by Regulatory Guide 1.100.

In summary the Class 1E electrical equipment is qualified for (a) the environment in which it is required to operate, including a limiting DBA at the end of design life, in accordance with 10 CFR 50.49, and (b) the seismic and dynamic conditions which it is required to withstand in accordance with the recommendations of Regulatory Guide 1.100. This issue is therefore resolved for the ABWR.

REFERENCES

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

2. IEEE Std. 323-1974, "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations," Institute of Electrical and Electronics Engineers.

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

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.

5. Regulatory Guide 1.89 Revision 1, "Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants," U.S. NRC, June 1984.

6. Regulatory Guide 1.100 Revision 2, "Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants", U.S. NRC, June 1988.

7. NEDE-24326-1-P, "General Electric Environmental Qualification Program, January 1983 (Proprietary).

8. IEEE Std. 323-1974, "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations," Institute of Electrical and Electronics Engineers.

9. IEEE Std. 344-1987, "Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations," Institute of Electrical and Electronics Engineers.

19B.2.9 A-25: NON-SAFETY LOADS ON CLASS 1E POWER SOURCES

ISSUE

Generic Safety Issue (GSI) A-25 in NUREG-0933 (Reference 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 restrict the connection of primarily safety loads 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 (LOOP) 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 384-1981 (Reference 2), and guidance is provided in Regulatory Guide 1.75, Revision 2 (Reference 3). [Supplemental information on Class 1E safety systems may be found in IEEE Standard 603-1980, IEEE Standard 279-1971, and IEEE Standard 308-1980, (References 4, 5 and 6 respectively).]

Both industry and the NRC, through IEEE Standard 384-1981 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 384-1981 and other applicable standards, References 2 and 4 through 6, respectively, taking into consideration the guidance provided in Regulatory Guide 1.75, Revision 2.

RESOLUTION

The ABWR Standard Plant 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.

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 LOOP 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.

The ABWR design excludes non-Class-1E from the Class 1E busses, with the exception of the alternate rod insertion (ARI) function which is accomplished by the rod control and information system (RC&IS) and the fine-motion control rod drive (FMCRD) subsystem. The reliability of this subsystem is enhanced for the anticipated transient without scram (ATWS) event by using Class 1E power for the drive motors.

Class 1E load breakers in the switchgear are part of the isolation scheme between the Class 1E power and the non-Class 1E FMCRD loads. In addition to the normal overcurrent tripping of these load breakers, zone selective interlocking (ZAI) is provided between them and the upstream Class 1E bus feed breakers. The Class 1E load breakers, in conjunction with the ZSI feature, provides the needed isolation between the Class 1E bus and the non-Class 1E loads. (See 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 384-1981 and meet the intent of Regulatory Guide 1.75, Revision 2, 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 Standard Plant.

REFERENCES

1. NUREG-0933, "A Prioritization of Generic Safety Issues" (with supplements), U.S. NRC, April 1989.

2. IEEE Standard 384-1981, "Criteria for Separation of Class 1E Equipment and Circuits," The Institute of Electrical and Electronics Engineers, Inc.

3. Regulatory Guide 1.75, Rev. 2, "Physical Independence of Electric Systems," U.S. NRC, September 1978.

4. IEEE Standard 603-1980, "Standard Criteria for Safety Systems for Nuclear Power Generating Stations," The Institute of Electrical and Electronics Engineers, Inc.

5. IEEE Standard 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations," The Institute of Electrical and Electronic Engineers, Inc.

6. IEEE Standard 308-1980, "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 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 Revision 3 (Reference 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 10CFR50 Appendix A (Reference 3) General Design Criteria (GDC) 1 through 5.

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) is composed of three electrically and mechanically independent divisions designated as A, B, and C with each division containing the necessary piping, pumps, valves, and heat exchangers (see ABWR SSAR Section 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.

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.

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 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.

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 (see ABWR SSAR 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.

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 (see ABWR SSAR Section 5.4.7.1.6).

The RHR Shutdown Cooling System is designed to meet the intent of SRP Section 5.4.7 Rev. 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 SSAR Standard Design.

REFERENCES

1. NUREG-0933, "A Status Report on Unresolved Safety Issues", U.S. NRC, April 1989.

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

3. 10CFR50 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

Unresolved Safety Issue (USI) A-35 in NUREG-0933 (Reference 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 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; and (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 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 Standard Plant 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 6.9kV 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 8.3.1.1. Protection against degraded voltage is specifically addressed in Subsection (8) of 8.3.1.1.7. The protection schemes are designed according to the recommendations of IEEE Standard 741-1986 (Reference 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 GDC 18 (Reference 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 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

1. NUREG-0933, "A Prioritization of Generic Safety Issues" (with supplements), U.S. NRC, April 1989.

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

3. ANSI/IEEE 741-1986, "Criteria for the Protection of Class 1E Power Systems and Equipment in Nuclear Power Generating Stations," Institute of Electrical and Electronics Engineers, Inc.

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.2.13 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 and was originally identified in NUREG-0371 but was later determined to be a USI.

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 GESSAR II, Revision 1. Deviation from this configuration shall be reviewed on 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 3).

RESOLUTION

Safety/Relief Valves (S/RVs) are utilized in a BWR pressure suppression system to provide pressure relief during certain reactor transients. S/RV 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 S/RV 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 a X-quencher such as has been used in prior plants (Reference 1).

Following the actuation of a S/RV, water contained initially in the discharge line is rapidly discharged through the X-Quencher discharge device attached at the end of the S/RV 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 S/RV 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 (Reference 5).

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 S/RV blowdown loading on submerged structures in the pool and on the pool boundary (Reference 5).

The final stage of S/RV blowdown is the steady steam flow phase. Submerged structure and pool boundary loading is from condensing steam jet oscillations at the quencher (Reference 1).

This USI was resolved with issue of SRP (Reference 6) Section 6.2.1.1.C. NUREG-0763 (Reference 2), NUREG-0783 (Reference 3), and NUREG-0802 (Reference 4) were also issued for Mark I, II, and III containments, respectively. The load definition methodology for defining the S/RV air bubble loads on submerged structures will be consistent with that used for prior plants. Therefore, this issue is resolved for the ABWR (Reference 5).

REFERENCES

1. ABWR SSAR Section 3B.2.1: Safety/Relief Valve Actuation.

2. NUREG-0763, "Guidelines for Confirmatory In-plant Tests of Safety Relief Valve Discharges for BWR Plants," U.S. NRC, May 1981.

3. NUREG-0783, "Suppression Pool Temperature Limits for BWR Containments," U.S. NRC, November 1981.

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

5. ABWR SSAR Section 3B.5.4: S/RV Submerged Structures.

6. NUREG-0800, "Standard Review Plan" U.S. NRC.

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

ISSUE

Generic Safety Issue (GSI) 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 NSI 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 NSI 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 over 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 2.5.2, 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, December 1989.

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, April 1988.

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

19B.2.15 A-42: PIPE CRACKS IN BOILING WATER REACTORS

ISSUE

Generic Safety Issue (GSI) A-42 in NUREG-0933 (Reference 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 GSI 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 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 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 design.

REFERENCES

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

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 off-site or on-site 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 1).

ACCEPTANCE CRITERIA

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

a) SECY-90-016–Evolutionary LWR Certification Requirement (Reference 1)

b) NRC Commissioner Policy Statement Certification Requirement (Reference 2)

c) 10CFR50.63, Loss of all Alternating Current Power (Reference 3)

d) Regulatory Guide 1.155, Station Blackout (Reference 4)

e) NUMARC-87-00 Guidelines and Technical Basis for Resolution of SBO (Reference 5)

f) EPRI-URD–Utility Requirements for Evolutionary LWRS (Reference 6)

g) NUREG-1469-NRC-STAFF DFSER–Section 8.3.9 (Reference 7)

RESOLUTION

The ABWR design satisfies the acceptance criteria by demonstrating (in SSAR Appendix 1C: Station Blackout Performance) in that the ABWR can withstand a station blackout without core damage or loss of containment integrity for a time period internal from 10 minutes to at least 8 hours depending on the power recovery through the use of the combustion Turbine Generator or the On-Site DGS or the Off-Site Power Sources for a wide variety of SBO events. Therefore this issue is resolved for the ABWR.

REFERENCES

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

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

3. 10CFR50.63, "Loss All Alternating Current Power (Station Blackout–SBO), July 21, 1988.

4. RG-1.155, "Station Blackout", July 1988.

5. NUMARC-87-00, "Guidelines and Technical Basis for NUMARC Initiation Addressing Station Blackout at LWR's" Plus Supplemental Q/A, January 4, 1990.

6. EPRI-URD, "EPRI-Utility Requirements Document for Evolutionary ALRW", July, 1990.

7. NUREG-1469, "Draft Final Safety Evolution Report – Design Certification of GE-ABWR (DFSER), October.

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 results is the release of large quantities of combustible gases, principally hydrogen, that may accumulate inside the primary reactor containment as a result of:

1. 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; and

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 1) and General Design Criteria 41, "Containment Atmosphere Cleanup," (Reference 2) in Appendix A to 10 CFR Part 50 (Reference 3), 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 requires that provision be made for a hydrogen control system that can safely accommodate hydrogen generated by the equivalent of a 100 percent fuel-clad metal-water reaction.

An inerted contained and the provision of permanently installed hydrogen recombiners are acceptable as hydrogen control measures.

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 5). 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 10CFR50.44, published December 2, 1981.

The ABWR containment is inerted and per 10CFR50.34 (f)(2)(ix) (Reference 4) 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. GE SSAR Section 6.2.5.3 (Reference 6) states that this is equivalent to the reaction of the active clad to a depth of 0.00023 inches or 0.72% of the active clad. Therefore, this issue is resolved for the ABWR.

REFERENCES

1. 10 CFR 50.44, "Standards for Combustible Gas Control System in Light-Water-Cooled Power Reactors".

2. General Design Criteria 41, "Containment Atmosphere Cleanup".

3. 10 CFR Part 50, Appendix A.

4. 10 CFR 50.34 Paragraph (f)(2)(ix).

5. SECY 89-122 dated April 19, 1989.

6. GE SSAR Subsection 6.2.5.3.

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 1).

Following a postulated LOCA, escaping steam forces the suppression pool 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 6).

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 (see Reference 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 2).

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 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 prior Mark III design, but includes some unique design features. These unique features include pressurization of the wetwell airspace, the presence of a lower drywell, the smaller number of horizontal vents (30 in ABWR vs. 120 in Mark III), extension of horizontal vents into the pool, vent submergence, and suppression pool width (Reference 3).

The ABWR horizontal vent test (HVT) program produced test data which can be used to confirm 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 (Reference 7).

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 Reference 4. The load definition methodology for defining the LOCA induced loads on submerged structures is consistent with the methodology used for prior plants (Reference 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

1. NUREG-0471, "Generic Task Problem Descriptions (Categories B, C, and D)," US. NRC, June 1978.
2. NUREG-0978, "Mark III LOCA-Related Hydrodynamic Load Definition," U.S. NRC, February 1984.
3. ABWR SSAR Section 3B.1.2: ABWR Containment Design Features.
4. ABWR SSAR Section 3B.2.2: Loss-of-Coolant Accidents.
5. ABWR SSAR Section 3B.5: Submerged Structure Loads.
6. NUREG-0933, "A Status Report on Unresolved Safety Issues," U.S. NRC, April 1989.
7. 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 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, is whether or not to require an automatic switchover from the injection mode to the recirculation mode following a LOCA (Reference 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. 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 NRC requirements 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 bases accident. 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. In fact, even in the long term, operator action is only required for one situation – initiation of containment cooling transients. 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 CRTs in the control room shall be utilized for monitoring and alarm functions for safety-related and non-safety-related systems (References 2, 3, 4). Therefore, this issue is resolved for the ABWR.

REFERENCES

1. NUREG-0933, "A Prioritization of Generic Safety Issues," (and Supplements 1-12), July 1991.

2. EPRI NP-4361, "Power Plant Alarm Systems: A Survey and Recommended Approach For Evaluating Improvements", December 1985.

3. EPRI NP-5693P, "Evaluation of Alternative Power Plant Alarm Presentations".

4. EPRI NP-3448, "A Procedure For Reviewing and Improving Power Plant Alarm Systems", April 1984.

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

This NUREG-0471, Reference 1, item involves developing revisions to current guidance and staff technical positions regarding ESF and normal ventilation system air filtration and adsorption units.

ACCEPTANCE CRITERIA

Develop revisions to BTP ETSB11, 2 Reference 2, and Regulatory Guide 1.52, Reference 3, to address technical advances that have shown that some current positions are unjustifiably conservative some are unnecessary, and in some cases additional positions are necessary.

RESOLUTION

Criteria developed as a result of this issue have been documented in Regulatory Guide 152 Revision 2, Reference 3, issued in March 1978 and in Regulatory Guide 1.140 Revision 1, Reference 4, issued in October 1979, and Reference 5. Thus, this item has been resolved.

REFERENCES

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

2. NUREG-0800, "Standard Review Plan," U.S. NRC.

3. 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.

4. 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.

5. Memorandum for T. Speis from J. Funches, "Prioritization of Generic Issues – Environmental and Licensing Improvements," February 24, 1983.

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. The performance of these valves, however, is under continual surveillance and the consequences of their failures are subject to review.

RESOLUTION

The B-55 issue is not applicable to the ABWR. The ABWR uses a direct acting safety/relief valve design. This design does not have a pilot stage such as that present in the Target Rock 2-stage 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 2-stage safety/relief valves. GE has identified the principal cause of the most significant concern with these Target Rock 2-stage 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.

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

Generic Safety Issue (GSI) B-56 in NUREG-0933 (Reference 1), addresses emergency diesel generator reliability. The reliability goal identified in NSAC-108, (Reference 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 (DRAFT) (Reference 3).

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

ACCEPTANCE CRITERIA

The acceptance criteria for the resolution of GSI 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 (DRAFT) and 1.155.

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) [see 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 (see 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 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 [see 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 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 (DRAFT), and 1.155. Therefore, this issue is resolved for the ABWR Standard Plant design.

REFERENCES

1. NUREG-0933, "A Prioritization of Generic Safety Issues" (with supplements), U.S. NRC, January 1989.

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

3. Regulatory Guide 1.9, Revision 3 (DRAFT), "Selection, Design, Qualification, Testing, and Reliability of Diesel Generator Units Used as Onsite Electrical Power Systems at Nuclear Power Plants," U.S. NRC, November 1988.

4. Regulatory Guide 1.155, "Station Blackout," U.S. NRC, August 1988.

5. EPRI, "Advanced Light Water Reactor Utility Requirements Document," Electric Power Research Institute, Chapter 11, April 1989.

19B.2.24 B-61: ALLOWABLE ECCS EQUIPMENT OUTAGE PERIODS

ISSUE

Generic Safety Issue (GSI) B-61 in NUREG-0933 (Reference 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 indicated that from 30 to 80 percent of the ECCS system unavailability was due to testing, maintenance and allowed outage periods. The NRC is therefore evaluating whether overall ICCS unavailability, and resulting core damage frequency, can be reduced by extending the intervals between testing and maintenance of equipment 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 (Reference 8).

As a part of this program a computer code (References 2 and 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 GSI B-61 for future plant designs is that the Technical Specification LCOs surveillance periods and allowable outage times of ECCS equipment shall be developed in accordance with current industry practice.

The LCOs surveillance periods and outage times shall be accounted for in the overall plant PRA required by 1 CFR 52 (Reference 4). Any subsequent proposed changes to the LCOs' provisions for ECCS surveillance shall be demonstrated to be within the results of an existing PRA (Reference 8).

RESOLUTION

The ABWR Standard Plant Design (Reference 7) 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 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 (Reference 7).

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 5), and basic failure rate data is obtained from the ERPI ALWR Requirements Document (Reference 6) supplemented with other nuclear sources. 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.

The PRA demonstrates that the ABWR Standard Design meets the industry goal of 1.0 x E-5 core damage frequency per reactor year for future reactors 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

1. NUREG-0933, "A Prioritization of Generic Safety Issues" (with supplements), U.S. NRC, December 1989.

2. NUREG-0193, "FRANTIC – A Computer Code for Time Dependent Unavailability Analysis", U.S. NRC, October 1977.

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

4. 10CFR52, Early Site Permits; Standard Design Certification; and Combined Licenses for Nuclear Power Reactors", Office of the Federal Register, National Archives and Records Administration.

5. NUREG/CR-2815, "Probabilistic Safety Analysis Procedures Guide", Brookhaven National Laboratory, January 1984.

6. "Advanced Light Water Reactor Requirements Document – Chapter 1: Overall Requirements, Appendix A: PRA Key Assumptions and Groundrules", Electric Power Research Institute, Draft, April 1987.

7. CESSAR Design Certification Amendment 1, December 21, 1990.

8. ABWR SSAR Section 6.3: ECCS.

19B.2.26 B-66: CONTROL ROOM INFILTRATION MEASUREMENTS

ISSUE

Issue B-66 in NUREG-0933 (Reference 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 2), including the habitability review provisions of TMI Action Plan Item III.D.3.4 (Reference 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 6.4.1, Revision 2, 9.4.1 and 15.6.5.5 Revision 2.

RESOLUTION

The ABWR control room is heated, cooled and pressurized by a system mixing recirculated air with filtered outdoor air. It is designed to ensure that the operators can remain in the control room and take actions to operate the plant 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. Radiation monitoring sensors in each air intake warn operators of airborne contamination, and cause the HVAC system to switch automatically to an emergency system employing HEPA and charcoal filters for cleanup.

This control room heating, ventilating and air-conditioning (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 10CFR50, Appendix A, General Design Criterion 19 (Reference 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.

This ABWR control room and its design bases are described in Section 6.4, Habitability Systems, and Section 9.4.1, Control Room Habitability Area HVAC.

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

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. 10CFR50 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 12), addresses concerns regarding the long-term capability of hermetically-sealed instruments and equipment which must function in post-accident environments. NUREG-0471 (Reference 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 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 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, GDC-4, 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 of the ABWR SSAR.

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 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, Equipment Qualification Environmental Design Criteria (EQEDC). 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 5) and permitted by 10CFR50.49(f) (Reference 6).

The qualification methodology is described in detail in the NRC approved Licensing Topical Report on GE's environmental qualification program (Reference 7). This report also addresses compliance with the applicable portions of the General Design Criteria of 10CFR50, Appendix A, and the Quality Assurance Criteria of 10CFR50, Appendix B. Additionally, the report describes conformance to NUREG-0588, and Regulatory Guides and IEEE Standards referenced in Section 3.11 of NUREG-0800, "Standard Review Plan," (Reference 8).

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

1. NUREG-0933, "A Prioritization of Generic Safety Issues" (with supplements), U.S. NRC, January 1989.

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

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

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

5. IEEE Standard 323-1983, "Qualifying Class 1E Equipment for Nuclear Power Generating Stations," The Institute of Electrical and Electronic Engineers, Inc.

6. 10 CFR 50, "General Design Criteria for Nuclear Power Plants," Code of Federal Regulations, Office of the Federal Register, National Archives and Records Administration.

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

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

Spray Subsystem, the Primary Containment System, and the SPCU System are listed in reference ABWR SSAR 16.7.5, 16.9.1, and 16.9.2 (see ABWR SSAR Section 3.6.2 for addition LCOs).

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 (see ABWR SSAR Sections 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.

GSI C-10 in NUREG-0933 is therefore resolved for the ABWR Standard Design.

REFERENCES

1. NUREG-0933, "A Status Report on Unresolved Safety Issues," U.S. NRC, December 1989.

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

3. ANSI/AN 56.5-1979, "PWR and BWR Containment Spray System Design Criteria," American National Standards Institute.

4. 10CFR50 Appendix A, Code of Federal Regulations, 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 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 that there is no current criteria for acceptability of solidification agents. This NUREG-0471 (Reference 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 2). A BTP 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 (Subsection 11.4.1.2). Thus this item has been RESOLVED for the ABWR.

REFERENCES

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

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

19B.2.30 15: RADIATION EFFECTS ON REACTOR VESSEL SUPPORTS

ISSUE

Generic Safety Issue (GSI) 015 in NUREG-0933 (Reference 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 2 and 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 is reviewing its regulatory position relative to low temperature and neutron flux radiation embrittlement.

ACCEPTANCE CRITERIA

The acceptance criteria for the resolution of GSI 015 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 B&PV Code (Reference 4).

RESOLUTION

The RVSS for the ABWR is described in Subsection 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} nvt can be compared to the bounding expected value for the skirt welds of 3×10^{14} nvt for a 60 year exposure.

REFERENCES

1. NUREG-0933, "A Status Report on Unresolved Safety Issues," U.S. NRC, December 1989.

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

3. ORNL/TM-10966, "Impact of Radiation Embrittlement on the Integrity of Pressure Vessel Supports for Two PWR Plants," Oak Ridge National Laboratory, 1988.

4. American Society of Mechanical Engineers, Boiler & Pressure Vessel Code, Section III (Nuclear), American Society of Mechanical Engineers.

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 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 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. 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 scram initiated by 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 RPS to initiate an immediate scram. The 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.

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

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 212°F into the reactor building at a maximum flow rate of 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 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 1) is not applicable to me 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

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 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 2), to supplement the existing guidance and requirements which include the Standard Review Plant (SRP) Section 7.1, 10 CFR 50, Appendix A and industry standard ISA-67.02, (References 3, 4, and 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, (Rev. 3), 7.1 Appendix A, (Rev. 1), 7.5, (Rev. 3) and 7.7, (Rev. 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 incorp[illegible] instrument sensing lines in safety-related systems and components. All safety-related systems and compone[illegible]ed [illegible] the ABWR Standard Plant design, including instrument sensing lines, are located in temperature controlle[illegible]ments which are maintained above the freezing (or precipitation) point of the contained fluid. The tempera[illegible] of these environments are not expected to be less than 10 degrees C, as shown in Appendix 3I. In addition, the ABWR is committed to meet the requirements of Regulatory Guide 1.151 (see Table 1.8-20), which endorses and augments ISA-S67.02. Therefore, this issue is resolved for the ABWR Standard Plant Design.

REFERENCES

1. NUREG-0933, "A Prioritization of Generic Safety Issues" (with supplements), U.S. NRC, April 1989.

2. Regulatory Guide 1.151 "Instrument Sensing Lines," U.S. NRC, July 1983.

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

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.

5. ISA-S67.02, "Nuclear-Safety-Related Instrument Sensing Line Piping and Tubing Standards for Use in Nuclear Power Plants," Instrument Society of America, 1980.

19B.2.35 51: PROPOSED REQUIREMENTS FOR IMPROVING THE RELIABILITY OF OPEN CYCLE SERVICE WATER SYSTEMS

ISSUE

Generic Safety Issue (GSI) 51 in NUREG-0933 (Reference 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 eliminate heat sinks is a design goal of the ABWR. The Plant Designer is given specific requirements and guiding on achieving this goal, including instruction to consider designs and new requirements which further mitigate the fouling effects. Additionally, the Plant Designer is directed to investigate the problem with ice as a flow blockage mechanism and to dispose of and/or dissolve such ice as required.

Finally, the SWS design of ABWR units at multiplant sites avoids the reliability problems described in this issue be requiring the SWS to have two or more pumps, so that the loss of one pump will not prevent adequate SWS flow. The final design of the ABWR ultimate heat sinks and water flow systems will avoid or minimize as achievable the problems described in this issue.

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 (see Reference 2 and Section 19B.2.10, Service Water System Reliability):

(a) 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 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.

(b) Raw service water will be filtered and treated to reduce the effect of mud, silt, or organisms.

(c) Materials for piping, pumps, and heat exchangers are specified to offer greater resistance to the range of probable water chemistry conditions.

(d) Provisions will be made to facilitate the inspection of service water piping and replace sections of piping during plant life.

Sufficient redundancy of makeup pumps shall be provided 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 Reg. Guide 1.27 (Reference 3).

The safety related portions of these systems shall be designed 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.

However, the NRC staff has a concern regarding the non-inclusion of provisions for: periodic analyses of intake water and substrate, full flow testing of infrequently used loops, biocide treatment of the reactor service water and other systems susceptible to biofouling (Reference 4). Resolution of GSI 51 will require addressing these items in addition to the criteria of 19B.2.10 for a site-specific application.

REFERENCES

1. NUREG-0933, "A Status Report on Unresolved Safety Issues", U.S. NRC, April 1989.
2. Advanced Light Water Reactor Utility Requirement Document (Volume II), EPRI.
3. Regulatory Guide 1.27, Ultimate Heat Sink for Nuclear Power Plants, Revision 2, January 1976.
4. Enclosure 1, Generic Letter 89-13, "Service Water System Problems Affecting Safety-Related Equipment".

19B.2.36 057: EFFECTS OF FIRE PROTECTION SYSTEMS ACTUATION ON SAFETY-RELATED EQUIPMENT

ISSUE

Generic Safety Issue (GSI) 057 in NUREG-0933 (Reference 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 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 057, 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 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 preclude water spray from the fire protection system onto safety-related equipment. The sprinkler systems protecting the safety-related equipment is of the automatic sprinkler type. Actuation of these sprinkler systems requires the opening of the fusible link sprinkler heads and detection by combustible-products and/or heat detectors. In addition, the operator has the capability of isolating flow locally by manual isolation valves.

In order to prevent damage due to flooding, upon actuation of sprinkler systems, floor drains are provided and equipment is located to preclude the flooding of the equipment.

The basic layout of an ABWR and the choice of systems is such as to enhance the tolerance of the ABWR plant to fire. The systems are designed such that there are three independent safety-related divisions, any one of which is capable of providing safe shutdown of the reactor. It is assumed that a fire in any location in a divisional fire area results in an immediate loss of function of the entire division. The remaining two independent safety-related divisions are capable of performing the safe shutdown function.

Since the Fire Protection systems are designed to preclude inadvertent actuation and thus minimize damage to safety-related equipment and because these systems are designed in accordance with 10 CFR 50 Appendix A (GDC 3), this issue is resolved for the ABWR (See Section 9.5 Appendix 9A)

REFERENCES

1. NUREG-0933, "A Prioritization of Generic Safety Issues" (with supplements), U.S. NRC, April 1989.
2. IE Information Notice 83-41; Actuation of Fire Suppression System Causing Inoperability of Safety-Related Equipment"; SSINS No. 6835.
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 RG 1.97 Rev. 2 (Reference 1) which was implemented at Ginna.

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 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-037, Supplement 1. Refer to the applicable Sections of 7.5, 7.6.2.2, 7.6.2.6, 11.5, and 12.3.4. The ABWR certification Program is in full compliance with the latest issue of RG 1.97 (Ref. 3).

REFERENCES

1. NUREG-0737, NUREG-0737, "Clarification of TMI Action Plan Requirements", U.S. NRC, November 1980.
2. NUREG-0933, "A Prioritization of Generic Safety Issues" (with supplements), U.S. NRC, April 1989.
3. US/ABWR SSAR Doc. # 23A6100AF Amendment 26.
4. Regulatory Guide 1.97 Revision 3, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident", U.S. NRC, May, 1983.

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 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 2) and NUREG-1000 (Reference 3).

RESOLUTION

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

The ABWR design also addresses and fulfills the ATWS rule of 10CFR50.62 as described in 19B.2.5, A-19 ATWS and Section 15.8.

Therefore this issue 75 is resolved for ABWR.

REFERENCES

1. NUREG-0933, "A Prioritization of Generic Safety Issues", (with Supplements 1-12), July 1991.

2. Generic Letter No. 83-28, "Required Actions Based on Generic Implication of Salem ATWS Events", July 8, 1983.

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-9033 (Reference 1), addresses the concern that for Operating Plants, 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 environmental effects on the fatigue life of ASME III Class 1 carbon steel piping should be considered in accordance with GE Document number 408HA414 (Reference 3). In addition, for Operating Plants an assessment of the increase in CDF due to environmental effects on fatigue resistance of material should be performed.

Environmental effects are considered by increasing the local peak stress through four factors used as multipliers to the stress indices. The four factors are: (1) the notch factor, (2) the mean stress factor, (3) the environmental correction factor, and (4) the butt weld strength reduction factor. The fatigue cumulative usage factors are calculated using the calculated local peak stresses and the ASME Section III Design Fatigue curves.

RESOLUTION

For the ABWR, environmental effects are included in the design bases for 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, April 1989.

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

3. G.E. Document No. 408HA414, Revision 1, "Plain Carbon Steels," General Electric Company

19B.2.40 83: CONTROL ROOM HABITABILITY

ISSUE

Safety Issue 83 in NUREG-0933 (Reference 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, Section 9.4.1, and Section 15.6.5.5, (Reference 2). The design must be in accordance with 10CFR50, Appendix A, General Design Criteria (GDC) 2, 4, and 19 (Reference 3) and ASME AG-1 and AG-1a (Reference 5).

RESOLUTION

The ABWR main control room habitability system is described in Sections 9.4.1 and 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., 5 rem 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 4), as described in Subsection 9.4.1.1.7. Fire protection is provided by alarm systems, fire hoses and portable fire extinguishers, Sections 9.5.1, 9A.4.2. Testing and inspection requirements are identified in Section 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

1. NUREG-0933, "A Prioritization of Generic Safety Issues", U.S. NRC, July 1991 (and Supplements 1-12).

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

3. 10CFR50 Appendix A, "General Design Criteria for Nuclear Power Plants", Office of the Federal Register, National Archives and Records Administration.

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", Revision 2, March 1978.

5. ASME AG-1 (1991) Code on Nuclear Air and Gas Treatment and ASME AG-1a (1992) Addenda.

19B.2.42 87: FAILURE OF HPCI STEAM LINE WITHOUT ISOLATION

ISSUE

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

The HPCI steam supply line has two containment isolation valves (MOV's) 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 RWCU also has two normally open containment isolation valves (MOV's) 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 demonstrated.

Furthermore, NRC sponsored testing has increased the concern over whether MOV's 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 MOV's. 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 RWCU and HPIC 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 2).

ACCEPTANCE CRITERIA

The acceptance criteria for the resolution of Generic Safety Issue 87 is defined in Generic Letter 89-10 which requires adequately sized actuators for MOV's, verification of correct thrust and torque settings, and a program for testing, inspection and maintenance of MOV's 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 RWCU 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 Sections 3.9.3.2 Pump and Valve Operability Assurance. 3.9.6.2.2 In-Service Testing-Motor Operated valves and 19B.2.1 In-Situ Testing of Valves.

Thus, compliance with GL 89-10 resolves concerns on GSI-87 for the ABWR design.

REFERENCES

1. NUREG-0933, "A Prioritization of Generic Safety Issues" (with supplements), U.S. NRC, April 1989.

2. Generic Letter No. 89-10, "Safety-related Motor-operated Valve Testing and Surveillance".

19B.2.43 89: STIFF PIPE CLAMPS

ISSUE

Generic Safety Issue (GSI) 89 in NUREG-9033 (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. An 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 GSI 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. Since the clamps will only be installed on straight or very nearly straight runs of pipe, evaluations previously performed show that the peak piping system stresses will not occur at the clamp locations. The stress intensification that occurs at elbows, branch connections and lugs is much greater than that which occurs at pipe clamps on essentially straight runs of pipe.

Sample calculations were made for typical clamps used on BWR Main Steam and Recirculation piping systems, to evaluate stresses due to the following loads: (1) Differential thermal expansion of the pipe and the 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; (4) External loads produced by dynamic events such as earthquake and thermo-hydraulic loads.

Approximate pipe stress distributions were calculated for these loads and conservatively combined to obtain the incremental primary and secondary stresses. Maximum incremental primary stresses were less than 25% of primary stress allowables, and maximum incremental secondary stresses were less than 40% of secondary stress allowables. The stresses at the clamp locations calculated without considering local clamp effects typically fell between 15% and 30% of the ASME Section III Code allowables. The total piping primary and secondary stresses, including the clamp induced local stresses, were less than 70% of the ASME Section III Code allowables. The governing stress locations occurred at the piping branch connections, lugs, elbows and transitions, they did not occur at the clamp locations.

RESOLUTION

For the ABWR, pipe clamps will only be installed on straight runs of pipe or on bends with radius of at least five pipe diameters. Based on the analyses summarized above, the total piping stress including the additional stresses induced by the pipe clamp will be less than the governing stresses that occur at piping branch connections, elbows, lugs and transitions. Therefore, the pipe clamp induced stresses can be considered negligible and do not warrant explicit consideration.

REFERENCES

1. NUREG-0933, "A Prioritization of Generic Safety Issues" (with supplements), U.S. NRC, April 1989.

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

19B.2.~~18~~ 45 105: Interfacing System LOCA at ABWRs

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. This leads to the situation that systems in BWRs are designed for a pressure lower than that of the primary system. For example, the BWR primary system operates an about 70 kg/cm^2g, while the Residual Heat Removal (RHR)System can operate at pressures up to 35 kg/cm^2g, and the pump suction lines are designed for 14 kg/cm^2. Isolation valves, at least two, and piping to the primary system are designed for about 88 kg/cm^2g. 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 systems or other systems interfacing with the RCPB, 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 1).

Acceptance Criteria

Reference 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 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 1040 psia 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 1025 psig (i.e., 410 psig). 410 psig = 28.8 atg, where 1 at = 1 kg/cm^2 and atg is gage.

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., 410 psig). This is accomplished in the SSAR by the revised boundary symbols of the P&IDs to the 28.8 atg 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 Boiler and Pressure Vessel 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 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 the SSAR 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:

1. 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,
SLC main tank,
LCW receiving tank,
HCW receiving tank,
FPC skimmer surge tank, and
FPC 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.

2. 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.

3. It is impractical to design piping systems that are connected to low pressure sink features 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 tne 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 (SLC) System
6. Reactor Water Cleanup (CUW) System
7. Fuel Pool Cooling Cleanup (FPC) System
8. Nuclear Boiler (NB) System
9. Reactor Recirculation (RRS) System
10. Makeup Water (Condensate) (MUWC) System
11. Makeup Water (Purified) (MUWP) System
12. Radwaste System (LCW Receiving Tank, HCW Receiving Tank).

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 insure 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 ISLOCA evaluation.

Condensate, Feedwater and Condensate Air Extraction (C,FDW,AO) System
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 the SSAR 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 4, page 8, paragraph 7.

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 9.49 kg/cm^2 gage (135 psig).

In summary, based on the NRC staff's new guidance cited in References 2 through 5, the ABWR is in full compliance. For ISLOCA considerations, a design pressure of 28.8 atg or (410 psig) 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 72.1 atg (1025 psig) by applying the guidance recommended by Reference 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

1. NUREG-0933, "A Prioritization of Generic Safety Issues," (and Supplements 1 through 12), July 1991

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

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

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," Jan. 12, 1990

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

Combustible gases such as hydrogen, propane and acetylene are used during normal operation of nuclear power plants and in plant laboratories. Most gases are used in limited quantities and for relatively short periods of time. Hydrogen, the most prevalent combustible gas used in nuclear power plants, is used as a coolant for electric generators. Hydrogen also is used in the volume control tank (VCT), located in the auxiliary systems building of PWRs. The concern is that a hydrogen leakage and accumulation in this building could ignite and disable safety-related equipment.

ACCEPTANCE CRITERIA

The acceptance criteria for the resolution of issue 106, is that the hydrogen and other combustible gas piping be designed to preclude large releases and accumulation of combustible mixtures in buildings which enclose safety-related equipment. Furthermore, the designer shall follow the guidance described in SRP 9.5.1, Fire Protection Program (Reference 2).

RESOLUTION

The ABWR design incorporates various compressed gas systems for plant operating applications such as the hydrogen water chemistry (HWC) system and the main generator hydrogen system. Both of these systems are non-nuclear, non-safety-related and are required to be safe and reliable, consistent with the requirements for using hydrogen gas as described in Subsection 9.3.9, Hydrogen Water Chemistry, and Subsection 10.2, Turbine-Generator.

Since the combustible gas systems are designed in compliance with SRP 9.5.1, so that their failure will not jeopardize safety-related equipment, this issue is resolved for the ABWR.

REFERENCES

1. NUREG-0933, "A Prioritization of Generic Safety Issues", (and Supplements 1-12), July 1991.

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

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19B.2.48 118: TENDON ANCHORAGE FAILURE

ISSUE

Generic Safety Issue (GSI) 118 in NUREG-0933 (Reference 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 ounces to about 1.5 gallons were found in the grease caps.

ACCEPTANCE CRITERIA

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

RESOLUTION

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

REFERENCES

1. NUREG-0933, "A Prioritization of Generic Safety Issues" (with supplements), U.S. NRC, April 1989.

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 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 10CFR50.34(f) (Reference 2) on limiting the distributed hydrogen concentration to 10 percent, 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

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

2. 10CFR50 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 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

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

19B.2.52 128: ELECTRICAL POWER RELIABILITY

ISSUE

Generic Safety Issue (GSI) 128 in NUREG-0933 (Reference 1), 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 NRC performed a generic evaluation of the reliability of safety-related dc power and published the results in NUREG-0305 (Reference 2) and NUREG-0666 (Reference 3).

NUREG-0666 provided recommendations and supporting technical bases for augmenting the minimum design criteria and procedural requirements which will provide greater assurance of dc power supply reliability. These recommendations for augmenting the minimum requirements for dc power systems are: (1) prohibiting certain design and operation features of the dc power systems, such as use of a bus tie breaker, which could compromise division independence; (2) augmenting the test and maintenance activities presently required for battery operability to also include preventive maintenance on bus connections, procedures to demonstrate dc power availability from the battery to the bus, and administrative controls to reduce the likelihood of battery damage during testing, maintenance, and charging activities; (3) requiring staggered test and maintenance activities to minimize the potential for human error-related common cause failure associated with these operations; and (4) requiring design and operation features adequate to maintain reactor core cooling in the hot standby condition following the loss of any other system required for shutdown cooling.

For plants not yet built, the NRC is considering further enhancing the reliability of the dc power supplies by (1) placing non-safety-related loads on completely separate dc power supplies (i.e., non-safety-related balance-of-plant and switchyard batteries), and (2) dividing the dc power supplies which are safety-related or essential into separate systems to reduce the probability of a reactor trip in the event of the loss of a single dc bus.

Also under consideration is NRC endorsement of IEEE Standards 603 (Reference 4) and 308 (Reference 5) with possible revisions to the related Regulatory Guides.

RESOLUTION

The resolution for GSI 128, as stated above, suggests elements which are applicable only to the design or the administrative operation of operating plants, and are not applicable to the design of the ABWR. For the ABWR, the problems described in this issue are completely avoided by the following inherent design features (which are described in detail in Section 8.3):

1. The ABWR utilizes four completely independent Class 1E dc divisions which power two-out-of-four logic to actuate safety systems. If a division is taken out of service, the logic reverts to two-out-of-three. Because of this level of redundant trip channels, no single power supply failure results in a reactor scram, even when a division is out of service.

2. There are no bus tie breakers between divisions. However, it is possible, through special administrative controls and key interlocks, to manually power one division's dc loads from a different division through the spare charger (see Figure 8.3-4, and Subsection 8.3.4.18).

3. All non-Class 1E dc loads are powered from non-Class 1E dc sources with only one exception. This special case is the Alternate Rod Insertion (ARI) function utilizing the Fine-Motion Control Rod Drive (FMCRD)

motors. For ATWS considerations, the reliability of this subsystem is enhanced by using Class 1E power for the drive motors. This power interface exists only on Division I, and is isolated by zone-selective interlocked circuit breakers (see 8.3.1.1.1).

4. Three of the four dc divisions are backed by independent Class 1E diesel generators. (The fourth division battery charger is supplied power from Division II, and hence, is backed by the Division II diesel.) The non-Class 1E plant investment protection (PIP) loads are backed by an on-site combustion turbine generator (CTG).

5. There are two separate and independent connections from the off-site sources to each of the three Class 1E buses, and to each of the three PIP buses.

6. The ABWR fully complies with IEEEs 308 and 603.

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 on-site sources of ac and dc power, and at least two connections for off-site 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

1. NUREG-0933, "A Prioritization of Generic Safety Issues" (with supplements), U.S. NRC, April 1989.

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

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

4. IEEE Standards 603-1980, "Standard Criteria for Safety Systems for Nuclear Power Generating Stations," The Institute of Electrical and Electronics Engineers, Inc.

5. IEEE Standard 308-1980, "Criteria for Class 1E Electric Systems for Nuclear Power Generating Stations," The Institute of Electrical and Electronic Engineers, Inc.

19B.2.53 142: LEAKAGE THROUGH ELECTRICAL ISOLATORS IN INSTRUMENT 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 SPDS evaluation tests that, 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.

RESOLUTION

Fiber optic data links are the only type of isolation device used for electrical isolation of logic level and analog signals between protection divisions and from protection divisions to non-safety-related equipment.

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 cause itself, are the same as those used in existing, qualified isolation devices.

When using fiber optic devices as Class 1E isolation devices, only the input side of the transmitting device and out side of the receiving device use electrical power. The low voltage power supplies for these devices use the same power source as the logic that drives the isolating device. For ABWR safety systems, this power is:

1. Divisional 120 Volt Vital AC (UPS) – For Reactor Protection System (RPS) logic and Main Steam Isolation Valve (MSIV) logic.

2. 125 Volt Plant DC Power Supply – For ECCS logic and Leak Detection and Isolation System (LDS) logic.

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 has not been considered. Thus, plants with similar inherent deficiencies may not be aware of these problems.

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

ACCEPTANCE CRITERIA

A possible resolution to this issue would require a reevaluation of each plant's room heat load and heat-up rate in order to identify areas in which a reduction in the dependence of equipment operability on HVAC and room cooling may be implemented. While the total elimination of this dependence may not be possible at all plants, this analysis would identify areas in which this dependence is critical. After the critical dependencies are identified, each plant would implement procedural changes (to provide alternate cooling) to eliminate or reduce the dependencies where possible. Hardware modifications may be needed for situations in which a procedure change cannot be implemented to reduce a critical dependency.

The next step in the possible solution to this issue would be the issuance of a generic letter that would require licensees to: (1) evaluate the dependencies of plant safety systems and equipment operability on HVAC and room cooling; (2) identify areas in which this dependence is critical; (3) identify appropriate procedure changes and hardware modifications to minimize the effects of the dependencies on plant risk; (4) submit this evaluation to the NRC for review and approval of the proposed modifications; and (5) implement the approved proposed procedural changes and hardware modifications. The generic letter would include guidance on acceptable procedures licensees could use to evaluate the potential dependencies in the designs of these systems. The generic letter would also include alternative solutions for improving the independence of systems that are critical to plant risk. It is assumed that a research project would form the basis for a more fully-developed solution and for the guidance in the generic letter.

RESOLUTION

The safety related equipment areas housing key components such as residual heat removal pumps, switchgear, and diesel generators shall be provided with calibrated pressure and temperature monitors which can be calibrated in site. Pressure and temperature (ambient and differential temperature) along with flow requirements can be used to monitor and diagnose the applicable equipments performances. This implementation will assure the total control of loss of HVAC systems, and protects the systems against fire. Therefore, this issue is resolved for the ABWR (Reference 1).

REFERENCES

1. Advanced Light Water Reactor Utility Requirement Document (Volume II), EPRI.

19B.2.56 151: RELIABILITY OF ANTICIPATED TRANSIENT WITHOUT SCRAM RECIRCULATION PUMP TRIP IN BWRs

ISSUE

Generic Safety Issue (GSI) 151 in NUREG-0933 (Reference 1), addresses the issue of the reliability of the ATWS RPT in BWRs. GSI-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 2).

ACCEPTANCE CRITERIA

The acceptance criterion for the resolution of GSI-151 is the use of reactor recirculation system pump trip hardware or method that is more reliable then 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 then 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 incore internal pumps (RIPs) are used. The ABWR RPT trip hardware (not yet specifically identified) will be completely different: 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. See Subsection 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.

REFERENCES

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

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

This issue addresses the potential unavailability of the essential service water (ESW) system for all LWRs except those seven multiplant sites addressed under Issue 130. The 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. It is known by different names at various types of plants. The design and operational characteristics of the ESW system are different for PWRs and BWRs. In addition, these characteristics may differ significantly in each of these reactor types.

Under Issue 153, the staff will examine all potential causes for ESW system unavailability, except those that are considered to be resolved by implementing the resolutions addressed in GL 89-13 (Reference 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 not yet 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 safety service water (SSW) system are summarized as follows:

1. Performance Requirements

   - The SSW system will be designed to meet the required heat loads.

   - The SSW 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.

2. System Arrangement

   - The SSW system will be divided into approximately equal-sized divisions, three for the BWR and ABWR.

   - 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.

The performance requirements and system arrangement for the SSW system indicated above do not adequately address the safety concerns of Issue 153. These concerns include partial or complete loss of ESW system functions resulting from common causes, degradation of the SSW 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.

The design of the ESW system varies substantially from plant to plant and the ESW system is highly dependent on the NSSS. As a result, generic solutions (if needed) are likely to be different for PWRs and BWRs. The possible solutions are: (1) installation of a redundant intake structure including a service water pump; (2) hardware changes of the ESW system; (3) installation of a dedicated RCP seal cooling system; or (4) changes to TS or operational procedures.

RESOLUTION

The ABWR Reactor Service Water (RSW) system removes heat from the Reactor Building Cooling Water (FCW) 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 exchange 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 9.2.11.)

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 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, or required), component material selection suited to site water conditions, and 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.)

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 and the system design features which will be provided by the COL applicant.

REFERENCES

1. GL89-13, Service Water System Problems Affecting Safety Related Equipment, July 18, 1989.

2. Advanced Light Water Reactor Utility Requirement Document (Volume II), EPRI.

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 1) and is contained in Regulatory Guides 1.3 and 1.4. The regulatory guides specify a release into containment of 100 percent of the core inventory of noble gases and 50 percent 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 (a) the post-accident environment for which safety-related equipment should be qualified, (b) post-accident habitability requirements for the control room, and (c) 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 10CFR100.

RESOLUTION

The ABWR is currently licensed to and analyzed to the existing Regulatory Guides, Standard Review Plans, and General Design Criteria which are based upon TID-14844. The use of revised source terms based upon NUREG-1465 (Reference 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

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

2. Soffer, L. et al, "Accident Source Terms for Light-Water Nuclear Power Plants", NUREG-1465, USNRC, Draft Report for Comment, June 1992.

19B.2.59 A-17: SYSTEMS INTERACTION

ISSUE

Issue A-17 in NUREG-0933 (Reference 1) addresses the concern that unintended or unrecognized dependencies may exist between systems that could lead to safety-significant events.

Nuclear plant design includes interdisciplinary reviews to assure the functional compatibility of the plant structures, systems and components and compliance with licensing requirements. Safety reviews and accident analyses provide further assurance that system functional and licensing requirements will be met. Thus the design and analysis of the plant take into account systems interactions. Nevertheless, the process may not consider all the interactions of various plant systems. Based on this possibility, adverse systems interaction is defined as actions or consequences in one system that could adversely affect the redundance or independence of safety systems in another system or systems. The issue involves preventing any adverse systems interactions that affect plant functions or regulatory requirements.

ACCEPTANCE CRITERIA

The acceptance criteria for the resolution of issue A-17 is that attention shall be given in the detailed plant design to detecting and minimizing the potential for adverse system interactions using the guidance of NUREG-1174 (Reference 2).

RESOLUTION

To respond to issue A-17 the detailed design of ABWR shall provide the following elements:

Design requirements that ensure separation and isolation of electrical power systems to preclude interactions that could adversely safety-related power as described in Chapter 8, Electric Power.

Design requirements that ensure spatial separation of systems and equipment to prevent interaction between redundant safety grade equipment and systems or adverse interaction of non-safety grade equipment with safety grade equipment. Consideration shall include seismically coupled and flooding spatial interactions. For example refer to Subsection 6.3.1, ECCS – Design Bases and Summary Description.

Plant probabilistic evaluations to detect potential systems interactions (e.g., to ensure that redundant safety grade systems and equipment are not installed in the same room or fire area and to assess the impacts of high energy line breaks and flooding) as described in Appendix 19D, Probabilistic Evaluations.

REFERENCES

1. NUREG-0933, "A Prioritization of Generic Safety Issues", July, 1991 (and Supplements 1-12).

2. NUREG-1174, "Evaluation of Systems Interactions in Nuclear Power Plants," 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 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 Section 9.5, Fire Protection, Section 3.4, Floods and Section 3.6, Pipe Whip Protection.

Appendix 19C, Design Consideration Reducing Sabotage Risk, 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, Physical Security, describes the physical protection systems and controls for compliance with 10CFR73.55 (Reference 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.

In summary, the ABWR design is highly resistant to sabotage, because of the feature described which protect against internal and external sabotage. Therefore, this issue is resolved for the ABWR.

REFERENCES

1. NUREG-0933, "A Prioritization of Generic Safety Issues", (and Supplements 1-12).

2. 10CFR73.55, "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-05: DUCTILITY OF TWO-WAY SLABS AND SHELLS AND BUCKLING BEHAVIOR OF STEEL CONTAINMENTS

ISSUE

Generic Safety Issue (GSI) B-05 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.

ACCEPTANCE CRITERIA

The acceptance criterion for the first concern is that analysis methods used for two-way reinforced concrete slabs adequately address dynamic loading in biaxial membrane tension, flexure, and shear that occur due to a HELB or LOCA.

The acceptance criterion for the second concern is that all applied loads must be adequately addressed by the steel containment vessel design.

RESOLUTION

Since the ABWR containment design is based upon ACI 349-85, which establishes methods by which the above loading conditions and the latest codes for the first concern of this issue are addressed, and the steel containment design meets the requirements of the ASME Code for the second concern of this issue, both concerns are fully resolved for the ABWR design, Reference 5.

REFERENCES

1. NUREG-0933, "A Status Report on Unresolved Safety Issues", U.S. NRC, December 1989.

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.

19B.2.62 029: BOLTING DEGRADATION OR FAILURE IN NUCLEAR POWER PLANTS

ISSUE

Issue 029 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 and 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 029 are that proven bolting designs, materials, and fabrication techniques shall be employed. In addition, reactor coolant pressure boundary (RCPB) bolting shall meet the requirements of ASME Code, Section III (Reference 2). 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 (Reference 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 and 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. 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 Design.

REFERENCES

1. NUREG-0933, "A Status Report on Unresolved Safety Issues", U.S. NRC, April 1989.

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

19B.2.63 82: BEYOND DESIGN BASIS ACCIDENTS IN SPENT FUEL POOLS

ISSUE

Issue 82 in NUREG-0933 (Reference 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 2) and Regulatory Guide 1.13 (Reference 3).

RESOLUTION

The ABWR design includes a spent fuel storage facility, a fuel pool cooling and cleanup system and a fuel handling system that meets the intent of Regulatory Guide 1.13 and SRP 9.1.2 – 9.1.5 as described in Section 9.1, Fuel Storage and Handling. Since the acceptance criteria are met for the spent fuel storage facility, this issue is resolved for the ABWR.

REFERENCES

1. NUREG-0933, "A Prioritization of Generic Safety Issues," (and Supplements 1-12), July 1991.
2. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition."
3. Regulatory Guide 1.13, "Design Objective for Light-Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations," Revision 2, December 1981.

19B.3 COL LICENSE INFORMATION

INSERT 19B.3.1

19B.3.1 Quality Assurance Program

COL applicants referencing the ABWR design shall have a Quality Assurance Program satisfying the requirements of Subsection 19B.2.1(2) including the right to impose additional quality assurance requirements.

19B.3.2 Prevention of Core Damage

COL applicants referencing the ABWR design shall approve applicable design deviations in divisional total independence and separation both mechanically and electrically as required by Subsections 19B.2.3(3), 19B.2.4(2), 19B.2.5(3), and 19B.2.11(2).

19B.3.3 Protection from External Threats

COL applicants referencing the ABWR design shall evaluate listed man-made hazards except sabotage on a site unique basis as required by Subsection 19B.2.4(6).

19B.3.4 Ultimate Heat Sink Models

COL applicants referencing the ABWR design shall implement the development of predictive analytical models as required by Subsection 19B.2.9(1) through 19B.2.9(4).

19B.3.5 Ultimate Heat Sink Reliability

COL applicants referencing the ABWR design shall have an ultimate heat sink design goal for the service water flow as required by Subsection 19B.2.10(1) through 19B.2.10(13).

19B.3.6 Main Transformer Design

COL applicants referencing the ABWR design shall provide main transformer fire protection as required by Subsection 19B.2.18.

19B.3.7 Plant Siting

COL applicants referencing the ABWR design shall approve in writing the listed final design parameters to be used at the plant site as required by Subsection 19B.2.19(3).

19B.3.8 Interdisciplinary Design Reviews

COL applicants referencing the ABWR design shall establish an interdisciplinary design review group and direct reviews for site specific design and construction work as required by Subsection 19B.2.25(4).

19B.3.9 Sabotage Vulnerability During Plant Shutdown

The sabotage vulnerability analysis required by Subsection 19B.2.4(10) has been performed for the ABWR and is contained in Appendix 19C. However, applicants referencing the ABWR design shall include provision in the plant start-up procedures to inspect critical safety equipment within the containment for possible tampering just prior to sealing the containment in preparation for start-up. Such equipment includes the ADS/SRV valves and associated accumulators and their charging lines and the inboard valves associated with the emergency core cooling systems (i.e., HPCF, RHR and RCIC).

19B.3.10 Impact of Security System on Plant Operation, Testing and Maintenance

In the design of the security system, applicants referencing the ABWR design shall include an evaluation of its impact on plant operation, testing and maintenance. This evaluation shall be conducted as required by Subsections 19B.2.4(12) and 9.5.13.11. This analysis should include consideration of an emergency requiring evacuation of the control room in the control building to the remote shutdown panel in the reactor building.

19B.3.11 Security Plan Compatibility with ALWR Requirements

The ABWR security plan will comply with the ALWR requirements as defined in 19B.2.4. Future amendments of the ALWR Requirements Document must be reviewed for ABWR compliance by the applicants referencing the ABWR Standard Plant.

19B.3.12 Plant Security Systems Electrical Requirements

COL applicants will provide non-Class 1E vital (uninterruptible) ac power for the site security system. [See Subsection 19B.2.4(20)].

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19B.3.1 COL Applicant Safety Issues
The COL applicant shall provide resolutions for the issues identified as COL applicant in Table 19B.1-1.

19B.3.13 Bolting Degradation or Failure

COL applicants shall provide the bolting information detailed in Subsection 19B.2.12(6).

19B.3.14 Outside Sabotage

COL applicants shall provide sufficient analyses to ensure that the plant is adequately protected from acts of outsider sabotage.

19B.3 COL LICENSE INFORMATION

INSERT 19B.3.1

19B.3.1 Quality Assurance Program

COL applicants referencing the ABWR design shall have a Quality Assurance Program satisfying the requirements of Subsection 19B.2.1(2) including the right to impose additional quality assurance requirements.

19B.3.2 Prevention of Core Damage

COL applicants referencing the ABWR design shall approve applicable design deviations in divisional total independence and separation both mechanically and electrically as required by Subsections 19B.2.3(3), 19B.2.4(2), 19B.2.5(3), and 19B.2.11(2).

19B.3.3 Protection from External Threats

COL applicants referencing the ABWR design shall evaluate listed man-made hazards except sabotage on a site unique basis as required by Subsection 19B.2.4(6).

19B.3.4 Ultimate Heat Sink Models

COL applicants referencing the ABWR design shall implement the development of predictive analytical models as required by Subsection 19B.2.9(1) through 19B.2.9(4).

19B.3.5 Ultimate Heat Sink Reliability

COL applicants referencing the ABWR design shall have an ultimate heat sink design goal for the service water flow as required by Subsection 19B.2.10(1) through 19B.2.10(13).

19B.3.6 Main Transformer Design

COL applicants referencing the ABWR design shall provide main transformer fire protection as required by Subsection 19B.2.18.

19B.3.7 Plant Siting

COL applicants referencing the ABWR design shall approve in writing the listed final design parameters to be used at the plant site as required by Subsection 19B.2.19(3).

19B.3.8 Interdisciplinary Design Reviews

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COL applicants referencing the ABWR design shall establish an interdisciplinary design review group and direct reviews for site specific design and construction work as required by Subsection 19B.2.25(4).

19B.3.9 Sabotage Vulnerability During Plant Shutdown

The sabotage vulnerability analysis required by Subsection 19B.2.4(10) has been performed for the ABWR and is contained in Appendix 19C. However, applicants referencing the ABWR design shall include provision in the plant start-up procedures to inspect critical safety equipment within the containment for possible tampering just prior to sealing the containment in preparation for start-up. Such equipment includes the ADS/SRV valves and associated accumulators and their charging lines and the inboard valves associated with the emergency core cooling systems (i.e., HPCF, RHR and RCIC).

19B.3.10 Impact of Security System on Plant Operation, Testing and Maintenance

In the design of the security system, applicants referencing the ABWR design shall include an evaluation of its impact on plant operation, testing and maintenance. This evaluation shall be conducted as required by Subsections 19B.2.4(12) and 9.5.13.11. This analysis should include consideration of an emergency requiring evacuation of the control room in the control building to the remote shutdown panel in the reactor building.

19B.3.11 Security Plan Compatibility with ALWR Requirements

The ABWR security plan will comply with the ALWR requirements as defined in 19B.2.4. Future amendments of the ALWR Requirements Document must be reviewed for ABWR compliance by the applicants referencing the ABWR Standard Plant.

19B.3.12 Plant Security Systems Electrical Requirements

COL applicants will provide non-Class 1E vital (uninterruptible) ac power for the site security system. [See Subsection 19B.2.4(20)].

INSERT 19B.3.1

19B.3.1 COL Applicant Safety Issues
The COL applicant shall provide resolutions for the issues identified as COL applicant in Table 19B.1-1.

19B.3.13 Bolting Degradation or Failure

COL applicants shall provide the bolting information detailed in Subsection 19B.2.12(6).

19B.3.14 Outside Sabotage

COL applicants shall provide sufficient analyses to ensure that the plant is adequately protected from acts of outsider sabotage.