GE Nuclear Energy

General Electric Company
175 Curtner Avenue, San Jose, CA 95125

May 4, 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 are the nine issues inadvertently omitted from my April 30, 1993 transmittal. In addition, I have included the second page of GSI 142.

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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19B2.12 A-36: CONTROL OF HEAVY LOADS NEAR SPENT FUEL

ISSUE

Issue A-36 in NUREG-0933 (Reference 1), addresses the consequence of dropping heavy loads on spent fuel. Overhead cranes are used to lift heavy objects in the vicinity of spent fuel. If the heavy object, such as a spent fuel shipping cask or shielding block, were to fall on to spent fuel there could be a release of radioactivity to the environment that could exceed 10CRF100 guidelines. This issue was resolved by the NRC with the publication of NUREG-0612 (Reference 2) and SRP Section 9.1.5 (Reference 3).

ACCEPTANCE CRITERIA

The acceptance criteria for the resolution of issue A-36 is that the overhead heavy load handling systems shall be designed to provide the equipment, procedures and operator training such that no credible drop can cause a release of radioactivity, a criticality accident, an inability to cool fuel within the reactor vessel or spent fuel pool, or prevent a safe shutdown of the reactor. Where applicable the design shall conform to the industrial and electrical codes, the relevant requirements of General Design Criteria 2, 4, and 61 of 10CFR50, Appendix A (Reference 4) and NUREG-0612.

RESOLUTION

The ABWR design addresses the above criteria as follows:

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

2. The major heavy load handling equipment components (i.e., cranes, hoists, etc.) will be provided with an operating instruction and maintenance manual for Reference and utilization by operations and maintenance personnel for use in operating procedures, maintenance procedures and operator training programs. The handling equipment operating procedures will comply with the requirements of NUREG-0612, Subsection 5.1.1(2). Refer to ABWR, Subsection 9.1.5.4, 9.1.5.8 and 9.1.6.6.

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

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

5. The reactor vessel head lifting strongback and the dryer/separator lifting strongback are designed in accordance with the acceptable factors of safety. This is in accordance with ANSI-N14.6, Reference 5 and in accordance with NUREG-0612. Refer to ABWR Subsection 9.1.4.2.5.

6. The heavy load handling system is designed in accordance with relevant requirements of GDC 2, 4, and 61 and the guidance of references 2 and 5 through 7. The ABWR design is for a single unit therefore GDC 5 is not applicable. Refer to ABWR Subsection 9.1.5.1 and 3.1.

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

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REFERENCES

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

2. NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants", U.S. NRC.

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

4. 10CFR50 Appendix A, "General Design Criteria for Nuclear Power Plants", Code of Federal Regulations, Office of the Federal Register, National Archives and Records Administration.

5. ANSI-N14.6, "American National Standard for Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 KG) or More for Nuclear Materials".

6. ANSI/ANS-57.2, "Design Requirements for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Plants".

7. ANSI/ANS-57.1, "Design Requirements for Light Water Reactors Fuel Handling Systems".

8. NUREG-0554, "Single Failure Proof Cranes for Nuclear Power Plants", U.S. NRC.

19B.2.17 A-47: SAFETY IMPLICATIONS OF CONTROL SYSTEMS

ISSUE

This issue A47 (Reference 1) concerns the potential for accidents or transients (e.g., overpressure, overfilling, reactivity events) being made more severe as a result of control system failures including control and instrumentation power support faults. These failures or malfunctions may occur independently or as a result of an accident or transient and would be in addition to any control system failure that may have initiated the event. Although it is generally believed that control system failures are not likely to result in loss of safety functions which could lead to serious events or result in conditions that safety systems are not able to cope with, in-depth studies have not been performed. The purpose is to define generic criteria that may be used for plant-specific reviews.

ACCEPTANCE CRITERIA

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

RESOLUTION

The reactor vessel overfill protection is described in Subsection 7.7.1.4. The BWR Owners Group improved technical specification submittals of limiting conditions for operations and surveillance requirements are consistent with the NRC resolution. The ABWR resolution will follow the NRC-approved Owners Group submittals. Therefore, this issue is resolved for the ABWR.

REFERENCES

1. NUREG-0933, "A Prioritization of Generic Safety Issues", U.S. NRC (including Supplement 15).

2. NUREG-1217, "Evaluation of Safety Implications of Control Systems in LWR Nuclear Power Plants", June 1989.

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

19B.2.25 B-63: ISOLATION OF LOW PRESSURE SYSTEMS CONNECTED TO THE REACTOR COOLANT PRESSURE BOUNDARY

ISSUE

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

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

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

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

ACCEPTANCE CRITERIA

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

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

RESOLUTION

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

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

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

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

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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 – LWR Edition," U.S. NRC.

3. ASME Boiler & Pressure Vessel Code, Sections III and XI.

19B.2.28 C-10: EFFECTIVE OPERATION OF CONTAINMENT SPRAYS IN A LOCA

ISSUE

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

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

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

ACCEPTANCE CRITERIA

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

RESOLUTION

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

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

The water in the 304L stainless steel-lined suppression pool is maintained at high purity (low corrosion attack) by the Suppression Pool Cleanup (SPCU) System. In the event of a LOCA, the SPCU function is automatically terminated to accomplish containment isolation. The pH range (5.3-8.6) is maintained to minimize any corrosive attack on the pool liner (304L SS) over the life of the plant. The post-LOCA aqueous phase pH in all areas of containment will have a flat time history (i.e., the liquid coolant will remain at its design basis pH throughout the event). The use of organic coatings within the containment has been kept to a minimum. The major use of such coatings is on the carbon steel containment liner, internal steel structures and equipment inside the drywell and wetwell. The epoxy coatings are specified to meet the requirements of Regulatory Guide 1.54 and are qualified using the standard ANSI tests, including ANSI N101.4. All safety-related equipment in the containment is environmentally qualified, and protected against spray actuation (see ABWR Section 3.11).

The system design adheres to the appropriate criteria guidelines of ANSI/ANS 56.5-1979. Application of accepted human factors principles and methodologies to the RHR System instrumentation and controls design minimizes the possibility of inadvertent actuation as a result of operator error (see ABWR Section 18.3.1). Pre-operational testing for operability is performed on the RHR Containment Spray Subsystem (see ABWR Section 14.2.12.1.8). Technical Specifications/Limiting Conditions for Operation (LCOs) of the RHR Containment Spray Subsystem, the Primary Containment System, and the SPCU System are listed in reference ABWR 16.7.5, 16.9.1, and 16.9.2 (see ABWR Section 3.6.2 for additional 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 Subsections 3.1.2.4.12.2, 3.1.2.4.13.2, and 3.1.2.4.14.2).

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

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

REFERENCES

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

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

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

19B.2.31 25: REACTOR COOLANT PUMP SEAL FAILURES

ISSUE

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

ACCEPTANCE CRITERIA

Not applicable. Issue does not apply to BWRs.

RESOLUTION

The ABWR wet motor Reactor Internal Pumps (RIPs) as described in the ABWR Section 5.4.1 do not include seals. This feature is further described in ABWR Subsection 1A.2.30. Therefore this Issue 23 is resolved for ABWR.

References

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

19B.2.41 86: LONG RANGE PLAN FOR DEALING WITH STRESS CORROSION CRACKING IN BWR PIPING

ISSUE

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

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

ACCEPTANCE CRITERIA

The acceptance criteria for the resolution of Issue 86 are that IGSCC resistant materials and fabrication techniques to minimize sensitization shall be used. In addition, the ABWR water shall be maintained at the lowest practically achievable impurity levels. Furthermore, the material and fabrication techniques shall comply with the guidelines of NUREG-0313 (Reference 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 the piping is fabricated, tested and installed in accordance with ASME Section III (Reference 3) and NUREG-0313. Also, the owner-operator is required to comply with ASME Code, Section XI (Reference 3) for the performance of inservice inspection. Therefore, this issue is resolved for the ABWR Standard design.

REFERENCES

1. NUREG-0933, "A Status Report on Unresolved Safety Issues," U.S. NRC, July 1991.

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

3. American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section III and Section XI, American Society of Mechanical Engineers.

19B2.44 103: DESIGN FOR PROBABLE MAXIMUM PRECIPITATION

ISSUE

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

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

ACCEPTANCE CRITERIA

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

RESOLUTION

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

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

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

4. Regulatory Guide 1.59, "Design Basis Floods for Nuclear Power Plants", U.S. NRC, August 1977.

5. Regulatory Guide 1.102, "Flood Protection for Nuclear Power Plants", U.S. Nuclear Regulatory Commission, September 1976.

19B.2.47 113: DYNAMIC QUALIFICATION TESTING OF LARGE BORE HYDRAULIC SNUBBERS

ISSUE

Issue 113 in NUREG-0933 (Reference 1), addresses the need for requirements for dynamic qualification testing of large bore hydraulic snubbers (>50 kips load rating).

ACCEPTANCE CRITERIA

The acceptance criteria for the resolution of Issue 113 for the ABWR design are specified in 19B.2.7, Issue A-13, Snubber Operability Assurance.

RESOLUTION

For the ABWR design, the requirements for dynamic qualification testing of large bore hydraulic snubbe.. specified in Issue A-13.

REFERENCES

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

19B.2.49 120: ON-LINE TESTABILITY OF PROTECTION SYSTEMS

ISSUE

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

ACCEPTANCE CRITERIA

The requirements for at-power testability of components are included in GDC 21 of Appendix A to 10 CFR 50 (Reference 5) Supplementary guidance is provided in Regulatory Guides 1.22 Reference 2 and 1.118 (Reference 3 and IEEE Standard 338 (Reference 3) to ensure that protection systems (including logic, actuation devices, and associated actuated equipment) will be designed to permit testing while the plant is operating without adversely affecting the plant's operation. These requirements apply to both the RPS and the ESFAS. Existing Standard Technical Specification indicate that it is desirable to test all protection systems every 6 months.

RESOLUTION

In the ABWR design the RPS and ESFAS can be tested during reactor operation by six separate tests. The first five tests are primarily manual tests and, although each individually is a partial test, when combined with the sixth test they constitute a complete system test. The sixth test is a self test of the safety system logic and control which automatically tests the complete system, excluding sensors and actuators. Online testability of protection systems is explained in Section 7.1.2.1.6. In the ABWR design, all actuation logic is solid state and in software.

Automatic system self-testing occurs during a portion of every periodic transmission period of the data communication network. Since exhaustive tests cannot be performed during any one transmission interval, the test software is written so that sufficient overlap coverage is provided to prove system performance during tests of portions of the circuitry, as allowed in IEEE 338 (Reference 4).

REFERENCES

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

2. Regulatory Guide 1.22, Periodic Testing of Protection System Actuation functions.

3. Regulatory Guide 1.118, "Periodic Testing of Electric Power and Protection Systems" Revision 2 dated June 1978.

4. IEEE Standards 338-1977 (Criteria for the Periodic Testing of Nuclear Power Generating Station Safety Systems.

5. 10CFR50 Appendix A, "General Design Criteria for Nuclear Power Plants" Office of the Federal Register, National Archives and Records Administration.

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

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

The results of the NUREG tests show that the fiber optic type of isolators exhibited no or very little effects from the major fault and lightning surge tests. Only surge and EMI tests applied to the isolator power supplies caused damage to the isolator input side, mainly because of the output and input supplies sharing a common, commercial AC power line. However, as noted in the NUREG, BWRs do not directly use a commercial power source. For the ABWR, RPS and ESF functions are supplied from different plant power sources (120 Volt Vital AC and 125 Vdc, respectively). The low voltage DC supplies fed from these sources are highly regulated and filtered. This, isolator circuit is isolated from most power source transients.

REFERENCES

1. Memorandum for B. Morris from B. Sheron, "Proposed Generic Issue on Leakage Through Electrical Isolators," June 23, 1987.

2. NUREG/CR-3453, "Electronic Isolators Used in Safety Systems of U.S. Nuclear Power Plants," U.S. NRC, March 1986.