CHAPTER XI

AGING MANAGEMENT PROGRAMS (AMPs)

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XI.M1 ASME SECTION XI INSERVICE INSPECTION, SUBSECTIONS IWB, IWC, AND IWD

Program Description

The Code of Federal Regulations, 10 CFR 50.55a, imposes the inservice inspection (ISI) requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section XI, for Class 1, 2, and 3 pressure-retaining components and their integral attachments in light-water cooled power plants. Inspection, repair, and replacement of these components are covered in Subsections IWB, IWC, and IWD, respectively, in the 1995 edition through the 1996 addenda. The program generally includes periodic visual, surface, and/or volumetric examination and leakage test of all Class 1, 2, and 3 pressure-retaining components and their integral attachments.

The ASME Section XI inservice inspection program in accordance with Subsections IWB, IWC, or IWD has been shown to be generally effective in managing aging effects in Class 1, 2, or 3 components and their integral attachments in light-water cooled power plants. However, in certain cases, the ASME inservice inspection program is to be augmented to manage effects of aging for license renewal and is so identified in the GALL report.

Evaluation and Technical Basis

1. ***Scope of Program:*** The ASME Section XI program provides the requirements for ISI, repair, and replacement. The components within the scope of the program are specified in Subsections IWB-1100, IWC-1100, and IWD-1100 for Class 1, 2, and 3 components, respectively, and include all pressure-retaining components and their integral attachments in light-water cooled power plants. The components described in Subsections IWB-1220, IWC-1220, and IWD-1220 are exempt from the examination requirements of Subsections IWB-2500, IWC-2500, and IWD-2500.

2. ***Preventive Actions:*** The ASME Section XI does not provide guidance on methods to mitigate degradation.

3. ***Parameters Monitored/Inspected:*** The ASME Section XI ISI program detects degradation of components by using the examination and inspection requirements specified in ASME Section XI Tables IWB-2500-1, IWC-2500-1, or IWD-2500-1, respectively, for Class 1, 2, or 3 components.

4. ***Detection of Aging Effects:*** The extent and schedule of the inspection and test techniques prescribed by the program are designed to maintain structural integrity and ensure that aging effects will be discovered and repaired before the loss of intended function of the component. Inspection can reveal crack initiation and growth; loss of material due to corrosion; leakage of coolant; and indications of degradation due to wear or stress relaxation, such as verification of clearances, settings, physical displacements, loose or missing parts, debris, wear, erosion, or loss of integrity at bolted or welded connections.

   Components are examined and tested as specified in Tables IWB-2500-1, IWC-2500-1, and IWD-2500-1, respectively, for Class 1, 2, and 3 components. The tables specify the extent and schedule of the inspection and examination methods for the components of the pressure-retaining boundaries. Alternative approved methods that meet the requirements of IWA-2240 are also specified in these tables.

The program uses three types of examination — visual, surface, and volumetric — in accordance with the general requirements of Subsection IWA-2000. Visual VT-1 examination detects discontinuities and imperfections, such as cracks, corrosion, wear, or erosion, on the surface of components. Visual VT-2 examination detects evidence of leakage from pressure-retaining components, as required during the system pressure test. Visual VT-3 examination (a) determines the general mechanical and structural condition of components and their supports by verifying parameters, such as clearances, settings, and physical displacements; (b) detects discontinuities and imperfections, such as loss of integrity at bolted or welded connections, loose or missing parts, debris, corrosion, wear, or erosion; and (c) observes conditions that could affect operability or functional adequacy of constant-load and spring-type components and supports.

Surface examination uses magnetic particle, liquid penetrant, or eddy current examinations to indicate the presence of surface discontinuities and flaws.

Volumetric examination uses radiographic, ultrasonic, or eddy current examinations to indicate the presence of discontinuities or flaws throughout the volume of material included in the inspection program.

For BWRs, the nondestructive examination (NDE) techniques appropriate for inspection of vessel internals and their implementation needs, including the uncertainties inherent in delivering and executing and NDE technique in a boiling water reactor (BWR), are included in the approved boiling water reactor vessel and internals project (BWRVIP)-03. Also, an applicant may use the guidelines of the approved BWRVIP-62 for inspection relief for vessel internal components with hydrogen water chemistry.

The ASME Section XI examination categories used in this report are given below. These examination categories are based on the 1989 edition of Section XI of the ASME Code; any differences in the examination categories in the 1995 edition through the 1996 addenda from those in the 1989 edition are identified.

Class 1 Components, Table IWB-2500-1

Examination category B-B for pressure-retaining welds in vessels other than reactor vessels: This category specifies volumetric examination of circumferential and longitudinal shell-to-head welds and circumferential and meridional head welds in pressurizers, and circumferential and meridional head welds and tubesheet-to-head welds in steam generators (primary side). The welds selected during the first inspection interval are reexamined during successive inspection intervals.

Examination category B-D, for full penetration welds of nozzles in reactor vessels, pressurizers, steam generators (primary side), and heat exchangers (primary side): This category specifies volumetric examination of all nozzle-to-vessel welds and the nozzle inside radius.

Examination category B-E, for pressure-retaining partial penetration welds in vessels: This category specifies visual VT-2 examination of partial penetration welds in nozzles and penetrations in reactor vessels and pressurizers during the hydrostatic test. In the 1995 edition of the ASME Code, examination category B-E is covered under examination category B-P.

Examination category B-F, for pressure-retaining dissimilar metal welds in reactor vessels, pressurizers, steam generators, heat exchangers, and piping: This category specifies volumetric examination of the inside diameter (ID) region and surface examination of the outside diameter (OD) surface for all nozzle-to-safe end butt welds of nominal pipe size (NPS) 4 in. or larger. Only surface examination is conducted for all butt welds less than NPS 4 in. and for all nozzle-to-safe end socket welds. Examinations are required for each safe end weld in each loop and connecting branch of the reactor coolant system. In the 1995 edition of the ASME Code, examination category B-F for piping is covered under examination category B-J for all pressure-retaining welds in piping.

Examination category B-G-1 for pressure-retaining bolting greater than 2 in. in diameter, and category B-G-2 for pressure-retaining bolting less than 2 in. in diameter in reactor vessels, pressurizers, steam generators, heat exchangers, piping, pumps, and valves: Category B-G-1 specifies volumetric examination of studs in place, from the top of the nut to the bottom of the flange hole; and surface and volumetric examination of studs when removed; volumetric examination of flange threads; and visual VT-1 examination of the surfaces of nuts, washers, and bushings. Category B-G-2 specifies visual VT-1 examination of the surfaces of nuts, washers, and bushings. For heat exchangers, piping, pumps, and valves, examinations are limited to components selected for examination under examination categories B-B, B-J, B-L-2, and B-M-2.

Examination category B-H for integral attachments for vessels: This category specifies volumetric or surface examination of essentially 100% of the length of the attachment weld at each attachment subject to examination.

Examination category B-J for pressure-retaining welds in piping: This category specifies volumetric examination of the ID region and surface examination of the OD for circumferential and longitudinal welds in each pipe or branch run NPS 4 in. or larger. Surface examination is conducted for circumferential and longitudinal welds in each pipe or branch run less than NPS 4 in. and for all socket welds. The pipe welds selected during the first inspection interval are reexamined during each successive inspection interval.

Examination category B-L-1, for pressure-retaining welds in pump casing, and category B-L-2, for pump casing: Category B-L-1 specifies volumetric examination of all welds, and category B-L-2 specifies visual VT-3 examination of internal surfaces of the pump casing. All welds from at least one pump in each group of pumps performing similar functions in the system (such as recirculating coolant pumps) are inspected during each inspection interval. Visual examination is required only when the pump is disassembled for maintenance, repair, or volumetric examination, but one pump in a particular group of pumps is visually examined at least once during the inspection interval.

Examination category B-M-1, for pressure-retaining welds in valve bodies and category B-M-2, for valve bodies: Category B-M-1 specifies volumetric examination for all welds in valve bodies NPS 4 in. or larger, and surface examination of OD surfaces for all welds in valve bodies less than NPS 4 in. Category B-M-2 specifies visual VT-3 examination of internal surfaces of valve bodies. All welds from at least one valve in each group of valves that are of the same size, construction design (such as globe, gate, or check valves), and manufacturing method, and that perform similar functions in the system (such as the containment isolation valve) are inspected during each inspection interval. Visual examination is required only when the valve is disassembled for maintenance, repair, or volumetric examination, but one valve in a particular group of valves is visually examined at least once during the inspection interval.

Examination category B-N-1, for the interior of reactor vessels: Category B-N-1 specifies visual VT-3 examination of interior surfaces that are made accessible for examination by removal of components during normal refueling outages. *Examination category B-N-2, for integrally welded core support structures and interior attachments to reactor vessels*: Category B-N-2 specifies visual VT-1 examination of all accessible welds in interior attachments within the beltline region; visual VT-3 examination of all accessible welds in interior attachments beyond the beltline region; and, for BWRs, visual VT-3 examination of all accessible surfaces in the core support structure. *Examination category B-N-3, which is applicable to pressurized water reactors (PWRs), for removable core support structures*: Category B-N-3 specifies visual VT-3 examination of all accessible surfaces of reactor core support structures that can be removed from the reactor vessel.

Examination category B-O, for pressure-retaining welds in control rod housing: This category specifies volumetric or surface examination of the control rod drive (CRD) housing welds, including the weld buttering.

Examination category B-P, for all pressure-retaining components: This category specifies visual VT-2 examination of all pressure-retaining boundary components during the system leakage test and hydrostatic test (IWA-5000 and IWB-5000). The pressure-retaining boundary during the system leakage test corresponds to the reactor coolant system boundary, with all valves in the normal position, which is required for normal reactor operation startup. However, VT-2 visual examination extends to and includes the second closed valve at the boundary extremity. The 1995 edition of the ASME Code eliminates the hydrostatic test because equivalent results are obtained from the leakage test. The pressure-retaining boundary for the hydrostatic test (1989 edition) and system leakage test (1995 edition) conducted at or near the end of each inspection interval extends to all Class 1 pressure-retaining components within the system boundary.

Class 2 Components, Table IWC-2500-1

Examination category C-A, for pressure-retaining welds in pressure vessels: This category specifies volumetric examination of circumferential welds at gross structural discontinuities, such as junctions between shells of different thickness or cylindrical shell-to-conical shell junctions, and head-to-shell, shell (or head)-to-flange, and tubesheet-to-shell welds.

Examination category C-F-1, for pressure-retaining welds in austenitic stainless steel or high-alloy piping: This category specifies, for circumferential and longitudinal welds in each pipe or branch run NPS 4 in. or larger, volumetric and surface examination of the ID region, and surface examination of the OD surface for piping welds ≥3/8 in. nominal wall thickness for piping >NPS 4 in. or for piping welds >1/5 in. nominal wall thickness for piping ≥NPS 2 in. and ≤NPS 4 in. Surface examination is conducted for circumferential and longitudinal welds in pipe branch connections of branch piping ≥NPS 2 in. and for socket welds.

Examination category C-G, for all pressure-retaining welds in pumps and valves: This category specifies surface examination of either the inside or outside surface of all welds in the pump casing and valve body. In a group of multiple pumps or valves of similar design, size, function, and service in a system, examination of only one pump or one valve among each group of multiple pumps or valves is required to detect the loss of intended function of the pump or valve.

Examination category C-H, for all pressure-retaining components: This category specifies visual VT-2 examination during system pressure tests (IWA-5000 and IWC-5000) of all pressure-retaining boundary components. The pressure-retaining boundary includes only those portions of the system required to operate or support the safety function, up to and including the first normally closed valve (including a safety or relief valve) or valve capable of automatic closure when the safety function is required. The 1995 edition of the ASME Code eliminates the hydrostatic test because equivalent results are obtained from the leakage test.

Class 3 Components, Table IWD-2500-1

Examination category D-A (1989 edition), for systems in support of reactor shutdown function, and category D-B (1989 edition), for systems in support of emergency core cooling, containment heat removal, atmosphere cleanup, and reactor residual heat removal: Categories D-A and D-B specify visual VT-2 examination during system pressure tests (IWA-5000 and IWD-5000) of all pressure-retaining boundary components. The pressure-retaining boundary extends up to and includes the first normally closed valve or valve capable of automatic closure as required to perform the safety-related system function. Examination categories D-A and D-B, from the 1989 edition of the ASME Code, have been combined into examination category D-B for all pressure-retaining components in the 1995 edition of the ASME Code.

5. ***Monitoring and Trending:*** For Class 1, 2, or 3 components, the inspection schedule of IWB-2400, IWC-2400, or IWD-2400, respectively, and the extent and frequency of IWB-2500-1, IWC-2500-1, or IWD-2500-1, respectively, provides for timely detection of degradation. The sequence of component examinations established during the first inspection interval is repeated during each successive inspection interval, to the extent practical. If flaw indications or relevant conditions of degradation are evaluated in accordance with IWB-3100 or IWC-3100, and the component is qualified as acceptable for continued service, the areas containing such flaw indications and relevant conditions are reexamined during the next three inspection periods of IWB-2410 for Class 1 components and for the next inspection period of IWC-2410 for Class 2 components. Examinations that reveal indications that exceed the acceptance standards described below are extended to include additional examinations in accordance with IWB-2430, IWC-2430, or IWD-2430 (1995 edition) for Class 1, 2, or, 3 components, respectively.

6. ***Acceptance Criteria:*** Any indication or relevant conditions of degradation detected are evaluated in accordance with IWB-3000, IWC-3000, or IWD-3000, for Class 1, 2, or 3 components, respectively. Examination results are evaluated in accordance with IWB-3100 or IWC-3100 by comparing the results with the acceptance standards of IWB-3400 and IWB-3500 or IWC-3400 and IWC-3500, respectively, for Class 1 or Class 2 and 3 components. Flaws that exceed the size of allowable flaws, as defined in IWB-3500 or IWC-3500, are evaluated by using the analytical procedures of IWB-3600 or IWC-3600, respectively, for Class 1 or Class 2 and 3 components. Approved BWRVIP-14, BWRVIP-59, and BWRVIP-60 documents provide guidelines for evaluation of crack growth in stainless steels, nickel alloys, and low-alloy steels, respectively.

7. ***Corrective Actions:*** For Class 1, 2, and 3, respectively, repair is in conformance with IWB-4000, IWC-4000, and IWD-4000, and replacement according to IWB-7000, IWC-7000, and IWD-7000. Approved BWRVIP-44 and BWRVIP-45 documents, respectively, provide guidelines for weld repair of nickel alloys and for weldability of irradiated structural components. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable in addressing corrective actions.

8. ***Confirmation Process:*** Site quality assurance (QA) procedures, review and approval processes, and administrative controls are implemented in accordance with the requirements of 10 CFR Part 50, Appendix B. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable in addressing the confirmation process and administrative controls.

9. ***Administrative Controls:*** See Item 8, above.

10. ***Operating Experience:*** Because the ASME Code is a consensus document that has been widely used over a long period, it has been shown to be generally effective in managing aging effects in Class 1, 2, and 3 components and their integral attachments in light-water cooled power plants (see Chapter I of the GALL report, Vol. 2).

    Some specific examples of operating experience of component degradation are as follows:

    BWR: Cracking due to intergranular stress corrosion cracking (IGSCC) has occurred in small- and large-diameter BWR piping made of austenitic stainless steels and nickel alloys. The IGSCC has also occurred in a number of vessel internal components, such as core shrouds, access hole covers, top guides, and core spray spargers (NRC IE Bulletin 80-13, NRC Information Notice [IN] 95-17, NRC General Letter [GL] 94-03, and NUREG-1544). Crack initiation and growth due to thermal and mechanical loading have occurred in high-pressure coolant injection (HPCI) piping (NRC IN 89-80) and instrument lines NRC Licensee Event Report [LER] 50-249/99-003-1). Jet pump BWRs are designed with access holes in the shroud support plate at the bottom of the annulus between the core shroud and the reactor vessel wall. These holes are used for access during construction and are subsequently closed by welding a plate over the hole. Both circumferential (NRC IN 88-03) and radial cracking (NRC IN 92-57) have been observed in access hole covers. Failure of the isolation condenser tube bundles due to thermal fatigue and transgranular stress corrosion cracking (TGSCC) due to leaky valves has also occurred (NRC LER 50-219/98-014).

    PWR Primary System: Although the primary pressure boundary piping of PWRs has generally not been found to be affected by SCC because of low dissolved oxygen levels and control of primary water chemistry, SCC has occurred in safety injection lines (NRC IN 97-19 and 84-18), charging pump casing cladding (NRC IN 80-38 and 94-63), instrument nozzles in safety injection tanks (NRC IN 91-05), CRD seal housing (NRC Inspection Report 50-255/99012), and safety-related stainless steel (SS) piping systems that contain oxygenated, stagnant, or essentially stagnant borated coolant (NRC IN 97-19). Cracking has occurred in SS baffle former bolts in a number of foreign plants (NRC IN 98-11) and has now been observed in plants in the United States. Crack initiation and growth due to thermal and mechanical loading has occurred in high-pressure injection and safety injection piping (NRC IN 97-46 and NRC BL 88-08).

    PWR Secondary System: Steam generator tubes have experienced outside diameter stress corrosion cracking (ODSCC), intergranular attack (IGA), wastage, and pitting (NRC IN 97-88). Carbon steel support plates in steam generators have experienced general corrosion. Steam generator shells have experienced pitting and stress corrosion cracking (NRC INs 82-37, 85-65, and 90-04).

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NRC Licensee Event Report LER # 99-003-01, *Supplement to Reactor Recirculation B Loop, High Pressure Flow Element Venturi Instrument Line Steam Leakage Results in Unit 3 Shutdown Due to Fatigue Failure of Socket Welded Pipe Joint*, U.S. Nuclear Regulatory Commission, August 30, 1999.

XI.M2 WATER CHEMISTRY

Program Description

The main objective of this program is to mitigate damage caused by corrosion and stress corrosion cracking (SCC). The water chemistry program for boiling water reactors (BWRs) relies on monitoring and control of reactor water chemistry based on guidelines in the boiling water reactor vessel and internals project (BWRVIP)-29 (Electric Power Research Institute [EPRI] TR-103515). The BWRVIP-29 has three sets of guidelines: one for primary water, one for condensate and feedwater, and one for control rod drive (CRD) mechanism cooling water. The water chemistry program for pressurized water reactors (PWRs) relies on monitoring and control of reactor water chemistry based on the EPRI guidelines in TR-105714 for primary water chemistry and TR-102134 for secondary water chemistry.

The water chemistry programs are generally effective in removing impurities from intermediate and high flow areas. The Generic Aging Lessons Learned (GALL) report identifies those circumstances in which the water chemistry program is to be augmented to manage the effects of aging for license renewal. For example, the water chemistry program may not be effective in low flow or stagnant flow areas. Accordingly, in certain cases as identified in the GALL report, verification of the effectiveness of the chemistry control program is undertaken to ensure that significant degradation is not occurring and the component intended function will be maintained during the extended period of operation. As discussed in the GALL report for these specific cases, an acceptable verification program is a one-time inspection of selected components at susceptible locations in the system.

Evaluation and Technical Basis

1. ***Scope of Program:*** The program includes periodic monitoring and control of known detrimental contaminants such as chlorides, fluorides (PWRs only), dissolved oxygen, and sulfate concentrations below the levels known to result in loss of material or crack initiation and growth. Water chemistry control is in accordance with the guidelines in BWRVIP-29 (EPRI TR-103515) for water chemistry in BWRs; EPRI TR-105714, Rev. 3, for primary water chemistry in PWRs; EPRI TR-102134, Rev. 3, for secondary water chemistry in PWRs; or later revisions or updates of these reports as approved by the staff.

2. ***Preventive Actions:*** The program includes specifications for chemical species, sampling and analysis frequencies, and corrective actions for control of reactor water chemistry. System water chemistry is controlled to minimize contaminant concentration and mitigate loss of material due to general, crevice and pitting corrosion and crack initiation and growth caused by SCC. For BWRs, maintaining high water purity reduces susceptibility to SCC.

3. ***Parameters Monitored/Inspected:*** The concentration of corrosive impurities listed in the EPRI guidelines discussed above, which include chlorides, fluorides (PWRs only), sulfates, dissolved oxygen, and hydrogen peroxide, are monitored to mitigate degradation of structural materials. Water quality (pH and conductivity) is also maintained in accordance with the guidance. Chemical species and water quality are monitored by in process methods or through sampling. The chemistry integrity of the samples is maintained and verified to ensure that the method of sampling and storage will not cause a change in the concentration of the chemical species in the samples.

   BWR Water Chemistry: The guidelines in BWRVIP-29 (EPRI TR-103515) for BWR reactor water recommend that the concentration of chlorides, sulfates, and dissolved oxygen are monitored and kept below the recommended levels to mitigate corrosion. The two impurities, chlorides and sulfates, determine the coolant conductivity; dissolved oxygen, hydrogen peroxide, and hydrogen determine electrochemical potential (ECP). The EPRI guidelines recommend that the coolant conductivity and ECP are also monitored and kept below the recommended levels to mitigate SCC and corrosion in BWR plants. The EPRI guidelines in BWRVIP-29 (TR-103515) for BWR feedwater, condensate, and control rod drive water recommends that conductivity, dissolved oxygen level, and concentrations of iron and copper (feedwater only) are monitored and kept below the recommended levels to mitigate SCC. The EPRI guidelines in BWRVIP-29 (TR-103515) also include recommendations for controlling water chemistry in auxiliary systems: torus/pressure suppression chamber, condensate storage tank, and spent fuel pool.

PWR Primary Water Chemistry: The EPRI guidelines (EPRI TR-105714) for PWR primary water chemistry recommend that the concentration of chlorides, fluorides, sulfates, lithium, and dissolved oxygen and hydrogen are monitored and kept below the recommended levels to mitigate SCC of austenitic stainless steel, Alloy 600, and Alloy 690 components. TR-105714 provides guidelines for chemistry control in PWR auxiliary systems such as boric acid storage tank, refueling water storage tank, spent fuel pool, letdown purification systems, and volume control tank.

PWR Secondary Water Chemistry: The EPRI guidelines (EPRI TR-102134) for PWR secondary water chemistry recommend monitoring and control of chemistry parameters (e.g., pH level, cation conductivity, sodium, chloride, sulfate, lead, dissolved oxygen, iron, copper, and hydrazine) to mitigate steam generator tube degradation caused by denting, intergranular attack (IGA), outer diameter stress corrosion cracking (ODSCC), or crevice and pitting corrosion. The monitoring and control of these parameters, especially the pH level, also mitigates general (carbon steel components), crevice, and pitting corrosion of the steam generator shell and the balance of plant materials of construction (e.g., carbon steel, stainless steel, and copper).

4. ***Detection of Aging Effects:*** This is a mitigation program and does not provide for detection of any aging effects, such as loss of material and crack initiation and growth.

   In certain cases as identified in the GALL report, inspection of select components is to be undertaken to verify the effectiveness of the chemistry control program and to ensure that significant degradation is not occurring and the component intended function will be maintained during the extended period of operation.

5. ***Monitoring and Trending:*** The frequency of sampling water chemistry varies (e.g., continuous, daily, weekly, or as needed) based on plant operating conditions and the EPRI water chemistry guidelines. Whenever corrective actions are taken to address an abnormal chemistry condition, increased sampling is utilized to verify the effectiveness of these actions.

6. ***Acceptance Criteria:*** Maximum levels for various contaminants are maintained below the system specific limits as indicated by the limits specified in the corresponding EPRI water chemistry guidelines. Any evidence of the presence of aging effects or unacceptable water chemistry results is evaluated, the root cause identified, and the condition corrected.

7. ***Corrective Actions:*** When measured water chemistry parameters are outside the specified range, corrective actions are taken to bring the parameter back within the acceptable range and within the time period specified in the EPRI water chemistry guidelines. As discussed in

the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable in addressing corrective actions.

8. ***Confirmation Process:*** Following corrective actions, additional samples are taken and analyzed to verify that the corrective actions were effective in returning the concentrations of contaminants such as chlorides, fluorides, sulfates, dissolved oxygen, and hydrogen peroxide to within the acceptable ranges. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable in addressing the confirmation process.

9. ***Administrative Controls:*** Site quality assurance (QA) procedures, review and approval processes, and administrative controls are implemented in accordance with the requirements of 10 CFR Part 50, Appendix B. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable in addressing administrative controls.

10. ***Operating Experience:*** The EPRI guideline documents have been developed based on plant experience and have been shown to be effective over time with their widespread use. The specific examples of operating experience are as follows:

    BWR: Intergranular stress corrosion cracking (IGSCC) has occurred in small- and large-diameter BWR piping made of austenitic stainless steels and nickel-base alloys. Significant cracking has occurred in recirculation, core spray, residual heat removal (RHR) systems, and reactor water cleanup (RWCU) system piping welds. IGSCC has also occurred in a number of vessel internal components, including core shroud, access hole cover, top guide, and core spray spargers (Nuclear Regulatory Commission [NRC] Information Bulletin 80-13, NRC Information Notice [IN] 95-17, NRC General Letter [GL] 94-03, and NUREG-1544). No occurrence of SCC in piping and other components in standby liquid control systems exposed to sodium pentaborate solution has ever been reported (NUREG/CR-6001).

    PWR Primary System: The primary pressure boundary piping of PWRs has generally not been found to be affected by SCC because of low dissolved oxygen levels and control of primary water chemistry. However, the potential for SCC exists due to inadvertent introduction of contaminants into the primary coolant system from unacceptable levels of contaminants in the boric acid; introduction through the free surface of the spent fuel pool, which can be a natural collector of airborne contaminants; or introduction of oxygen during cooldown (NRC IN 84–18). Ingress of demineralizer resins into the primary system has caused IGSCC of Alloy 600 vessel head penetrations (NRC IN 96-11, NRC GL 97-01). Inadvertent introduction of sodium thiosulfate into the primary system has caused IGSCC of steam generator tubes. The SCC has occurred in safety injection lines (NRC INs 97-19 and 84-18), charging pump casing cladding (NRC INs 80-38 and 94-63), instrument nozzles in safety injection tanks (NRC IN 91-05), and safety-related SS piping systems that contain oxygenated, stagnant, or essentially stagnant borated coolant (NRC IN 97-19). Steam generator tubes and plugs and Alloy 600 penetrations have experienced primary water stress corrosion cracking (PWSCC) (NRC INs 89-33, 94-87, 97-88, 90-10, and 96-11; NRC Bulletin 89-01 and its two supplements).

    PWR Secondary System: Steam generator tubes have experienced ODSCC, IGA, wastage, and pitting (NRC IN 97-88, NRC GL 95-05). Carbon steel support plates in steam generators have experienced general corrosion. The steam generator shell has experienced pitting and stress corrosion cracking (NRC INs 82-37, 85-65, and 90-04).

Such operating experience has provided feedback to revisions of the EPRI water chemistry guideline documents.

References

BWRVIP-29 (EPRI TR-103515), *BWR Water Chemistry Guidelines-1993 Revision, Normal and Hydrogen Water Chemistry*, Electric Power Research Institute, Palo Alto, CA, February 1994.

EPRI TR-102134, *PWR Secondary Water Chemistry Guideline-Revision 3*, Electric Power Research Institute, Palo Alto, CA, May 1993.

EPRI TR-105714, *PWR Primary Water Chemistry Guidelines–Revision 3*, Electric Power Research Institute, Palo Alto, CA, Nov. 1995.

NRC IE Bulletin 80-13, *Cracking in Core Spray Piping*, U.S. Nuclear Regulatory Commission, May 12, 1980.

NRC IE Bulletin 89-01, *Failure of Westinghouse Steam Generator Tube Mechanical Plugs*, U.S. Nuclear Regulatory Commission, May 15, 1989.

NRC IE Bulletin 89-01, Supplement 1, *Failure of Westinghouse Steam Generator Tube Mechanical Plugs*, U.S. Nuclear Regulatory Commission, November 14, 1989.

NRC IE Bulletin 89-01, Supplement 2, *Failure of Westinghouse Steam Generator Tube Mechanical Plugs*, U.S. Nuclear Regulatory Commission, June 28, 1991.

NRC Generic Letter 94-03, *Intergranular Stress Corrosion Cracking of Core Shrouds in Boiling Water Reactors*, U.S. Nuclear Regulatory Commission, July 25, 1994.

NRC Generic Letter 95-05, *Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking*, U.S. Nuclear Regulatory Commission, August 3, 1995.

NRC Generic Letter 97-01, *Degradation of Control Rod Drive Mechanism Nozzle and Other Vessel Closure Head Penetrations*, U.S. Nuclear Regulatory Commission, April 1,1997.

NRC Information Notice 80-38, *Cracking In Charging Pump Casing Cladding*, U.S. Nuclear Regulatory Commission, October 31, 1980.

NRC Information Notice 82-37, *Cracking in the Upper Shell to Transition Cone Girth Weld of a Steam Generator at an Operating PWR*, U.S. Nuclear Regulatory Commission, September 16,1982.

NRC Information Notice 84-18, *Stress Corrosion Cracking in Pressurized Water Reactor Systems*, U.S. Nuclear Regulatory Commission, March 7, 1984.

NRC Information Notice 85-65, *Crack Growth in Steam Generator Girth Welds*, U.S. Nuclear Regulatory Commission, July 31, 1985.

NRC Information Notice 89-33, *Potential Failure of Westinghouse Steam Generator Tube Mechanical Plugs*, U.S. Nuclear Regulatory Commission, March 23, 1989.

NRC Information Notice 90-04, *Cracking of the Upper Shell-to-Transition Cone Girth Welds in Steam Generators*, U.S. Nuclear Regulatory Commission, January 26, 1990.

NRC Information Notice 90-10, *Primary Water Stress Corrosion Cracking (PWSCC) of Inconel 600*, U.S. Nuclear Regulatory Commission, February 23, 1990.

NRC Information Notice 91-05, *Intergranular Stress Corrosion Cracking In Pressurized Water Reactor Safety Injection Accumulator Nozzles*, U.S. Nuclear Regulatory Commission, January 30, 1991.

NRC Information Notice 94-63, *Boric Acid Corrosion of Charging Pump Casing Caused by Cladding Cracks*, U.S. Nuclear Regulatory Commission, August 30, 1994.

NRC Information Notice 94-87, *Unanticipated Crack in a Particular Heat of Alloy 600 Used for Westinghouse Mechanical Plugs for Steam Generator Tubes*, U.S. Nuclear Regulatory Commission, December 22, 1994.

NRC Information Notice 95-17, *Reactor Vessel Top Guide and Core Plate Cracking*, U.S. Nuclear Regulatory Commission, March 10, 1995.

NRC Information Notice 96-11, *Ingress of Demineralizer Resins Increase Potential for Stress Corrosion Cracking of Control Rod Drive Mechanism Penetrations*, U.S. Nuclear Regulatory Commission, February 14, 1996.

NRC Information Notice 97-19, *Safety Injection System Weld Flaw at Sequoyah Nuclear Power Plant, Unit 2*, U.S. Nuclear Regulatory Commission, April 18, 1997.

NRC Information Notice 97-88, *Experiences during Recent Steam Generator Inspections*, U.S. Nuclear Regulatory Commission, December 16, 1997.

NUREG-1544*, Status Report: Intergranular Stress Corrosion Cracking of BWR Core Shrouds and Other Internal Components*, U.S. Nuclear Regulatory Commission, March 1996.

NUREG/CR-6001, *Aging Assessment of BWR Standby Liquid Control Systems,* G. D. Buckley, R. D. Orton, A. B. Johnson Jr., and L. L. Larson, 1992.

XI.M3 Reactor Head Closure Studs

Program Description

This program includes (a) inservice inspection (ISI) in conformance with the requirements of the American Society of Mechanical Engineers (ASME), Code, Section XI, Subsection IWB (1995 edition through the 1996 addenda), Table IWB 2500-1, and (b) preventive measures to mitigate cracking.

Evaluation and Technical Basis

1. ***Scope of Program:*** The program includes (a) ISI to detect crack initiation and growth due to stress corrosion cracking (SCC) or intergranular stress corrosion cracking (IGSCC); loss of material due to wear; and coolant leakage from reactor vessel closure stud bolting for both boiling water reactors (BWRs) and pressurized water reactors (PWRs), and (b) preventive measures of NRC Regulatory Guide 1.65 to mitigate cracking. The program is applicable to closure studs and nuts constructed from materials with a maximum tensile strength limited to less than 1,172 MPa (170 ksi) (Nuclear Regulatory Commission [NRC] Regulatory Guide [RG] 1.65).

2. ***Preventive Actions:*** Preventive measures include avoiding the use of metal-plated stud bolting to prevent degradation due to corrosion or hydrogen embrittlement and to use manganese phosphate or other acceptable surface treatments and stable lubricants (RG 1.65). Implementation of these mitigation measures is an effective option for reducing SCC or IGSCC and for this program to be effective.

3. ***Parameters Monitored/Inspected:*** The ASME Section XI ISI program detects and sizes cracks, detects loss of material, and detects coolant leakage by following the examination and inspection requirements specified in Table IWB-2500-1.

4. ***Detection of Aging Effects:*** The extent and schedule of the inspection and test techniques prescribed by the program are designed to maintain structural integrity and ensure that aging effects will be discovered and repaired before the loss of intended function of the component. Inspection can reveal crack initiation and growth, loss of material due to corrosion or wear, and leakage of coolant.

   The program uses visual, surface, and volumetric examinations in accordance with the general requirements of Subsection IWA-2000. Surface examination uses magnetic particle, liquid penetration, or eddy current examinations to indicate the presence of surface discontinuities and flaws. Volumetric examination uses radiographic, ultrasonic, or eddy current examinations to indicate the presence of discontinuities or flaws throughout the volume of material. Visual VT-2 examination detects evidence of leakage from pressure-retaining components, as required during the system pressure test.

   Components are examined and tested as specified in Table IWB-2500-1. Examination category B-G-1, for pressure-retaining bolting greater than 2 in. in diameter in reactor vessels specifies volumetric examination of studs in place, from the top of the nut to the bottom of the flange hole, and surface and volumetric examination of studs when removed. Also specified are volumetric examination of flange threads and visual VT-1 examination of surfaces of nuts, washers, and bushings. Examination category B-P for all pressure-retaining components, specifies visual VT-2 examination of all pressure-retaining boundary components during the system leakage test and the system hydrostatic test.

5. ***Monitoring and Trending:*** The Inspection schedule of IWB-2400, and the extent and frequency of IWB-2500-1 provide timely detection of cracks, loss of material, and leakage.

6. ***Acceptance Criteria:*** Any indication or relevant condition of degradation in closure stud bolting is evaluated in accordance with IWB-3100 by comparing ISI results with the acceptance standards of IWB-3400 and IWB-3500.

7. ***Corrective Actions:*** Repair and replacement are in conformance with the requirements of IWB-400 and IWB-7000, respectively, and the material and inspection guidance of RG 1.65. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable in addressing corrective actions.

8. ***Confirmation Process:*** Site quality assurance (QA) procedures, review and approval processes, and administrative controls are implemented in accordance with the requirements of 10 CFR Part 50, Appendix B. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable in addressing the confirmation process and administrative controls.

9. ***Administrative Controls:*** See Item 8, above.

10. ***Operating Experience:*** The SCC has occurred in BWR pressure vessel head studs (Stoller 1991). The aging management program (AMP) has provisions regarding inspection techniques and evaluation, material specifications, corrosion prevention, and other aspects of reactor pressure vessel head stud cracking. Implementation of the program provides reasonable assurance that the effects of cracking due to SCC or IGSCC and loss of material due to wear will be adequately managed so that the intended functions of the reactor head closure studs and bolts will be maintained consistent with the current licensing basis (CLB) for the period of extended operation.

References

10 CFR 50.55a, *Codes and Standards*, Office of the Federal Register, National Archives and Records Administration, 2000.

ASME Section XI, *Rules for Inservice Inspection of Nuclear Power Plant Components*, ASME Boiler and Pressure Vessel Code, 1995 edition through the 1996 addenda, American Society of Mechanical Engineers, New York, NY.

NRC Regulatory Guide 1.65, *Material and Inspection for Reactor Vessel Closure Studs*, U.S. Nuclear Regulatory Commission, October 1973.

Stoller, S. M., *Reactor Head Closure Stud Cracking, Material Toughness Outside FSAR - SCC in Thread Roots*, Nuclear Power Experience, BWR-2, III, 58, p. 30, 1991.

XI.M4 BWR VESSEL ID ATTACHMENT WELDS

Program Description

The program includes (a) inspection and flaw evaluation in accordance with the guidelines of staff-approved boiling water reactor vessel and internals project (BWRVIP)-48 and (b) monitoring and control of reactor coolant water chemistry in accordance with the guidelines of BWRVIP-29 (Electric Power Research Institute [EPRI] TR-103515) to ensure the long-term integrity and safe operation of boiling water reactor (BWR) vessel inside diameter (ID) attachment welds.

Evaluation and Technical Basis

1. ***Scope of Program:*** The program is focused on managing the effects of crack initiation and growth due to stress corrosion cracking (SCC), including intergranular stress corrosion cracking (IGSCC). The program contains preventive measures to mitigate SCC; inservice inspection (ISI) to detect cracking and monitor the effects of cracking on the intended function of the components; and repair and/or replacement, as needed, to maintain the ability to perform the intended function.

   The guidelines of BWRVIP-48 include inspection recommendations and evaluation methodologies for the attachment welds between the vessel wall and vessel ID brackets that attach safety-related components to the vessel (e.g., jet pump riser braces and core-spray piping brackets). In some cases, the attachment is a simple weld; in others, it includes a weld build-up pad on the vessel. The BWRVIP-48 guidelines include information on the geometry of the vessel ID attachments; evaluate susceptible locations and safety consequence of failure; provide recommendations regarding the method, extent, and frequency of inspection; and discuss acceptable methods for evaluating the structural integrity significance of flaws detected during these examinations.

2. ***Preventive Actions:*** The BWRVIP-48 provides guidance on detection, but does not provide guidance on methods to mitigate cracking. Maintaining high water purity reduces susceptibility to SCC or IGSCC. Reactor coolant water chemistry is monitored and maintained in accordance with the guidelines in BWRVIP-29 (EPRI TR-103515). The program description and evaluation and technical basis of monitoring and maintaining reactor water chemistry are presented in Section XI.M2, “Water Chemistry.”

3. ***Parameters Monitored/Inspected:*** The program monitors the effects of SCC and IGSCC on the intended function of vessel attachment welds by detection and sizing of cracks by ISI in accordance with the guidelines of approved BWRVIP-48 and the requirements of the American Society of Mechanical Engineers (ASME) Code, Section XI, Table IWB 2500-1 (1995 edition through the 1996 addenda). An applicant may use the guidelines of BWRVIP-62 for inspection relief for vessel internal components with hydrogen water chemistry.

4. ***Detection of Aging Effects:*** The extent and schedule of the inspection and test techniques prescribed by BWRVIP-48 guidelines are designed to maintain structural integrity and ensure that aging effects will be discovered and repaired before the loss of intended function. Inspection can reveal crack initiation and growth. Vessel ID attachment welds are inspected in accordance with the requirements of ASME Section XI, Subsection IWB, examination category B-N-2. The Section XI inspection specifies visual VT-1 examination to detect discontinuities and imperfections on the surfaces of components and visual VT-3

examination to determine the general mechanical and structural condition of the component supports. The inspection and evaluation guidelines of BWRVIP-48 recommend more stringent inspections for certain selected attachments. The guidelines recommend enhanced visual VT-1 examination of all safety-related attachments and those nonsafety-related attachments identified as being susceptible to IGSCC. Visual VT-1 examination is capable of achieving 1/32 in. resolution; the enhanced visual VT-1 examination method is capable of achieving a 1-mil wire resolution. The nondestructive examination (NDE) techniques appropriate for inspection of BWR vessel internals and their implementation needs, including the uncertainties inherent in delivering and executing NDE techniques in a BWR, are included in BWRVIP-03.

5. ***Monitoring and Trending:*** Inspections scheduled in accordance with IWB-2400 and approved BWRVIP-48 guidelines provide timely detection of cracks. If flaws are detected, the scope of examination is expanded.

6. ***Acceptance Criteria:*** Any indication detected is evaluated in accordance with ASME Section XI or the staff-approved BWRVIP-48 guidelines. Applicable and approved BWRVIP-14, BWRVIP-59, and BWRVIP-60 documents provide guidelines for evaluation of crack growth in stainless steels (SSs), nickel alloys, and low-alloy steels, respectively.

7. ***Corrective Actions:*** Repair and replacement procedures are equivalent to those requirements in the ASME Section XI. Repair is in conformance with IWB-4000 and replacement occurs according to IWB-7000. As discussed in the appendix to this report, the staff finds that licensee implementation of the guidelines in BWRVIP-48, as modified, will provide an acceptable level of quality for inspection and flaw evaluation of the safety-related components addressed in accordance with 10 CFR Part 50, Appendix B, corrective actions.

8. ***Confirmation Process:*** Site quality assurance (QA) procedures, review and approval processes, and administrative controls are implemented in accordance with the requirements of 10 CFR Part 50, Appendix B. As discussed in the appendix to this report, the staff finds that licensee implementation of the guidelines in BWRVIP-48, as modified, will provide an acceptable level of quality for inspection and flaw evaluation of the safety-related components addressed in accordance with the 10 CFR Part 50, Appendix B, confirmation process and administrative controls.

9. ***Administrative Controls:*** See Item 8, above.

10. ***Operating Experience:*** Cracking due to SCC/IGSCC has occurred in BWR components. The program guidelines are based on evaluation of available information, including BWR inspection data and information on the elements that cause IGSCC, to determine which attachment welds may be susceptible to cracking. Implementation of the program provides reasonable assurance that crack initiation and growth will be adequately managed and the intended functions of the vessel ID attachments will be maintained consistent with the current licensing basis (CLB) for the period of extended operation.

References

10 CFR 50.55a, *Codes and Standards*, Office of the Federal Register, National Archives and Records Administration, 2000.

ASME Section XI, *Rules for Inservice Inspection of Nuclear Power Plant Components*, ASME Boiler and Pressure Vessel Code, 1995 edition through the 1996 addenda, American Society of Mechanical Engineers, New York, NY.

BWRVIP-03, *BWR Vessel and Internals Project, Reactor Pressure Vessel and Internals Examination Guidelines, (EPRI TR-105696 R1, March 30, 1999),* Final Safety Evaluation Report by the Office of Nuclear Reactor Regulation for BWRVIP-03, July 15, 1999.

BWRVIP-14, *Evaluation of Crack Growth in BWR Stainless Steel RPV Internals, (EPRI TR-105873, July 11, 2000),* Final Safety Evaluation Report by the Office of Nuclear Reactor Regulation for BWRVIP-14, December 3, 1999.

BWRVIP-29, *BWR Vessel and Internals Project*, *BWR Water Chemistry Guidelines-1993 Revision, Normal and Hydrogen Water Chemistry*, *(EPRI TR-103515),* Electric Power Research Institute, Palo Alto, CA, February 1994.

BWRVIP-48, *BWR Vessel and Internals Project, Vessel ID Attachment Weld Inspection and Flaw Evaluation Guidelines, (EPRI TR-108724, February 1998),* Final License Renewal Safety Evaluation Report by the Office of Nuclear Reactor Regulation for BWRVIP-48 for Compliance with the License Renewal Rule (10 CFR Part 54), January 17, 2001.

BWRVIP-59, *Evaluation of Crack Growth in BWR Nickel-Base Austenitic Alloys in RPV Internals, (EPRI TR-108710),* BWRVIP and Electric Power Research Institute, Palo Alto, CA, March 24, 2000.

BWRVIP-60, *BWR Vessel and Internals Project, Evaluation of Crack Growth in BWR Low Alloy Steel RPV Internals, (EPRI TR-108709, April 14, 2000),* Final Safety Evaluation Report by the Office of Nuclear Reactor Regulation for BWRVIP-60, July 8, 1999.

BWRVIP-62, *BWR Vessel and Internals Project, Technical Basis for Inspection Relief for BWR Internal Components with Hydrogen Injection, (EPRI TR-108705),* BWRVIP and Electric Power Research Institute, Palo Alto, CA, March 7, 2000.

XI.M5 BWR FEEDWATER NOZZLE

Program Description

This program includes (a) enhanced inservice inspection (ISI) in accordance with the requirements of the American Society of Mechanical Engineers (ASME) Code, Section XI, Subsection IWB, Table IWB 2500-1 (1995 edition through the 1996 addenda) and the recommendation of General Electric (GE) NE-523-A71-0594, and (b) system modifications to mitigate cracking. The program specifies periodic ultrasonic inspection of critical regions of boiling water reactor (BWR) feedwater nozzle.

Evaluation and Technical Basis

1. ***Scope of Program:*** The program includes enhanced ISI to monitor the effects of crack initiation and growth on the intended function of the component and systems modifications to mitigate cracking.

2. ***Preventive Actions:*** Mitigation occurs by systems modifications, such as removal of stainless steel cladding and installation of improved spargers. Mitigation is also accomplished by changes to plant-operating procedures, such as improved feedwater control and rerouting of the reactor water cleanup system, to decrease the magnitude and frequency of temperature fluctuations.

3. ***Parameters Monitored/Inspected:*** The aging management program (AMP) monitors the effects of cracking on the intended function of the component by detection and sizing of cracks by ISI in accordance with ASME Section XI, Subsection IWB and the recommendation of GE NE-523-A71-0594, as described below.

4. ***Detection of Aging Effects:*** The extent and schedule of the inspection prescribed by the program are designed to ensure that aging effects will be discovered and repaired before the loss of intended function of the component. Inspection can reveal crack initiation and growth. The GE NE-523-A71-0594 specifies UT of specific regions of the blend radius and bore. The UT examination techniques and personnel qualifications are in accordance with to the guidelines of GE NE-523-A71-0594. Based on the inspection method and techniques and plant-specific fracture mechanics assessments, the inspection schedule is in accordance with Table 6-1 of GE NE-523-A71-0594. Leakage monitoring may be used to modify the inspection interval.

5. ***Monitoring and Trending:*** Inspections scheduled in accordance with GE NE-523-A71-0594 provides timely detection of cracks.

6. ***Acceptance Criteria:*** Any cracking is evaluated in accordance with IWB-3100 by comparing inspection results with the acceptance standards of IWB-3400 and IWB-3500.

7. ***Corrective Actions:*** Repair is in conformance with IWB-4000 and replacement in accordance with IWB-7000. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable in addressing corrective actions.

8. ***Confirmation Process:*** Site quality assurance (QA) procedures, review and approval processes, and administrative controls are implemented in accordance with the requirements of 10 CFR Part 50, Appendix B. As discussed in the appendix to this report,

the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable in addressing the confirmation process and administrative controls.

9. ***Administrative Controls:*** See Item 8, above.

10. ***Operating Experience:*** Cracking has occurred in several BWR plants (NUREG-0619, NRC Generic Letter 81-11). This AMP has been implemented for nearly 20 years and found to be effective in managing the effect of cracking on the intended function of feedwater nozzles.

References

ASME Section XI, *Rules for Inservice Inspection of Nuclear Power Plant Components*, ASME Boiler and Pressure Vessel Code, 1995 edition through the 1996 addenda, American Society of Mechanical Engineers, New York, NY.

GE-NE-523-A71-0594, Rev. 1, *Alternate BWR Feedwater Nozzle Inspection Requirements*, BWR Owner's Group, August 1999.

NRC Generic Letter 81-11, *BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking (NUREG-0619),* U.S. Nuclear Regulatory Commission, February 29, 1981.

NUREG-0619, *BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking*, U.S. Nuclear Regulatory Commission, November 1980.

XI.M6 BWR CONTROL ROD DRIVE RETURN LINE NOZZLE

Program Description

This program includes (a) enhanced inservice inspection (ISI) in conformance with the American Society of Mechanical Engineers (ASME) Code, Section XI, Subsection IWB, Table IWB 2500-1 (1995 edition through the 1996 addenda) and the recommendations of NUREG-0619, and (b) system modifications and maintenance programs to mitigate cracking. The program specifies periodic liquid penetrant and ultrasonic inspection of critical regions of boiling water reactor (BWR) control rod drive return line (CRDRL) nozzle.

Evaluation and Technical Basis

1. ***Scope of Program:*** The program includes systems modifications, enhanced ISI, and maintenance programs to monitor the effects of crack initiation and growth on the intended function of CRDRL nozzles.

2. ***Preventive Actions:*** Mitigation occurs by system modifications, such as rerouting the CRDRL to a system that connects to the reactor vessel. For some classes of BWRs, or those that can prove satisfactory system operation, mitigation also is accomplished by confirmation of proper return flow capability, two-pump operation and cutting and capping the CRDRL nozzle without rerouting.

3. ***Parameters Monitored/ Inspected:*** The aging management program (AMP) monitors the effects of cracking on the intended function of the component by detecting and sizing cracks by ISI in accordance with Table IWB 2500-1 and NUREG-0619.

4. ***Detection of Aging Effects:*** The extent and schedule of inspection, as delineated in NUREG 0619, assures detection of cracks before the loss of intended function of the component. Inspection recommendations include liquid penetrant testing (PT) of the CRDRL nozzle blend radius and bore regions and the reactor vessel wall area beneath the nozzle, return-flow-capacity demonstration, CRD-system-performance testing and ultrasonic inspection of welded connections in the rerouted line. The inspection is to include base metal to a distance of one-pipe-wall thickness or 0.5 in., whichever is greater, on both sides of the weld.

5. ***Monitoring and Trending:*** The inspection schedule of NUREG-0619 provides timely detection of cracks.

6. ***Acceptance Criteria:*** Any cracking is evaluated in accordance with IWB-3100 by comparing inspection results with the acceptance standards of IWB-3400 and IWB-3500. All cracks found in the CRDRL nozzles are to be removed by grinding.

7. ***Corrective Actions:*** Repair is in conformance with IWB-4000 and replacement in accordance with IWB-7000. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable in addressing corrective actions.

8. ***Confirmation Process:*** Site quality assurance (QA) procedures, review and approval processes, and administrative controls are implemented in accordance with the requirements of 10 CFR Part 50, Appendix B. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable in addressing the confirmation process and administrative controls.

9. ***Administrative Controls:*** See Item 8, above.

10. ***Operating Experience:*** Cracking has occurred in several BWR plants (NUREG-0619). The present AMP has been implemented for nearly 20 years and found to be effective in managing the effect of cracking on the intended function of CRDRL nozzles.

References

10 CFR 50.55a, *Codes and Standards*, Office of the Federal Register, National Archives and Records Administration, 2000.

ASME Section XI, *Rules for Inservice Inspection of Nuclear Power Plant Components*, ASME Boiler and Pressure Vessel Code, 1995 edition through the 1996 addenda, American Society of Mechanical Engineers, New York, NY.

NUREG-0619, *BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking*, U.S. Nuclear Regulatory Commission, November 1980.

XI.M7 BWR STRESS CORROSION CRACKING

Program Description

The program to manage intergranular stress corrosion cracking (IGSCC) in boiling water reactor (BWR) coolant pressure boundary piping made of stainless steel (SS) is delineated in NUREG-0313, Rev. 2, and Nuclear Regulatory Commission (NRC) Generic Letter (GL) 88-01 and its Supplement 1. The program includes (a) preventive measures to mitigate IGSCC, and (b) inspection and flaw evaluation to monitor IGSCC and its effects. The staff-approved boiling water reactor vessel and internals project (BWRVIP)-75 report allows for modifications of inspection scope in the GL 88-01 program.

Evaluation and Technical Basis

1. ***Scope of Program:*** The program focuses on (a) managing and implementing countermeasures to mitigate IGSCC and (b) performing inservice inspection (ISI) to monitor IGSCC and its effects on the intended function of BWR components. The program is applicable to all BWR piping made of austenitic SS that is 4 in. or larger in nominal diameter and contains reactor coolant at a temperature above 93°C (200°F) during power operation, regardless of code classification. The program also applies to pump casings, valve bodies and reactor vessel attachments and appurtenances, such as head spray and vent components. NUREG-0313 and NRC GL-88-01, respectively, describe the technical basis and staff guidance regarding mitigation of IGSCC in BWRs. Attachment A of NRC GL 88-01 delineates the staff-approved positions regarding materials, processes, water chemistry, weld overlay reinforcement, partial replacement, stress improvement of cracked welds, clamping devices, crack characterization and repair criteria, inspection methods and personnel, inspection schedules, sample expansion, leakage detection, and reporting requirements.

2. ***Preventive Actions:*** The comprehensive program outlined in NUREG-0313 and NRC GL 88-01 addresses improvements in all three elements that, in combination, cause IGSCC. These elements consist of a susceptible (sensitized) material, a significant tensile stress, and an aggressive environment. Sensitization of nonstabilized austenitic SSs containing greater than 0.03 wt.% carbon involves precipitation of chromium carbides at the grain boundaries during certain fabrication or welding processes. The formation of carbides creates an envelope of chromium depleted region that, in certain environments, is susceptible to stress corrosion cracking (SCC). Residual tensile stresses are introduced from fabrication processes, such as welding, surface grinding, or forming. High levels of dissolved oxygen or aggressive contaminants, such as sulfates or chlorides, accelerate the SCC processes.

   The program delineated in NUREG-0313 and NRC GL 88-01 and in the staff-approved BWRVIP-75 report includes recommendations regarding selection of materials that are resistant to sensitization, use of special processes that reduce residual tensile stresses, and monitoring and maintenance of coolant chemistry. The resistant materials are used for new and replacement components and include low-carbon grades of austenitic SS and weld metal, with a maximum carbon of 0.035 wt.% and a minimum ferrite of 7.5% in weld metal and cast austenitic stainless steel (CASS). Inconel 82 is the only commonly used nickel-base weld metal considered to be resistant to SCC; other nickel-alloys, such as Alloy 600 are evaluated on an individual basis. Special processes are used for existing, new, and replacement components. These processes include solution heat treatment, heat sink welding, induction heating, and mechanical stress improvement.

The program delineated in NUREG-0313 and NRC GL 88-01 does not provide specific guidelines for controlling reactor water chemistry to mitigate IGSCC. Maintaining high water purity reduces susceptibility to SCC or IGSCC. Reactor coolant water chemistry is monitored and maintained in accordance with the guidelines in BWRVIP-29 (Electric Power Research Institute [EPRI] TR-103515). The program description, and evaluation and technical basis of monitoring and maintaining reactor water chemistry are presented in Section XI.M2, “Water Chemistry.”

3. ***Parameters Monitored/Inspected:*** The program detects and sizes cracks and detects leakage by using the examination and inspection guidelines delineated in NUREG 0313, Rev. 2, and NRC GL 88-01 or the referenced BWRVIP-75 guideline as approved by the NRC staff.

4. ***Detection of Aging Effects:*** The extent, method, and schedule of the inspection and test techniques delineated in NRC GL 88-01 or BWRVIP-75 are designed to maintain structural integrity and ensure that aging effects will be discovered and repaired before the loss of intended function of the component. The program uses volumetric examinations to detect IGSCC.

   The NRC GL 88-01 recommends that the detailed inspection procedure, equipment, and examination personnel be qualified by a formal program approved by the NRC. These inspection guidelines, updated in the approved BWRVIP-75 document, provide the technical basis for revisions to NRC GL 88-01 inspection schedules. Inspection can reveal crack initiation and growth and leakage of coolant. The extent and frequency of inspection recommended by the program are based on the condition of each weld (e.g., whether the weldments were made from IGSCC-resistant material, whether a stress improvement process was applied to a weldment to reduce residual stresses, and how the weld was repaired if it had been cracked). The inspection guidance in approved BWRVIP-75 replaces the extent and schedule of inspection in NRC GL 88-01.

5. ***Monitoring and Trending:*** The extent and schedule for inspection, in accordance with the recommendations of NRC GL 88-01 or approved BWRVIP-75 guidelines, provide timely detection of cracks and leakage of coolant. Based on inspection results, NRC GL 88-01 or approved BWRVIP-75 guidelines provide guidelines for additional samples of welds to be inspected when one or more cracked welds are found in a weld category.

6. ***Acceptance Criteria:*** As recommended in NRC GL 88-01, any indication detected is evaluated in accordance with the ASME Section XI, Subsection IWB-3640 (1995 edition through the 1996 addenda) and the guidelines of NUREG-0313.

   Applicable and approved BWRVIP-14, BWRVIP-59, BWRVIP-60, and BWRVIP-62 documents provide guidelines for evaluation of crack growth in SSs, nickel alloys, and low-alloy steels. An applicant may use BWRVIP-61 guidelines for BWR vessel and internals induction heating stress improvement effectiveness on crack growth in operating plants.

7. ***Corrective Actions***: The guidance for weld overlay repair and stress improvement or replacement is provided in NRC GL 88-01; ASME Section XI, Subsections IWB-4000 and IWB-7000, IWC-4000 and IWC-7000, or IWD-4000 and IWD-7000, respectively for Class 1, 2, or 3 components; and ASME Code Case N 504-1. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable in addressing corrective actions.

8. ***Confirmation Process:*** Site quality assurance (QA) procedures, review and approval processes, and administrative controls are implemented in accordance with the requirements of 10 CFR Part 50, Appendix B. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable in addressing the confirmation process and administrative controls.

9. ***Administrative Controls:*** See Item 8, above.

10. ***Operating Experience:*** Intergranular stress corrosion cracking has occurred in small- and large-diameter BWR piping made of austenitic stainless steel and nickel-base alloys. Cracking has occurred in recirculation, core spray, residual heat removal (RHR), and reactor water cleanup (RWCU) system piping welds (NRC GL 88-01, NRC Information Notices [INs] 82-39 and 84-41). The comprehensive program outlined in NRC GL 88-01 and NUREG-0313 and in the staff-approved BWRVIP-75 report addresses mitigating measures for SCC or IGSCC (e.g., susceptible material, significant tensile stress, and an aggressive environment). The GL 88-01 program has been effective in managing IGSCC in BWR reactor coolant pressure-retaining components and the revision to the GL 88-01 program, according to the staff-approved BWRVIP-75 report, will adequately manage IGSCC degradation.

References

10 CFR 50.55a, *Codes and Standards*, Office of the Federal Register, National Archives and Records Administration, 2000.

ASME Code Case N-504-1, *Alternative Rules for Repair of Class 1, 2, and 3 Austenitic Stainless Steel Piping*, Section XI, Division 1, 1995 Edition, ASME Boiler and Pressure Vessel Code – Code Cases – Nuclear Components, American Society of Mechanical Engineers, New York, NY.

ASME Section XI, *Rules for Inservice Inspection of Nuclear Power Plant Components*, ASME Boiler and Pressure Vessel Code, 1995 edition through the 1996 addenda, American Society of Mechanical Engineers, New York, NY.

BWRVIP-14, *Evaluation of Crack Growth in BWR Stainless Steel RPV Internals, (EPRI TR-105873, July 11, 2000),* Final Safety Evaluation Report by the Office of Nuclear Reactor Regulation for BWRVIP-14, December 3, 1999.

BWRVIP-29 (EPRI TR-103515), *BWR Vessel and Internals Project, BWR Water Chemistry Guidelines-1993 Revision, Normal and Hydrogen Water Chemistry*, Electric Power Research Institute, Palo Alto, CA, February 1994.

BWRVIP-59, *Evaluation of Crack Growth in BWR Nickel-Base Austenitic Alloys in RPV Internals, (EPRI TR-108710),* BWRVIP and Electric Power Research Institute, Palo Alto, CA, March 24, 2000.

BWRVIP-60, *BWR Vessel and Internals Project, Evaluation of Crack Growth in BWR Low Alloy Steel RPV Internals, (EPRI TR-108709, April 14, 2000),* Final Safety Evaluation Report by the Office of Nuclear Reactor Regulation for BWRVIP-60, July 8, 1999.

BWRVIP-61, *BWR Vessel and Internals Induction Heating Stress Improvement Effectiveness on Crack Growth in Operating Reactors, (EPRI TR-112076),* BWRVIP and Electric Power Research Institute, Palo Alto, CA, January 29, 1999.

BWRVIP-62, *BWR Vessel and Internals Project, Technical Basis for Inspection Relief for BWR Internal Components with Hydrogen Injection, (EPRI TR-108705),* BWRVIP and Electric Power Research Institute, Palo Alto, CA, March 7, 2000.

BWRVIP-75, *Technical Basis for Revisions to Generic Letter 88-01 Inspection Schedules (NUREG-0313), (EPRI TR-113932, Feb. 29, 2000),* Initial Safety Evaluation Report by the Office of Nuclear Reactor Regulation for BWRVIP-75, September 15, 2000.

NRC Generic Letter 88-01, *NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping*, U.S. Nuclear Regulatory Commission, January 25, 1988; Supplement 1, February 4, 1992.

NRC Information Notice 82-39, *Service Degradation of Thick Wall Stainless Steel Recirculation System Piping at a BWR Plant*, U.S. Nuclear Regulatory Commission, September 21, 1982.

NRC Information Notice 84-41, *IGSCC in BWR Plants*, U.S. Nuclear Regulatory Commission, June 1, 1984.

NUREG-0313, Rev. 2, *Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping*, W. S. Hazelton and W. H. Koo, U.S. Nuclear Regulatory Commission, 1988.

XI.M8 BWR PENETRATIONS

Program Description

The program includes (a) inspection and flaw evaluation in conformance with the guidelines of staff-approved boiling water reactor vessel and internals project (BWRVIP)-49 and BWRVIP-27 documents and (b) monitoring and control of reactor coolant water chemistry in accordance with the guidelines of BWRVIP-29 (Electric Power Research Institute [EPRI] TR-103515) to ensure the long-term integrity and safe operation of boiling water reactor (BWR) vessel internal components. The BWRVIP-49 provides guidelines for instrument penetrations, and BWRVIP-27 addresses the standby liquid control (SLC) system nozzle or housing.

Evaluation and Technical Basis

1. ***Scope of Program:*** The program is focused on managing the effects of crack initiation and growth due to stress corrosion cracking (SCC) or intergranular stress corrosion cracking (IGSCC). The program contains preventive measures to mitigate SCC or IGSCC, inservice inspection (ISI) to monitor the effects of cracking on the intended function of the components, and repair and/or replacement as needed to maintain the ability to perform the intended function.

   The inspection and evaluation guidelines of BWRVIP-49 and BWRVIP-27 contain generic guidelines intended to present appropriate inspection recommendations to assure safety function integrity. The guidelines of BWRVIP-49 provide information on the type of instrument penetration, evaluate their susceptibility and consequences of failure, and define the inspection strategy to assure safe operation. The guidelines of BWRVIP-27 are applicable to plants in which the SLC system injects sodium pentaborate into the bottom head region of the vessel (in most plants, as a pipe within a pipe of the core plate ΔP monitoring system). The BWRVIP-27 guidelines address the region where the ΔP and SLC nozzle or housing penetrates the vessel bottom head and include the safe ends welded to the nozzle or housing. Guidelines for repair design criteria are provided in BWRVIP-57 for instrumentation penetrations and BWRVIP-53 for SLC line.

2. ***Preventive Actions:*** Maintaining high water purity reduces susceptibility to SCC or IGSCC. Reactor coolant water chemistry is monitored and maintained in accordance with the guidelines in BWRVIP-29 (EPRI TR-103515). The program description and the evaluation and technical basis of monitoring and maintaining reactor water chemistry are presented in Chapter XI.M2, “Water Chemistry”.

3. ***Parameters Monitored/Inspected:*** The program monitors the effects of SCC/IGSCC on the intended function of the component by detection and sizing of cracks by ISI in accordance with the guidelines of approved BWRVIP-49 or BWRVIP-27 and the requirements of the American Society of Mechanical Engineers (ASME) Code, Section XI, Table IWB 2500-1 (1995 edition through the 1996 addenda). An applicant may use the guidelines of BWRVIP-62 for inspection relief for vessel internal components with hydrogen water chemistry.

4. ***Detection of Aging Effects:*** The evaluation guidelines of BWRVIP-49 and BWRVIP-27 recommend that the inspection requirements currently in ASME Section XI continue to be followed. The extent and schedule of the inspection and test techniques prescribed by the ASME Section XI program are designed to maintain structural integrity and ensure that

aging effects will be discovered and repaired before the loss of intended function of the component. Inspection can reveal crack initiation and growth and leakage of coolant. The nondestructive examination (NDE) techniques appropriate for inspection of BWR vessel internals and their implementation needs, including the uncertainties inherent in delivering and executing NDE techniques in a BWR, are included in BWRVIP-03.

Instrument penetrations and SLC system nozzles or housings are inspected in accordance with the requirements of ASME Section XI, Subsection IWB. Components are examined and tested as specified in Table IWB-2500-1, examination categories B-E for pressure-retaining partial penetration welds in vessel penetrations, B-D for full penetration nozzle-to-vessel welds, B-F for pressure-retaining dissimilar metal nozzle-to-safe end welds, or B-J for similar metal nozzle-to-safe end welds. In addition, these components are part of examination category B-P for pressure-retaining boundary. Further details for examination are described in Chapter XI.M1, “ASME Section XI, Inservice Inspection, Subsections IWB, IWC, and IWD,” of this report.

5. ***Monitoring and Trending:*** Inspections scheduled in accordance with IWB-2400 and approved BWRVIP-48 or BWRVIP-27 provide timely detection of cracks. The scope of examination expansion and reinspection beyond the baseline inspection are required if flaws are detected.

6. ***Acceptance Criteria:*** Any indication detected is evaluated in accordance with ASME Section XI or other acceptable flaw evaluation criteria, such as the staff-approved BWRVIP-49 or BWRVIP-27 guidelines. Applicable and approved BWRVIP-14, BWRVIP-59, and BWRVIP-60 documents provide guidelines for evaluation of crack growth in stainless steels (SSs), nickel alloys, and low-alloy steels, respectively.

7. ***Corrective Actions:*** Repair and replacement procedures in the staff-approved BWRVIP-57 and BWRVIP-53 are equivalent to those requirements in the ASME Section XI. Guidelines for repair design criteria are provided in BWRVIP-57 for instrumentation penetrations and BWRVIP-53 for standby liquid control line. As discussed in the appendix to this report, the staff finds that licensee implementation of the guidelines in BWRVIP-48, as modified, will provide an acceptable level of quality for inspection and flaw evaluation of the safety-related components addressed in accordance with 10 CFR Part 50, Appendix B.

8. ***Confirmation Process:*** Site quality assurance (QA) procedures, review and approval processes, and administrative controls are implemented in accordance with the requirements of 10 CFR Part 50, Appendix B. As discussed in the appendix to this report, the staff finds that licensee implementation of the guidelines in BWRVIP-48, as modified, will provide an acceptable level of quality for inspection and flaw evaluation of the safety-related components addressed in accordance with the 10 CFR Part 50, Appendix B, confirmation process and administrative controls.

9. ***Administrative Controls:*** See Item 8, above.

10. ***Operating Experience:*** Cracking due to SCC or IGSCC has occurred in BWR components made of austenitic stainless steels and nickel alloys. The program guidelines are based on evaluation of available information, including BWR inspection data and information about the elements that cause IGSCC, to determine which locations may be susceptible to cracking. Implementation of the program provides reasonable assurance that crack initiation and growth will be adequately managed so the intended functions of the instrument

penetrations and SLC system nozzles or housings will be maintained consistent with the current licensing basis (CLB) for the period of extended operation.

References

10 CFR 50.55a, *Codes and Standards*, Office of the Federal Register, National Archives and Records Administration, 2000.

ASME Section XI, *Rules for Inservice Inspection of Nuclear Power Plant Components*, ASME Boiler and Pressure Vessel Code, 1995 edition through the 1996 addenda, American Society of Mechanical Engineers, New York, NY.

BWRVIP-03, *BWR Vessel and Internals Project, Reactor Pressure Vessel and Internals Examination Guidelines, (EPRI TR-105696 R1, March 30, 1999),* Final Safety Evaluation Report by the Office of Nuclear Reactor Regulation for BWRVIP-03, July 15, 1999.

BWRVIP-14, *Evaluation of Crack Growth in BWR Stainless Steel RPV Internals, (EPRI TR-105873, July 11, 2000),* Final Safety Evaluation Report by the Office of Nuclear Reactor Regulation for BWRVIP-14, December 3, 1999.

BWRVIP-27, *BWR Vessel and Internals Project, BWR Standby Liquid Control System/Core Plate P Inspection and Flaw Evaluation Guidelines, (EPRI TR-107286, April 1997),* Final License Renewal Safety Evaluation Report by the Office of Nuclear Reactor Regulation for BWRVIP-27 for Compliance with the License Renewal Rule (10 CFR Part 54), December 20, 1999.

BWRVIP-29 (EPRI TR-103515), *BWR Vessel and Internals Project, BWR Water Chemistry Guidelines–1993 Revision, Normal and Hydrogen Water Chemistry*, Electric Power Research Institute, Palo Alto, CA, February 1994.

BWRVIP-48, *BWR Vessel and Internals Project, Vessel ID Attachment Weld Inspection and Flaw Evaluation Guidelines, (EPRI TR-108724, February 1998),* Final License Renewal Safety Evaluation Report by the Office of Nuclear Reactor Regulation for BWRVIP-48 for Compliance with the License Renewal Rule (10 CFR Part 54), January 17, 2001.

BWRVIP-49, *BWR Vessel and Internals Project, Instrument Penetration Inspection and Flaw Evaluation Guidelines, (EPRI TR-108695, March 1998),* Final License Renewal Safety Evaluation Report by the Office of Nuclear Reactor Regulation for BWRVIP-49 for Compliance with the License Renewal Rule (10 CFR Part 54), September 1, 1999.

BWRVIP-53, *BWR Vessel and Internals Project, Standby Liquid Control Line Repair Design Criteria, (EPRI TR-108716, March 24, 2000),* Initial Safety Evaluation Report by the Office of Nuclear Reactor Regulation for BWRVIP-53, October 26, 2000.

BWRVIP-57, *BWR Vessel and Internals Project, Instrument Penetration Repair Design Criteria, (EPRI TR-108721),* BWRVIP and Electric Power Research Institute, Palo Alto, CA, March 24, 2000.

BWRVIP-59, *Evaluation of Crack Growth in BWR Nickel-Base Austenitic Alloys in RPV Internals, (EPRI TR-108710),* BWRVIP and Electric Power Research Institute, Palo Alto, CA, March 24, 2000.

BWRVIP-60, *BWR Vessel and Internals Project, Evaluation of Crack Growth in BWR Low Alloy Steel RPV Internals, (EPRI TR-108709, April 14, 2000),* Final Safety Evaluation Report by the Office of Nuclear Reactor Regulation for BWRVIP-60, July 8, 1999.

BWRVIP-62, *BWR Vessel and Internals Project, Technical Basis for Inspection Relief for BWR Internal Components with Hydrogen Injection, (EPRI TR-108705),* BWRVIP and Electric Power Research Institute, Palo Alto, CA, March 7, 2000.

XI.M9 BWR VESSEL INTERNALS

Program Description

The program includes (a) inspection and flaw evaluation in conformance with the guidelines of applicable and staff-approved boiling water reactor vessel and internals project (BWRVIP) documents and (b) monitoring and control of reactor coolant water chemistry in accordance with the guidelines of BWRVIP-29 (Electric Power Research Institute [EPRI] TR-103515) to ensure the long-term integrity and safe operation of boiling water reactor (BWR) vessel internal components.

Evaluation and Technical Basis

1. ***Scope of Program:*** The program is focused on managing the effects of crack initiation and growth due to stress corrosion cracking (SCC), intergranular stress corrosion cracking (IGSCC), or irradiation assisted stress corrosion cracking (IASCC). The program contains preventive measures to mitigate SCC, IGSCC, or IASCC; inservice inspection (ISI) to monitor the effects of cracking on the intended function of the components; and repair and/or replacement as needed to maintain the ability to perform the intended function.

   The BWRVIP documents provide generic guidelines intended to present the applicable inspection recommendations to assure safety function integrity of the subject safety-related reactor pressure vessel internal components. The guidelines include information on component description and function; evaluate susceptible locations and safety consequences of failure; provide recommendations for methods, extent, and frequency of inspection; discuss acceptable methods for evaluating the structural integrity significance of flaws detected during these examinations; and recommend repair and replacement procedures.

   The various applicable BWRVIP guidelines are as follows:

   Core shroud: BWRVIPs-07, -63, and -76 provide guidelines for inspection and evaluation; BWRVIP-02, Rev. 2, provides guidelines for repair design criteria.

   Core plate: BWRVIP-25 provides guidelines for inspection and evaluation; BWRVIP-50 provides guidelines for repair design criteria.

   Shroud support: BWRVIP-38, provides guidelines for inspection and evaluation; BWRVIP-52 provides guidelines for repair design criteria.

   Low-pressure coolant injection (LPCI) coupling: BWRVIP-42 provides guidelines for inspection and evaluation; BWRVIP-56 provides guidelines for repair design criteria.

   Top guide: BWRVIP-26 provides guidelines for inspection and evaluation; BWRVIP-50 provides guidelines for repair design criteria.

   Core spray: BWRVIP-18 provides guidelines for inspection and evaluation; BWRVIP-16 and 19 provides guidelines for replacement and repair design criteria, respectively.

   Jet pump assembly: BWRVIP-41 provides guidelines for inspection and evaluation; BWRVIP-51 provides guidelines for repair design criteria.

Control rod drive (CRD) housing: BWRVIP-47 provides guidelines for inspection and evaluation; BWRVIP-58 provides guidelines for repair design criteria.

Lower plenum: BWRVIP-47 provides guidelines for inspection and evaluation; BWRVIP-57 provides guidelines for repair design criteria for instrument penetrations.

In addition, BWRVIP-44 provides guidelines for weld repair of nickel alloys; BWRVIP-45 provides guidelines for weldability of irradiated structural components.

2. ***Preventive Actions:*** Maintaining high water purity reduces susceptibility to cracking due to SCC or IGSCC. Reactor coolant water chemistry is monitored and maintained in accordance with the guidelines in BWRVIP-29 (EPRI TR-103515). The program description and evaluation, and technical basis of monitoring and maintaining reactor water chemistry are presented in Chapter XI.M2, "Water Chemistry."

3. ***Parameters Monitored/Inspected:*** The program monitors the effects of cracking on the intended function of the component by detection and sizing of cracks by inspection in accordance with the guidelines of applicable and approved BWRVIP documents and the requirements of the American Society of Mechanical Engineers (ASME) Code, Section XI, Table IWB 2500-1 (1995 edition through the 1996 addenda). An applicant may use the guidelines of BWRVIP-62 for inspection relief for vessel internal components with hydrogen water chemistry.

4. ***Detection of Aging Effects:*** The extent and schedule of the inspection and test techniques prescribed by the applicable and approved BWRVIP guidelines are designed to maintain structural integrity and ensure that aging effects will be discovered and repaired before the loss of intended function. Inspection can reveal crack initiation and growth. Vessel internal components are inspected in accordance with the requirements of ASME Section XI, Subsection IWB, examination category B-N-2. The ASME Section XI inspection specifies visual VT-1 examination to detect discontinuities and imperfections, such as cracks, corrosion, wear, or erosion, on the surfaces of components. This inspection also specifies visual VT-3 examination to determine the general mechanical and structural condition of the component supports by (a) verifying parameters, such as clearances, settings, and physical displacements, and (b) detecting discontinuities and imperfections, such as loss of integrity at bolted or welded connections, loose or missing parts, debris, corrosion, wear, or erosion.

   The applicable and approved BWRVIP guidelines recommend more stringent inspections, such as enhanced visual VT-1 examinations or ultrasonic methods of volumetric inspection, for certain selected components and locations. The nondestructive examination (NDE) techniques appropriate for inspection of BWR vessel internals and their implementation needs, including the uncertainties inherent in delivering and executing NDE techniques in a BWR, are included in BWRVIP-03.

5. ***Monitoring and Trending:*** Inspections scheduled in accordance with the applicable and approved BWRVIP guidelines provide timely detection of cracks. The scope of examination expansion and reinspection beyond the baseline inspection are required if flaws are detected.

6. ***Acceptance Criteria:*** Any indication detected is evaluated in accordance with ASME Section XI or the applicable staff-approved BWRVIP guidelines. Approved BWRVIP-14, BWRVIP-59, and BWRVIP-60 documents provide guidelines for evaluation of crack growth in stainless steels (SSs), nickel alloys, and low-alloy steels, respectively.

7. ***Corrective Actions:*** Repair and replacement procedures are equivalent to those requirements in ASME Section XI. Repair and replacement is in conformance with the applicable and approved BWRVIP guidelines listed above. As discussed in the appendix to this report, the staff finds that licensee implementation of the guidelines in the staff-approved BWRVIP reports will provide an acceptable level of quality for inspection and flaw evaluation of the safety-related components addressed in accordance with 10 CFR Part 50, Appendix B.

8. ***Confirmation Process:*** Site quality assurance (QA) procedures, review and approval processes, and administrative controls are implemented in accordance with the requirements of 10 CFR Part 50, Appendix B. As discussed in the appendix to this report, the staff finds that licensee implementation of the guidelines in the staff-approved BWRVIP reports will provide an acceptable level of quality for inspection and flaw evaluation of the safety-related components addressed in accordance with the 10 CFR Part 50, Appendix B, confirmation process and administrative controls.

9. ***Administrative Controls:*** See Item 8, above.

10. ***Operating Experience:*** Extensive cracking has been observed in core shrouds at both horizontal (Nuclear Regulatory Commission [NRC] Generic Letter [GL] 94-03) and vertical (NRC Information Notice [IN] 97-17) welds. It has affected shrouds fabricated from Type 304 and Type 304L SS, which is generally considered to be more resistant to SCC. Weld regions are most susceptible, although it is not clear whether this is due to sensitization and/or impurities associated with the welds or the high residual stresses in the weld regions. This experience is reviewed in NRC GL 94-03 and NUREG-1544; some experiences with visual inspections are discussed in NRC IN 94-42.

    Both circumferential (NRC IN 88-03) and radial cracking (NRC IN 92-57) has been observed in the shroud support access hole cover made from Alloy 600. Instances of cracking in core spray spargers have been reviewed in NRC IE Bulletin 80-13.

    Cracking of the core plate has not been reported, but the creviced regions beneath the plate are difficult to inspect. The NRC IN 95-17 discusses cracking in top guides of United States and overseas BWRs. Related experience in other components is reviewed in NRC GL 94-03 and NUREG-1544. Cracking has also been observed in the top guide of a Swedish BWR.

    Instances of cracking have occurred in the jet pump assembly (NRC IE Bulletin 80-07), hold-down beam (NRC IN 93-101), and jet pump riser pipe elbows (NRC IN 97-02).

    Cracking of dry tubes has been observed at 14 or more BWRs. The cracking is intergranular and has been observed in dry tubes without apparent sensitization, suggesting that IASCC may also play a role in the cracking.

    The program guidelines outlined in applicable and approved BWRVIP documents are based on evaluation of available information, including BWR inspection data and information on the elements that cause SCC, IGSCC, or IASCC, to determine which components may be susceptible to cracking. Implementation of the program provides reasonable assurance that crack initiation and growth will be adequately managed so the intended functions of the vessel internal components will be maintained consistent with the current licensing basis (CLB) for the period of extended operation.

References

10 CFR 50.55a, *Codes and Standards*, Office of the Federal Register, National Archives and Records Administration, 2000.

ASME Section XI, *Rules for Inservice Inspection of Nuclear Power Plant Components*, ASME Boiler and Pressure Vessel Code, 1995 edition through the 1996 addenda, American Society of Mechanical Engineers, New York, NY.

BWRVIP-02, *BWR Vessel and Internals Project, BWR Core Shroud Repair Design Criteria, Rev. 2,* BWRVIP and Electric Power Research Institute, Palo Alto, CA, March 7, 2000.

BWRVIP-03, *BWR Vessel and Internals Project, Reactor Pressure Vessel and Internals Examination Guidelines, (EPRI TR-105696 R1, March 30, 1999),* Final Safety Evaluation Report by the Office of Nuclear Reactor Regulation for BWRVIP-03, July 15, 1999.

BWRVIP-07, *BWR Vessel and Internals Project, Guidelines for Reinspection of BWR Core Shrouds, (EPRI TR-105747, Feb. 29, 1996),* Final Safety Evaluation Report by the Office of Nuclear Reactor Regulation for BWRVIP-07, April 27, 1998.

BWRVIP-14, *Evaluation of Crack Growth in BWR Stainless Steel RPV Internals, (EPRI TR-105873, July 11, 2000),* Final Safety Evaluation Report by the Office of Nuclear Reactor Regulation for BWRVIP-14, December 3, 1999.

BWRVIP-16, *Internal Core Spray Piping and Sparger Replacement Design Criteria, (EPRI TR-106708),* BWRVIP and Electric Power Research Institute, Palo Alto, CA, March 7, 2000.

BWRVIP-18, *BWR Vessel and Internals Project, BWR Core Spray Internals Inspection and Flaw Evaluation Guidelines, (EPRI TR-106740, July 1996),* Final License Renewal Safety Evaluation Report by the Office of Nuclear Reactor Regulation for BWRVIP-18 for Compliance with the License Renewal Rule (10 CFR Part 54), December 7, 2000.

BWRVIP-19, *Internal Core Spray Piping and Sparger Repair Design Criteria, (EPRI TR 106893),* BWRVIP and Electric Power Research Institute, Palo Alto, CA, March 7, 2000.

BWRVIP-25, *BWR Vessel and Internals Project, BWR Core Plate Inspection and Flaw Evaluation Guidelines, (EPRI TR-107284, Dec. 1996)*, Final License Renewal Safety Evaluation Report by the Office of Nuclear Reactor Regulation for BWRVIP-25 for Compliance with the License Renewal Rule (10 CFR Part 54), December 7, 2000.

BWRVIP-26, *BWR Vessel and Internals Project, Top Guide Inspection and Flaw Evaluation Guidelines, (EPRI TR-107285, Dec. 1996)*, Final License Renewal Safety Evaluation Report by the Office of Nuclear Reactor Regulation for BWRVIP-26 for Compliance with the License Renewal Rule (10 CFR Part 54), December 7, 2000.

BWRVIP-29, *BWR Vessel and Internals Project, BWR Water Chemistry Guidelines–1993 Revision, Normal and Hydrogen Water Chemistry*, *(EPRI TR-103515)*, Electric Power Research Institute, Palo Alto, CA, February 1994.

BWRVIP-38, *BWR Vessel and Internals Project, BWR Shroud Support Inspection and Flaw Evaluation Guidelines, (EPRI TR-108823, September 1997),* Final License Renewal Safety Evaluation Report by the Office of Nuclear Reactor Regulation for BWRVIP-38 for Compliance with the License Renewal Rule (10 CFR Part 54), March 1, 2001.

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BWRVIP-44, *Underwater Weld Repair of Nickel Alloy Reactor Vessel Internals, (EPRI TR-108708, April 3, 2000),* Final Safety Evaluation Report by the Office of Nuclear Reactor Regulation for BWRVIP-44, June 9, 1999.

BWRVIP-45, *Weldability of Irradiated LWR Structural Components, (EPRI TR-108707),* BWRVIP and Electric Power Research Institute, Palo Alto, CA, June 14, 2000.

BWRVIP-47, *BWR Vessel and Internals Project, BWR Lower Plenum Inspection and Flaw Evaluation Guidelines, (EPRI TR-108727, December 1997),* Final License Renewal Safety Evaluation Report by the Office of Nuclear Reactor Regulation for BWRVIP-47 for Compliance with the License Renewal Rule (10 CFR Part 54), December 7, 2000.

BWRVIP-50, *Top Guide/Core Plate Repair Design Criteria, (EPRI TR-108722),* BWRVIP and Electric Power Research Institute, Palo Alto, CA, April 3, 2000.

BWRVIP-51, *Jet Pump Repair Design Criteria, (EPRI TR-108718, March 7, 2000),* Initial Safety Evaluation Report by the Office of Nuclear Reactor Regulation for BWRVIP-51, October 28, 2000.

BWRVIP-52, *Shroud Support and Vessel Bracket Repair Design Criteria, (EPRI TR-108720, June 26, 1998),* Initial Safety Evaluation Report by the Office of Nuclear Reactor Regulation for BWRVIP-52, November 2, 2000.

BWRVIP-56, *LPCI Coupling Repair Design Criteria, (EPRI TR-108717),* BWRVIP and Electric Power Research Institute, Palo Alto, CA, March 24, 2000.

BWRVIP-57, *Instrument Penetration Repair Design Criteria, (EPRI TR-108721),* BWRVIP and Electric Power Research Institute, Palo Alto, CA, March 24, 2000.

BWRVIP-58, *CRD Internal Access Weld Repair, (EPRI TR-108703),* BWRVIP and Electric Power Research Institute, Palo Alto, CA, March 7, 2000.

BWRVIP-59, *Evaluation of Crack Growth in BWR Nickel-Base Austenitic Alloys in RPV Internals, (EPRI TR-108710),* BWRVIP and Electric Power Research Institute, Palo Alto, CA, March 24, 2000.

BWRVIP-60, *BWR Vessel and Internals Project, Evaluation of Crack Growth in BWR Low Alloy Steel RPV Internals, (EPRI TR-108709, April 14, 2000),* Final Safety Evaluation Report by the Office of Nuclear Reactor Regulation for BWRVIP-60, July 8, 1999.

BWRVIP-62, *BWR Vessel and Internals Project, Technical Basis for Inspection Relief for BWR Internal Components with Hydrogen Injection, (EPRI TR-108705),* BWRVIP and Electric Power Research Institute, Palo Alto, CA, March 7, 2000.

BWRVIP-63, *BWR Vessel and Internals Project, Shroud Vertical Weld Inspection and Evaluation Guidelines, (EPRI TR-113117, Feb. 29, 2000),* Initial Safety Evaluation Report by the Office of Nuclear Reactor Regulation for BWRVIP-63, April 18, 2000.

BWRVIP-76, *BWR Vessel and Internals Project, BWR Core Shroud Inspection and Flaw Evaluation Guidelines, (EPRI TR-114232, November 1999).*

NRC Generic Letter 94-03, *Intergranular Stress Corrosion Cracking of Core Shrouds in Boiling Water Reactors*, U.S. Nuclear Regulatory Commission, July 25, 1994.

NRC IE Bulletin No. 80-07, *BWR Jet Pump Assembly Failure*, U.S. Nuclear Regulatory Commission, April 4, 1980.

NRC IE Bulletin No. 80-13, *Cracking in Core Spray Spargers*, U.S. Nuclear Regulatory Commission, May 12, 1980.

NRC IE Bulletin No. 80-07, Supplement 1, *BWR Jet Pump Assembly Failure,* U.S. Nuclear Regulatory Commission, May 13, 1980.

NRC Information Notice 88-03, *Cracks in Shroud Support Access Hole Cover Welds*, U.S. Nuclear Regulatory Commission, February 2, 1988.

NRC Information Notice 92-57, *Radial Cracking of Shroud Support Access Hole Cover Welds*, U.S. Nuclear Regulatory Commission, August 11, 1992.

NRC Information Notice 93-101, *Jet Pump Hold-Down Beam Failure*, U.S. Nuclear Regulatory Commission, December 17, 1993.

NRC Information Notice 94-42, *Cracking in the Lower Region of the Core Shroud in Boiling Water Reactors*, U.S. Nuclear Regulatory Commission, June 7, 1994.

NRC Information Notice 95-17, *Reactor Vessel Top Guide and Core Plate Cracking*, U.S. Nuclear Regulatory Commission, March 10, 1995.

NRC Information Notice 97-02, *Cracks Found in Jet Pump Riser Assembly Elbows at Boiling Water Reactors*, U.S. Nuclear Regulatory Commission, February 6, 1997.

NRC Information Notice 97-17, *Cracking of Vertical Welds in the Core Shroud and Degraded Repair*, U.S. Nuclear Regulatory Commission, April 4, 1997.

NUREG-1544, *Status Report: Intergranular Stress Corrosion Cracking of BWR Core Shrouds and Other Internal Components*, U.S. Nuclear Regulatory Commission, March 1996.

XI.M10 BORIC ACID CORROSION

Program Description

The program relies on implementation of recommendations of Nuclear Regulatory Commission (NRC) Generic Letter (GL) 88-05 to monitor the condition of the reactor coolant pressure boundary for borated water leakage. Periodic visual inspection of adjacent structures, components, and supports for evidence of leakage and corrosion is an element of the NRC GL 88-05 monitoring program.

Evaluation and Technical Basis

1. ***Scope of Program:*** The program covers any carbon steel and low-alloy steel structures or components, and electrical components, on which borated reactor water may leak. The program includes recommendations of NRC GL 88-05. The staff guidance of NRC GL 88-05 provides a program consisting of systematic measures to ensure that corrosion caused by leaking borated coolant does not lead to degradation of the leakage source or adjacent structures and components, and provides assurance that the reactor coolant pressure boundary will have an extremely low probability of abnormal leakage, rapidly propagating failure, or gross rupture. Such a program provides for (a) determination of the principal location of leakage, (b) examination requirements and procedures for locating small leaks, and (c) engineering evaluations and corrective actions to ensure that boric acid corrosion does not lead to degradation of the leakage source or adjacent structures or components, which could cause the loss of intended function of the structures or components.

2. ***Preventive Actions:*** Minimizing reactor coolant leakage by frequent monitoring of the locations where potential leakage could occur, and timely repair if leakage is detected, prevents or mitigates boric acid corrosion. Preventive measures also include modifications in the design or operating procedures to reduce the probability of leaks at locations where they may cause corrosion damage and use of suitable corrosion resistant materials or the application of protective coatings.

3. ***Parameters Monitored/Inspected:*** The aging management program (AMP) monitors the effects of boric acid corrosion on the intended function of an affected structure and component by detection of coolant leakage. Coolant leakage results in deposits of white boric acid crystals and presence of moisture that can be observed by the naked eye.

4. ***Detection of Aging Effects:*** Degradation of the component due to boric acid corrosion cannot occur without leakage of coolant that contains boric acid. Conditions leading to boric acid corrosion, such as crystal buildup and evidence of moisture, are readily detectable by visual inspection. The program delineated in NRC GL 88-05 includes guidelines for locating small leaks, conducting examinations, and performing engineering evaluations. Thus the use of the NRC GL 88-05 program will assure detection of leakage before the loss of the intended function of the component.

5. ***Monitoring and Trending:*** The program delineated in NRC GL 88-05 provides for timely detection of leakage by observance of boric acid crystals during normal plant walkdowns and maintenance.

6. ***Acceptance Criteria*:** Any detected leakage or crystal buildup requires corrective actions prior to continued service.

7. ***Corrective Actions*****:** The leakage source and areas of general corrosion are located and corrective actions are implemented in conformance with the program proposed by NRC GL 88-05. The NRC GL 88-05 requires that corrective actions to prevent recurrences of degradation caused by boric acid leakage be included in the program implementation. These corrective actions include any modifications to be introduced in the present design or operating procedures of the plant that (a) reduce the probability of primary coolant leaks at locations where they may cause corrosion damage, and (b) entail the use of suitable corrosion resistant materials or the application of protective coatings or claddings. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable in addressing corrective actions.

8. ***Confirmation Process:*** Site quality assurance (QA) procedures, review and approval processes, and administrative controls are implemented in accordance with the requirements of 10 CFR Part 50, Appendix B. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable in addressing confirmation process and administrative controls.

9. ***Administrative Controls:*** See Item 8, above.

10. ***Operating Experience:*** Boric acid corrosion observed in nuclear power plants (NRC Information Notice [IN] 86-108 S3) may be classified into two types: (a) corrosion that increases the rate of leakage (e.g., corrosion of closure bolting or fasteners) and (b) corrosion that occurs some distance from the source of leakage. The guidance of NRC GL 88-05 is effective in managing the effects of boric acid corrosion on the intended function of reactor components.

References

10 CFR 50.55a, *Codes and Standards*, Office of the Federal Register, National Archives and Records Administration, 2000.

NRC Generic Letter 88-05, *Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants*, U.S. Nuclear Regulatory Commission, March 17, 1988.

NRC Information Notice 86-108 S3, *Degradation of Reactor Coolant System Pressure Boundary Resulting from Boric Acid Corrosion*, U.S. Nuclear Regulatory Commission, December 26, 1986; Supplement 1, April 20, 1987; Supplement 2, November 19, 1987; and Supplement 3, January 5, 1995.

XI.M11 NICKEL-ALLOY NOZZLES AND PENETRATIONS

Program Description

The program includes (a) primary water stress corrosion cracking (PWSCC) susceptibility assessment to identify susceptible components, (b) monitoring and control of reactor coolant water chemistry to mitigate PWSCC, and (c) inservice inspection (ISI) of reactor vessel head penetrations in accordance with the American Society of Mechanical Engineers (ASME) Code, Section XI, Subsection IWB, Table IWB 2500-1 (1995 edition through the 1996 addenda) to monitor PWSCC and its effect on the intended function of the component. For susceptible penetrations and locations, the program includes an industry-wide, integrated, long-term inspection program based on the industry responses to NRC Generic Letter (GL) 97-01 contained in the NEI letter dated December 11, 1998, from Dave Modeen to Gus Lainas, “Responses to NRC Requests for Additional Information (RAIs) on GL 97-01” and individual plant responses. Primary water chemistry is monitored and maintained in accordance with the Electric Power Research Institute (EPRI) guidelines in TR-105714 (Rev. 3, or later revisions or update) to minimize the potential of crack initiation and growth.

Evaluation and Technical Basis

1. ***Scope of Program:*** The program is focused on managing the effects of crack initiation and growth due to primary water stress corrosion cracking (PWSCC) of nickel-base alloys. The program includes ISI in accordance with ASME Subsection IWB, Table IWB 2500-1. For susceptible components and locations, the program includes an industry wide, integrated, long-term inspection program based on the industry responses to NRC GL 97-01 contained in the NEI letter dated December 11, 1998, from Dave Modeen to Gus Lainas, “Responses to NRC Requests for Additional Information (RAIs) on GL 97-01” and individual plant responses. Preventive measures are in accordance with EPRI guidelines in TR-105714 to mitigate PWSCC. An integrated cracking susceptibility assessment in accordance with industry susceptibility models and inspection results was performed in response to NRC GL 97-01, to define the most susceptible plants and rank them in accordance with their susceptibility. The information is used by each plant to determine the proper timing of vessel head penetration examinations, either during the current license period or the period of license renewal, if necessary. The components and locations to be included in the long-term inspection program are those that currently have been identified as susceptible to PWSCC, and those that will become susceptible during the period of license renewal. Significant changes in the industry models, as future plants perform inspection, may require reassessment.

2. ***Preventive Actions:*** Preventive measures to mitigate PWSCC are in accordance with EPRI guidelines in TR-105714. The program description and the evaluation and technical basis of monitoring and maintaining reactor water chemistry are presented in Chapter XI.M2, “Water Chemistry.”

3. ***Parameters Monitored/Inspected:*** The program monitors the effects of PWSCC on the intended function of the control rod drive (CRD) and other Alloy 600 head penetrations by detection and sizing of cracks and coolant leakage by ISI. In C-E-designed pressurized water reactors (PWRs), the CRD head penetration is called the control element drive (CED) head penetration.

4. ***Detection of Aging Effects:*** A review of the scope and schedule of the inspections, including the leakage detection system, based on NRC GL 97-01, assures detection of cracks before the loss of intended function of the components.

   The PWSCC susceptibility assessment was performed in response to NRC GL 97-01 utilizing the most current industry susceptibility models that were based on material and operating parameters and inspection results to date, to rank plants in accordance with their susceptibility. This information is used to develop a plant-specific long-term inspection program, including schedule, scope and determination whether an augmented inspection program of nozzle penetration, including a combination of surface and volumetric examination, is necessary. Because the leakage through cracks in nozzles can be small, this aging management program (AMP), in accordance with NRC GL 97-01, recommends implementation of an enhanced leakage detection method for detecting small leaks during plant operation.

5. ***Monitoring and Trending:*** An inspection schedule, in accordance with the integrated inspection program based on the NRC GL 97-01 susceptibility assessment, provides timely detection of cracks. Inspection results are used to update the susceptibility models. The frequency of subsequent inspections is based on the finding of the initial inspections and flaw evaluations performed with staff-approved crack growth rate models for nickel-alloys.

6. ***Acceptance Criteria:*** Any indication detected is evaluated in accordance with ASME Section XI or other acceptable flaw evaluation criteria. To verify the adequacy of the long-term inspection program and acceptance criteria, if there have been significant changes since the applicants response to NRC GL 97-01, the applicant either provides references to appropriate industry model revisions or provides updated information on crack initiation and crack growth data and models.

7. ***Corrective Actions:*** Repair and replacement procedures are equivalent to those requirements in ASME Section XI. Repair is in conformance with IWB-4000 and replacement is in accordance with IWB-7000. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable in addressing corrective actions.

8. ***Confirmation Process:*** Site quality assurance (QA) procedures, review and approval processes, and administrative controls are implemented in accordance with the requirements of 10 CFR Part 50, Appendix B. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable in addressing the confirmation process and administrative controls.

9. ***Administrative Controls:*** See Item 8, above.

10. ***Operating Experience:*** Cracking of Alloy 600 has occurred in domestic and foreign PWRs (NRC Information Notice [IN] 90-10). Furthermore, ingress of demineralizer resins has also occurred in operating plants (NRC IN 96-11), the program relies upon monitoring and control of primary water chemistry to manage the effects of such excursions. An integrated susceptibility assessment and inspection program, based on the guidelines in NRC GL 97-01, is effective in managing the effect of PWSCC on the intended function of reactor vessel head penetrations.

References

10 CFR 50.55a, *Codes and Standards*, Office of the Federal Register, National Archives and Records Administration, 2000.

ASME Section XI, *Rules for Inservice Inspection of Nuclear Power Plant Components*, ASME Boiler and Pressure Vessel Code, 1995 edition through the 1996 addenda, American Society of Mechanical Engineers, New York, NY.

EPRI TR-105714, *PWR Primary Water Chemistry Guidelines–Revision 3*, Electric Power Research Institute, Palo Alto, CA, November 1995.

Letter from David J. Modeen of Nuclear Energy Institute to Gus C. Lainais of Division of Engineering, *Responses to NRC Requests for Additional Information on Generic Letter 97-01*, December 11, 1998.

NRC Generic Letter 97-01, *Degradation of Control Rod Drive Mechanism Nozzle and Other Vessel Closure Head Penetrations*, U.S. Nuclear Regulatory Commission, April 1, 1997.

NRC Information Notice 90-10, *Primary Water Stress Corrosion Cracking (PWSCC) of Alloy 600*, U.S. Nuclear Regulatory Commission, February 23, 1990.

NRC Information Notice 96-11, *Ingress of Demineralizer Resins Increase Potential for Stress Corrosion Cracking of Control Rod Drive Mechanism Penetrations*, U.S. Nuclear Regulatory Commission, February 14, 1996.

XI.M12 THERMAL AGING EMBRITTLEMENT OF CAST AUSTENITIC STAINLESS STEEL (CASS)

Program Description

The reactor coolant system components are inspected in accordance with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI. This inspection is augmented to detect the effects of loss of fracture toughness due to thermal aging embrittlement of cast austenitic stainless steel (CASS) components. This aging management program (AMP) includes (a) determination of the susceptibility of CASS components to thermal aging embrittlement based on casting method, molybdenum content, and percent ferrite, and (b) for “potentially susceptible” components, as defined below, aging management is accomplished through either enhanced volumetric examination or plant- or component-specific flaw tolerance evaluation. Additional inspection or evaluations to demonstrate that the material has adequate fracture toughness are not required for components that are not susceptible to thermal aging embrittlement.

For pump casings and valve bodies, based on the assessment documented in the letter dated May 19, 2000, from Christopher Grimes, Nuclear Regulatory Commission (NRC), to Douglas Walters, Nuclear Energy Institute (NEI), screening for susceptibility to thermal aging is not required. The existing ASME Section XI inspection requirements, including the alternative requirements of ASME Code Case N-481 for pump casings, are adequate for all pump casings and valve bodies.

Evaluation and Technical Basis

1. ***Scope of Program:*** The program includes screening criteria to determine which CASS components are potentially susceptible to thermal aging embrittlement and require augmented inspection. The screening criteria are applicable to all primary pressure boundary and reactor vessel internal components constructed from SA-351 Grades CF3, CF3A, CF8, CF8A, CF3M, CF3MA, CF8M, with service conditions above 250°C (482°F). The screening criteria for susceptibility to thermal aging embrittlement are not applicable to niobium-containing steels; such steels require evaluation on a case-by-case basis. For potentially susceptible components, aging management is accomplished either through volumetric examination or plant- or component-specific flaw tolerance evaluation.

   Based on the criteria set forth in the May 19, 2000, NRC letter, the susceptibility to thermal aging embrittlement of CASS components is determined in terms of casting method, molybdenum content, and ferrite content. For low-molybdenum content (0.5 wt.% max.) steels, only static-cast steels with >20% ferrite are potentially susceptible to thermal embrittlement. Static-cast low-molybdenum steels with ≤20% ferrite and all centrifugal-cast low-molybdenum steels are not susceptible. For high-molybdenum content (2.0 to 3.0 wt.%) steels, static-cast steels with >14% ferrite and centrifugal-cast steels with >20% ferrite are potentially susceptible to thermal embrittlement. Static-cast high-molybdenum steels with ≤14% ferrite and centrifugal-cast high-molybdenum steels with ≤20% ferrite are not susceptible. In the susceptibility screening method, ferrite content is calculated by using the Hull’s equivalent factors (described in NUREG/CR-4513, Rev. 1) or a method producing an equivalent level of accuracy (±6% deviation between measured and calculated values). A fracture toughness value of 255 kJ/m^2 (1,450 $in.\text{-}lb/in.^2$) at a crack depth of 2.5 mm (0.1 in.) is used to differentiate between CASS materials that are nonsusceptible and potentially susceptible to thermal aging embrittlement. Extensive research data indicate that for nonsusceptible CASS materials, the saturated lower-bound fracture toughness is greater than 255 kJ/m^2 (NUREG/CR-4513, Rev. 1).

For pump casings and valve bodies, screening for susceptibility to thermal aging embrittlement is not required. The staff's conservative bounding integrity analysis shows that thermally aged CASS valve bodies and pump casings are resistant to failure. For all pump casings and valve bodies greater than nominal pipe size (NPS) 4 in., the existing ASME Section XI inspection requirements, including the alternative requirements of ASME Code Case N-481 for pump casings, are adequate. The ASME Section XI, Subsection IWB, requires only surface examination of valve bodies less than NPS 4 in. For valve bodies less than NPS 4 in., the adequacy of inservice inspection (ISI) according to ASME Section XI has been demonstrated by a NRC-performed bounding integrity analysis (see letter from Christopher Grimes).

2. ***Preventive Actions:*** The program consists of evaluation and inspection and provides no guidance on methods to mitigate thermal aging embrittlement.

3. ***Parameters Monitored/Inspected:*** The AMP monitors the effects of loss of fracture toughness on the intended function of the component by identifying the CASS materials that are susceptible to thermal aging embrittlement. For potentially susceptible materials, the program consists of either enhanced volumetric examination to detect and size cracks or plant- or component-specific flaw tolerance evaluation. (Loss of fracture toughness is of consequence only if cracks exist.)

4. ***Detection of Aging Effects:*** For pump casings and valve bodies and "not susceptible" piping, no additional inspection or evaluations are required to demonstrate that the material has adequate fracture toughness. For "potentially susceptible" piping, because the base metal does not receive periodic inspection per ASME Section XI, the CASS AMP provides for volumetric examination of the base metal, with the scope of the inspection covering the portions determined to be limiting from the standpoint of applied stress, operating time, and environmental considerations. Examination methods that meet the criteria of the ASME Section XI, Appendix VIII, are acceptable. Alternatively, a plant- or component-specific flaw tolerance evaluation, using specific geometry and stress information, can be used to demonstrate that the thermally-embrittled material has adequate toughness.

5. ***Monitoring and Trending:*** Inspection schedules in accordance with IWB-2400 or IWC-2400 and reliable examination methods provide timely detection of cracks.

6. ***Acceptance Criteria:*** Flaws detected in CASS components are evaluated in accordance with the applicable procedures of IWB-3500 or IWC-3500. Flaw tolerance evaluation for components with ferrite content up to 25% is performed according to the principles associated with IWB-3640 procedures for submerged arc welds (SAW), disregarding the Code restriction of 20% ferrite in IWB-3641(b)(1). Extensive research data indicate that the lower-bound fracture toughness of thermally aged CASS materials with up to 25% ferrite is similar to that for SAWs with up to 20% ferrite (Lee et al., 1997). Flaw evaluation for piping with >25% ferrite is performed on a case-by-case basis by using fracture toughness data provided by the applicant.

7. ***Corrective Actions:*** Repair is in conformance with IWA-4000 and IWB-4000 or IWC, and replacement is in accordance with IWA-7000 and IWB-7000 or IWC-7000. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable in addressing corrective actions.

8. ***Confirmation Process:*** Site quality assurance (QA) procedures, review and approval processes, and administrative controls are implemented in accordance with the requirements of 10 CFR Part 50, Appendix B. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable in addressing the confirmation process and administrative controls.

9. ***Administrative Controls:*** See Item 8, above.

10. ***Operating Experience:*** The proposed AMP was developed by using research data obtained on both laboratory-aged and service-aged materials. Based on this information, the effects of thermal aging embrittlement on the intended function of CASS components are effectively managed.

References

ASME Section XI, *Rules for Inservice Inspection of Nuclear Power Plant Components*, ASME Boiler and Pressure Vessel Code, 1995 edition through the 1996 addenda, American Society of Mechanical Engineers, New York, NY.

Lee, S., Kuo, P. T., Wichman, K., and Chopra, O., *Flaw Evaluation of Thermally Aged Cast Stainless Steel in Light-Water Reactor Applications*, Int. J. Pres. Ves. and Piping, 72, pp. 37-44, 1997.

Letter from Christopher I. Grimes, U.S. Nuclear Regulatory Commission, License Renewal and Standardization Branch, to Douglas J. Walters, Nuclear Energy Institute, License Renewal Issue No. 98-0030, *Thermal Aging Embrittlement of Cast Stainless Steel Components*, May 19, 2000.

NUREG/CR-4513, Rev. 1, *Estimation of Fracture Toughness of Cast Stainless Steels During Thermal Aging in LWR Systems*, U.S. Nuclear Regulatory Commission, August 1994.

XI.M13 THERMAL AGING AND NEUTRON IRRADIATION EMBRITTLEMENT OF CAST AUSTENITIC STAINLESS STEEL (CASS)

Program Description

The reactor vessel internals receive a visual inspection in accordance with the American Society of Mechanical Engineers (ASME) Code Section XI, Subsection IWB, Category B-N-3. This inspection is augmented to detect the effects of loss of fracture toughness due to thermal aging, neutron irradiation embrittlement and void swelling of cast austenitic stainless steel (CASS) reactor vessel internals. This aging management program (AMP) includes (a) identification of susceptible components determined to be limiting from the standpoint of thermal aging susceptibility (i.e., ferrite and molybdenum contents, casting process, and operating temperature) and/or neutron irradiation embrittlement (neutron fluence), and (b) for each "potentially susceptible" component, aging management is accomplished through either a supplemental examination of the affected component based on the neutron fluence to which the component has been exposed as part of the applicant's 10-year inservice inspection (ISI) program during the license renewal term, or a component-specific evaluation to determine its susceptibility to loss of fracture toughness.

Evaluation and Technical Basis

1. ***Scope of Program:*** The program provides screening criteria to determine the susceptibility of CASS components to thermal aging on the basis of casting method, molybdenum content, and percent ferrite. The screening criteria are applicable to all primary pressure boundary and reactor vessel internal components constructed from SA-351 Grades CF3, CF3A, CF8, CF8A, CF3M, CF3MA, CF8M, with service conditions above 250°C (482°F). The screening criteria for susceptibility to thermal aging embrittlement are not applicable to niobium-containing steels; such steels require evaluation on a case-by-case basis. For "potentially susceptible" components, the program provides for the consideration of the synergistic loss of fracture toughness due to neutron embrittlement and thermal aging embrittlement. For each such component, an applicant can implement either (a) a supplemental examination of the affected component as part of a 10-year ISI program during the license renewal term, or (b) a component-specific evaluation to determine the component's susceptibility to loss of fracture toughness.

   Based on the criteria set forth in the May 19, 2000, Nuclear Regulatory Commission (NRC) letter, the susceptibility to thermal aging embrittlement of CASS components is determined in terms of casting method, molybdenum content, and ferrite content. For low-molybdenum content (0.5 wt.% max.) steels, only static-cast steels with >20% ferrite are potentially susceptible to thermal embrittlement. Static-cast low-molybdenum steels with ≤20% ferrite and all centrifugal-cast low-molybdenum steels are not susceptible. For high-molybdenum content (2.0 to 3.0 wt.%) steels, static-cast steels with >14% ferrite and centrifugal-cast steels with >20% ferrite are potentially susceptible to thermal embrittlement. Static-cast high-molybdenum steels with ≤14% ferrite and centrifugal-cast high-molybdenum steels with ≤20% ferrite are not susceptible. In the susceptibility screening method, ferrite content is calculated by using the Hull's equivalent factors (described in NUREG/CR-4513, Rev. 1) or a method producing an equivalent level of accuracy (±6% deviation between measured and calculated values). A fracture toughness value of 255 kJ/m^2 (1,450 $in.\text{-}lb/in.^2$) at a crack depth of 2.5 mm (0.1 in.) is used to differentiate between CASS materials that are nonsusceptible and potentially susceptible to thermal aging embrittlement. Extensive research data indicate that for nonsusceptible CASS materials, the saturated lower-bound fracture toughness is greater than 255 kJ/m^2 (NUREG/CR-4513, Rev. 1).

2. ***Preventive Actions:*** The program consists of evaluation and inspection and provides no guidance on methods to mitigate thermal aging, neutron irradiation embrittlement or void swelling.

3. ***Parameters Monitored/Inspected:*** The program specifics depend on the neutron fluence and thermal embrittlement susceptibility of the component. The AMP monitors the effects of loss of fracture toughness on the intended function of the component by identifying the CASS materials that either have a neutron fluence of greater than 10^{17} n/cm^2 (E>1 MeV) or are determined to be susceptible to thermal aging embrittlement. For such materials, the program consists of either supplemental examination of the affected component based on the neutron fluence to which the component has been exposed, or component-specific evaluation to determine the component’s susceptibility to loss of fracture toughness.

4. ***Detection of Aging Effects:*** For all CASS components that have a neutron fluence of greater than 10^{17} n/cm^2 (E>1 MeV) or are determined to be susceptible to thermal embrittlement, the 10-year ISI program during the renewal period includes a supplemental inspection covering portions of the susceptible components determined to be limiting from the standpoint of thermal aging susceptibility (i.e., ferrite and molybdenum contents, casting process, and operating temperature), neutron fluence, and cracking susceptibility (i.e., applied stress, operating temperature, and environmental conditions). The inspection technique is capable of detecting the critical flaw size with adequate margin. The critical flaw size is determined based on the service loading condition and service-degraded material properties. One example of a supplemental examination is enhancement of the visual VT-1 examination of Section XI IWA-2210. A description of such an enhanced visual VT-1 examination could include the ability to achieve a 0.0005-in. resolution, with the conditions (e.g., lighting and surface cleanliness) of the inservice examination bounded by those used to demonstrate the resolution of the inspection technique. Alternatively, the applicant may perform a component-specific evaluation, including a mechanical loading assessment to determine the maximum tensile loading on the component during ASME Code Level A, B, C, and D conditions. If the loading is compressive or low enough (<5 ksi) to preclude fracture, then supplemental inspection of the component is not required. Failure to meet this criterion requires continued use of the supplemental inspection program. For each CASS component that has been subjected to a neutron fluence of less than 10^{17} n/cm^2 (E>1 MeV) and is potentially susceptible to thermal aging, the supplement inspection program applies; otherwise, the existing ASME Section XI inspection requirements are adequate if the components are not susceptible to thermal aging embrittlement.

5. ***Monitoring and Trending:*** Inspections scheduled in accordance with IWB-2400 and reliable examination methods provide timely detection of cracks.

6. ***Acceptance Criteria:*** Flaws detected in CASS components are evaluated in accordance with the applicable procedures of IWB-3500. Flaw tolerance evaluation for components with ferrite content up to 25% is performed according to the principles associated with IWB-3640 procedures for submerged arc welds (SAW), disregarding the Code restriction of 20% ferrite in IWB-3641(b)(1). Extensive research data indicate that the lower-bound fracture toughness of thermally aged CASS materials with up to 25% ferrite is similar to that for SAWs with up to 20% ferrite (Lee et al., 1997). Flaw evaluation for CASS components with

>25% ferrite is performed on a case-by-case basis by using fracture toughness data provided by the applicant.

7. ***Corrective Actions:*** Repair is in conformance with IWA-4000 and IWB-4000, and replacement is in accordance with IWA-7000 and IWB-7000. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable in addressing corrective actions.

8. ***Confirmation Process:*** Site quality assurance (QA) procedures, review and approval processes, and administrative controls are implemented in accordance with the requirements of 10 CFR Part 50, Appendix B. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable in addressing the confirmation process and administrative controls.

9. ***Administrative Controls:*** See Item 8, above.

10. ***Operating Experience:*** The proposed AMP was developed by using research data obtained on both laboratory-aged and service-aged materials. Based on this information, the effects of thermal aging embrittlement on the intended function of CASS components are effectively managed.

References

ASME Section XI, *Rules for Inservice Inspection of Nuclear Power Plant Components*, ASME Boiler and Pressure Vessel Code, 1995 edition through the 1996 addenda, American Society of Mechanical Engineers, New York, NY.

Lee, S., Kuo, P. T., Wichman, K., and Chopra, O., *Flaw Evaluation of Thermally Aged Cast Stainless Steel in Light-Water Reactor Applications*, Int. J. Pres. Ves. and Piping, 72, pp. 37-44, 1997.

Letter from Christopher I. Grimes, U.S. Nuclear Regulatory Commission, License Renewal and Standardization Branch, to Douglas J. Walters, Nuclear Energy Institute, License Renewal Issue No. 98-0030, *Thermal Aging Embrittlement of Cast Stainless Steel Components*, May 19, 2000.

NUREG/CR-4513, Rev. 1, *Estimation of Fracture Toughness of Cast Stainless Steels during Thermal Aging in LWR Systems*, U.S. Nuclear Regulatory Commission, August 1994.

XI.M14 LOOSE PART MONITORING

Program Description

The program relies on an inservice monitoring program to detect and monitor loose parts in light-water reactor (LWR) power plants. This inservice loose part monitoring (LPM) program is based on the recommendations from the American Society of Mechanical Engineers operation and maintenance standards and guides (ASME OM-S/G)-1997, Part 12, "Loose Part Monitoring in Light-Water Reactor Power Plants."

Evaluation and Technical Basis

1. ***Scope of Program:*** The program includes measures to monitor and detect metallic loose parts by using transient signals analysis on acoustic data generated due to loose parts impact. The program is applicable, but not necessarily limited to, the reactor vessel and primary coolant systems in pressurized water reactors (PWRs) and the reactor recirculation system in boiling water reactors (BWRs). The detection and monitoring system includes a set of accelerometers installed in the vicinity of regions where loose parts impact is likely to occur. The system incorporates the capability of automatic annunciation (audible and visual), audio monitoring, automatic and manual signal recording, and acoustic signal analysis/evaluation. Measures for personnel radiation exposure and safety are included as part of the requirements of the LPM system. The objective of the LPM program is to provide early indication of component degradation.

2. ***Preventive Actions:*** The aging management program (AMP) is a monitoring/detection program that provides early indication and detection of the onset of aging degradation. It does not rely on preventive actions.

3. ***Parameters Monitored/Inspected:*** The program relies on the use of transient acoustic signals to provide information on the occurrence of metallic loose part impact. Reactor coolant system (RCS) background noise may mask the noise generated due to loose part impact. These background noises may arise from sources such as coolant flow and mechanically and hydraulically generated vibrations. To differentiate loose part impact noise from background noise, ASME OM-S/G-1997, Part 12, recommends that the monitoring system sensitivity be set on the basis of the background noise and that maximum sensitivity be accomplished that is consistent with an acceptable false alarm rate arising from the background noise.

4. ***Detection of Aging Effects:*** Impact signals contain significant information on the size of the impacting object, the impact force and energy, and the composition and shape of both the component struck and the impacting object. In general, the magnitude of the impact signal increases with the impact mass and impact velocity. However, the frequency response increases with increasing velocity and decreasing mass. These data may be used to extract information on possible loose part impact and differentiation from background noise.

5. ***Monitoring and Trending:*** The impact signals, collected data, frequency, and characteristics are recorded, monitored, and evaluated to locate and identify the source and cause of the acoustic signals for the purpose of determining the need and urgency for a detailed inspection and examination of the suspected reactor vessel internals components. These activities are performed and associated personnel are qualified in accordance with

site controlled procedures and processes, as indicated by vendor, industry, or regulatory guidance documents.

6. ***Acceptance Criteria:*** The LPM alarms that suggest metallic impacts are further evaluated to verify LPM operability and to determine the location of the impact, the impact energy, and mass. Plant process data are reviewed for anomalous behavior, and diagnostic results are assessed by plant personnel.

7. ***Corrective Actions:*** If LPM diagnostics indicate the presence of loose parts, then corrective actions are taken. In some cases, the results of the diagnostic may indicate the signal is due to a change in the plant background characteristics and not due to the presence of loose parts. In such cases, the LPM alarm rates may in time become so high as to be unacceptable in practice. Adjustment of the alarm threshold (setpoints) is allowed. However, the reason for the change in background is to be investigated and understood, and the change is to be documented. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable in addressing corrective actions.

8. ***Confirmation Process:*** Site quality assurance (QA) procedures, review and approval processes, and administrative controls are implemented in accordance with the requirements of 10 CFR Part 50, Appendix B. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable in addressing the confirmation process and administrative controls.

9. ***Administrative Controls:*** See Item 8, above.

10. ***Operating Experience:*** The loose part monitoring program is extensively and effectively used by the industry. The program has been developed and published as a standard in the ASME "Standards and Guides for Operation and Maintenance of Nuclear Power Plants," Part 12, an American National Standard. Part 12 was developed on the basis of knowledge gained from operating experience and research conducted since the Nuclear Regulatory Commission (NRC) issued Regulatory Guide (RG) 1.133 in May 1981.

References

ANSI S2.11-1969, *American National Standard for the Selection of Calibrations and Tests for Electrical Transducers Used for Monitoring Shock and Vibrations*, American National Standards Institute, Washington, DC, 1969.

ASME OM-S/G-1997, Part 12, *Loose Part Monitoring in Light-Water Reactor Power Plants*, American Society of Mechanical Engineers, New York, NY, 1997.

NRC Regulatory Guide 1.133, Rev. 1, *Loose Part Detection Program for the Primary System of Light Water Cooled Reactors*, U.S. Nuclear Regulatory Commission, 1981.

XI.M15 NEUTRON NOISE MONITORING

Program Description

The program relies on monitoring the excore neutron detector signals due to core motion to detect and monitor significant loss of axial preload at the core support barrel's upper support flange in pressurized water reactors (PWRs). This inservice monitoring program is based on the recommendations from the American Society of Mechanical Engineers operation and maintenance standards and guides (ASME OM-S/G)-1997, Part 5, "Inservice Monitoring of Core Support Barrel Axial Preload in Pressurized Water Reactors Power Plants."

Evaluation and Technical Basis

1. ***Scope of Program:*** The program includes measures to monitor and detect loss of axial preload (loss of axial restraint) at the core support barrel's upper support flange in PWRs. The loss of axial restraint may arise from long-term changes resulting from abnormal wear at the reactor vessel core barrel mating surface or short-term changes due to improper installation of the reactor internals. The program also includes guidelines for further data acquisition that may be needed to define future plant operation and/or program plans in order to maintain the capability of the structure/components to perform the intended function.

2. ***Preventive Actions:*** The aging management program (AMP) is a monitoring/detection program that provides early indication and detection of the onset of aging degradation of the core support barrel holddown mechanism prior to a scheduled shutdown, thus reducing outage time and avoiding potential damage to the core support barrel and fuel assemblies. The AMP does not rely on preventive actions.

3. ***Parameters Monitored/Inspected:*** The program relies on the use of excore neutron detector signals to provide information on the conditions of the axial preload. The excore neutron flux signal is composed of a steady state, direct current (DC), component that arises from the neutron flux produced by the power operation of the reactor, as well as a fluctuating (noise-like) component. This fluctuating signal arises from the core reactivity changes due to lateral core motion from the loss of axial preload. This core motion is mainly the result of beam mode vibration of the core support barrel. Despite the fact that this beam mode vibration provides only a very weak neutron noise source, it may be reliably detected and identified through Fourier Analysis of the fluctuating signal component of the excore neutron flux signal. This signal component has the characteristics of having 180-degree shifts and a high degree of coherence between signals obtained from pairs of excore neutron detectors that are positioned on diametrically opposite sides of the core. The neutron noise signals are characterized by parameters, which include the auto correlation, cross correlation, coherence, and phase. These parameters are to be monitored and evaluated.

4. ***Detection of Aging Effects:*** Flow-induced vibration of the core support barrel will change the thickness of the downcomer annulus (water gap). This variation in the thickness will give rise to fluctuating changes in the neutron flux, as monitored by the excore neutron detectors. The natural frequencies and the amplitudes of the vibratory motion of the core barrel are related to the effective axial preload at the upper support flange of the core support barrel. Monitoring of the neutron noise signal obtained with the neutron flux detectors located around the external periphery of the reactor vessel provides detection of anomalous vibrational motion of the core support barrel, and hence significant loss of the

axial preload. Decrease in the axial preload leads to decreases in the core support barrel beam mode frequency and an increase in the magnitude of the noise signal. The overall effect of a decrease in the axial preload is to shift the neutron noise power spectrum toward larger amplitudes for the lower frequency region.

5. ***Monitoring and Trending:*** The neutron noise random fluctuation in the signals from the excore detectors are monitored, recorded, and analyzed to identify changes in the beam mode natural frequency of the core support barrel and its direction of motion for the purpose of a timely determination of the need and urgency for a detailed inspection and examination of the reactor vessel internals hold-down mechanism and mating component surfaces. These activities and analytical methodology are performed and associated personnel are qualified in accordance with site-controlled procedures and processes as indicated by vendor, industry, or regulatory guidance documents.

   The neutron noise monitoring program has three separate phases: a baseline phase, a surveillance phase, and a diagnostic phase. The baseline phase establishes the database to be used as a reference for developing limits and trends in the surveillance phase and to support data evaluation and interpretation in the diagnostic phase. During the baseline phase, data on the time history and DC level of each neutron flux detector and each cross-core detector pair are obtained. From this database, the characteristic amplitudes and frequencies of the core barrel motion are extracted. The wide and narrow frequency bands with their associated normalized root mean square (NRMS) values are established. The ASME-OMS/G-1997, Part 5, recommends collecting the baseline data during the first fuel cycle that the neutron noise monitoring program is applied to an already operating plant. Whenever significant changes takes place for the core, reactor internals, or operating conditions, then additional baseline data is obtained.

   In the surveillance phase, routine neutron noise monitoring of normal plant operations is performed over the life of the plant. The DC level and data for frequency analysis of each detector and two pair of cross-core detectors, may be collected. Comparisons of the measured amplitude and frequency data, with limits established from the baseline data, are made. In using neutron noise monitoring, accounts are taken of the effect of core burn-up, decreasing boron concentration, changes in fuel management, and in-core contact with the reactor vessel mechanical snubbers, which may affect the neutron noise signatures. Proper allowances for these factors during the baseline and surveillance phases will help toward detecting loss of axial preload before the core barrel becomes sufficiently free to wear against the reactor vessel and will also reduce the need to invoke the diagnostic phase.

   If the diagnostic phase becomes necessary, then evaluations are carried out to establish whether any deviations from the baseline data detected during the surveillance phase arises from core barrel motion due to loss of axial preload. The need and frequency of additional data collection on the time history and DC level of each neutron flux detector and each cross-core detector pair collection are guided by the results of these evaluations.

6. ***Acceptance Criteria:*** If evaluation of the baseline data indicates normal operation for the applicable structure/component then the surveillance phase may commence. If evaluation indicates anomalous behavior, then the monitoring program enters the diagnostic phase. During the surveillance phase, if deviations from the baseline fall within predetermined acceptable limits, then the surveillance will continue. Otherwise, the diagnostic phase will commence.

7. ***Corrective Actions:*** Initial results from the diagnostic phase of the program may be used to determine whether there is a need to increase the minimum frequency with which the surveillance data are acquired. In addition, if necessary, corrective actions may be taken to change the type of data acquisition and analysis from that previously recommended for the surveillance part of the program. The data trends may be established to guide further data acquisition that may be needed to define future plant operation and/or program plans. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable in addressing corrective actions.

8. ***Confirmation Process:*** Site quality assurance (QA) procedures, review and approval processes, and administrative controls are implemented in accordance with the requirements of 10 CFR Part 50, Appendix B. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable in addressing the confirmation process and administrative controls.

9. ***Administrative Controls:*** See Item 8, above.

10. ***Operating Experience:*** The neutron noise monitoring program and procedures were developed by the industry and published as a guide in ASME OM-S/G-1997, Part 5, an American National Standard. This monitoring program and procedures have been effective in limited industry use for monitoring and detecting loss of core support barrel axial preload in PWR power plants.

References

ASME OM-S/G-1997, Part 5, *Inservice Monitoring of Core Support Barrel Axial Preload in Pressurized Water Reactor Power Plants*, American Society of Mechanical Engineers, New York, NY, 1997.

XI.M16 PWR VESSEL INTERNALS

Program Description

The program includes (a) augmentation of the inservice inspection (ISI) in accordance with the American Society of Mechanical Engineers (ASME) Code, Section XI, Subsection IWB, Table IWB 2500-1 (1995 edition through the 1996 addenda) for certain susceptible or limiting components or locations, and (b) monitoring and control of reactor coolant water chemistry in accordance with the Electric Power Research Institute (EPRI) guidelines in TR-105714 to ensure the long-term integrity and safe operation of pressurized water reactor (PWR) vessel internal components. The ASME Section XI ISI is augmented with enhancing the VT-1 examinations for non-bolted components for example, to include the ability to achieve a 0.0005-inch resolution. The inspection methods for bolted components are to be demonstrated for detecting cracks between the bolt head and the shank.

Evaluation and Technical Basis

1. ***Scope of Program:*** The program is focused on managing the effects of crack initiation and growth due to stress corrosion cracking (SCC) or irradiation assisted stress corrosion cracking (IASCC), and loss of fracture toughness due to neutron irradiation embrittlement or void swelling. The program contains preventive measures to mitigate SCC or IASCC; ISI to monitor the effects of cracking on the intended function of the components; and repair and/or replacement as needed to maintain the ability to perform the intended function. Loss of fracture toughness is of consequence only if cracks exist. Cracking is expected to initiate at the surface and is detectable by augmented inspection.

   The program provides guidelines to assure safety function integrity of the subject safety-related reactor pressure vessel internal components, both non-bolted and bolted components. The program consists of the following elements: (a) identify the most susceptible or limiting items, (b) develop appropriate inspection techniques to permit detection and characterizing of the feature (cracks) of interest and demonstrate the effectiveness of the proposed technique, and (c) implement the inspection during the license renewal term. For example, appropriate inspection techniques may include enhancing visual VT-1 examinations for non-bolted components and demonstrated acceptable inspection methods for bolted components.

2. ***Preventive Actions:*** The requirements of ASME Section XI, Subsection IWB, provide guidance on detection, but do not provide guidance on methods to mitigate cracking. Maintaining high water purity reduces susceptibility to cracking due to SCC. Reactor coolant water chemistry is monitored and maintained in accordance with the EPRI guidelines in TR-105714. The program description and evaluation and technical basis of monitoring and maintaining reactor water chemistry are presented in Chapter XI.M2, “Water Chemistry.”

3. ***Parameters Monitored/Inspected:*** The program monitors the effects of cracking on the intended function of the component by detection and sizing of cracks by augmentation of ISI in accordance with the requirements of the ASME Code, Section XI, Table IWB 2500-1.

4. ***Detection of Aging Effects:*** The extent and schedule of the inspection and test techniques prescribed by the aging management program are designed to maintain structural integrity and ensure that aging effects will be discovered and repaired before the loss of intended function. Inspection can reveal crack initiation and growth. Vessel internal components are inspected in accordance with the requirements of ASME Section XI, Subsection IWB,

examination category B-N-3 for all accessible surfaces of reactor core support structure that can be removed from the vessel. The ASME Section XI inspection specifies visual VT-3 examination to determine the general mechanical and structural condition of the component supports by (a) verifying parameters, such as clearances, settings, and physical displacements, and (b) detecting discontinuities and imperfections, such as loss of integrity at bolted or welded connections, loose or missing parts, debris, corrosion, wear, or erosion.

However, visual VT-3 examination is to be augmented to detect tight or fine cracks. Also, historically the VT-3 examinations have not identified bolt cracking because cracking occurs at the juncture of the bolt head and shank, which is not accessible for visual inspection. Creviced and other inaccessible regions are difficult to inspect visually. This AMP recommends more stringent inspections such as enhanced visual VT-1 examinations or ultrasonic methods of volumetric inspection, for certain selected components and locations.

The inspection technique is capable of detecting the critical flaw size with adequate margin. The critical flaw size is determined based on the service loading condition and service-degraded material properties. For non-bolted components, augmented ISI may include enhancement of the visual VT-1 examination of Section XI IWA-2210. A description of such an enhanced visual VT-1 examination should include the ability to achieve a 0.0005-in. resolution, with the conditions (e.g., lighting and surface cleanliness) of the inservice examination bounded by those used to demonstrate the resolution of the inspection technique. For bolted components, augmented ISI is to include other demonstrated acceptable inspection methods to detect cracks between the bolt head and the shank. Alternatively, the applicant may perform a component-specific evaluation, including a mechanical loading assessment to determine the maximum tensile loading on the component during ASME Code Level A, B, C, and D conditions. If the loading is compressive or low enough (<5 ksi) to preclude fracture, then supplemental inspection of the component is not required. Failure to meet this criterion requires continued use of the augmented inspection methods.

5. ***Monitoring and Trending:*** Inspection schedules in accordance with IWB-2400, assessment of susceptible or limiting components or locations, and reliable examination methods provide timely detection of cracks. The scope of examination expansion and re-inspection beyond the baseline inspection are required if flaws are detected.

6. ***Acceptance Criteria:*** Any indication or relevant condition of degradation is evaluated in accordance with IWB-3100 by comparing ISI results with the acceptance standards of IWB-3400 and IWB-3500.

7. ***Corrective Actions:*** Repair and replacement procedures are equivalent to those requirements in ASME Section XI. Repair is in conformance with IWB-4000 and replacement occurs according to IWB-7000. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable in addressing corrective actions.

8. ***Confirmation Process:*** Site quality assurance (QA) procedures, review and approval processes, and administrative controls are implemented in accordance with the requirements of 10 CFR 50, Appendix B. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable in addressing the confirmation process and administrative controls.

9. ***Administrative Controls:*** See Item 8, above.

10. ***Operating Experience:*** Because the ASME Code is a consensus document that has been widely used over a long period, it has been shown to be generally effective in managing aging effects in Class 1, 2, or 3 components and their integral attachments in light-water cooled power plants.

    In PWRs, stainless steel components have generally not been found to be affected by SCC because of low dissolved oxygen levels and control of primary water chemistry. However, the potential for SCC exists due to inadvertent introduction of contaminants into the primary coolant system from unacceptable levels of contaminants in the boric acid; introduction through the free surface of the spent fuel pool, which can be a natural collector of airborne contaminants (NRC IN 84-18); introduction of relatively high levels of oxygen during shutdown, or from aggressive chemistries that may develop in creviced regions. Cracking has occurred in SS baffle former bolts in a number of foreign plants (NRC IN 98-11) and has now been observed in plants in the United States.

References

ASME Section XI, *Rules for Inservice Inspection of Nuclear Power Plant Components*, ASME Boiler and Pressure Vessel Code, 1995 edition through the 1996 addenda, American Society of Mechanical Engineers, New York, NY.

EPRI TR-105714, *PWR primary Water Chemistry Guidelines-Revision 3*, Electric Power Research Institute, Palo Alto, CA, November 1995.

NRC Information Notice 84-18, *Stress Corrosion Cracking in PWR Systems*, March 7, 1984.

NRC Information Notice 98-11, *Cracking of Reactor Vessel Internal Baffle Former Bolts in Foreign Plants*, March 25, 1998.

XI.M17 FLOW-ACCELERATED CORROSION

Program Description

The program relies on implementation of the Electric Power Research Institute (EPRI) guidelines in the Nuclear Safety Analysis Center (NSAC)-202L-R2 for an effective flow-accelerated corrosion (FAC) program. The program includes performing (a) an analysis to determine critical locations, (b) limited baseline inspections to determine the extent of thinning at these locations, and (c) follow-up inspections to confirm the predictions, or repairing or replacing components as necessary.

Evaluation and Technical Basis

1. ***Scope of Program:*** The FAC program, described by the EPRI guidelines in NSAC-202L-R2, includes procedures or administrative controls to assure that the structural integrity of all carbon steel lines containing high-energy fluids (two phase as well as single phase) is maintained. Valve bodies retaining pressure in these high-energy systems are also covered by the program. The FAC program was originally outlined in NUREG-1344 and was further described through the Nuclear Regulatory Commission (NRC) Generic Letter (GL) 89-08. A program implemented in accordance with the EPRI guidelines predicts, detects, and monitors FAC in plant piping and other components, such as valve bodies, elbows and expanders. Such a program includes the following recommendations: (a) conducting an analysis to determine critical locations; (b) performing limited baseline inspections to determine the extent of thinning at these locations; and (c) performing follow-up inspections to confirm the predictions, or repairing or replacing components as necessary. The NSAC-202L-R2 (April 1999) provides general guidelines for the FAC program. To ensure that all the aging effects caused by FAC are properly managed, the program includes the use of a predictive code, such as CHECWORKS, that uses the implementation guidance of NSAC-202L-R2 to satisfy the criteria specified in 10 CFR Part 50, Appendix B, criteria for development of procedures and control of special processes.

2. ***Preventive Actions:*** The FAC program is an analysis, inspection, and verification program; thus, there is no preventive action. However, it is noted that monitoring of water chemistry to control pH and dissolved oxygen content, and selection of appropriate piping material, geometry, and hydrodynamic conditions, are effective in reducing FAC.

3. ***Parameters Monitored/Inspected:*** The aging management program (AMP) monitors the effects of FAC on the intended function of piping and components by measuring wall thickness.

4. ***Detection of Aging Effects:*** Degradation of piping and components occurs by wall thinning. The inspection program delineated in NSAC-202L consists of identification of susceptible locations as indicated by operating conditions or special considerations. Ultrasonic and radiographic testing is used to detect wall thinning. The extent and schedule of the inspections assure detection of wall thinning before the loss of intended function.

5. ***Monitoring and Trending:*** CHECWORKS or a similar predictive code is used to predict component degradation in the systems conducive to FAC, as indicated by specific plant data, including material, hydrodynamic, and operating conditions. CHECWORKS is acceptable because it provides a bounding analysis for FAC. CHECWORKS was developed and benchmarked by using data obtained from many plants. The inspection schedule developed by the licensee on the basis of the results of such a predictive code provides

reasonable assurance that structural integrity will be maintained between inspections. If degradation is detected such that the wall thickness is less than the minimum predicted thickness, additional examinations are performed in adjacent areas to bound the thinning.

6. ***Acceptance Criteria:*** Inspection results are used as input to a predictive computer code, such as CHECWORKS, to calculate the number of refueling or operating cycles remaining before the component reaches the minimum allowable wall thickness. If calculations indicate that an area will reach the minimum allowed thickness before the next scheduled outage, the component is to be repaired, replaced, or reevaluated.

7. ***Corrective Actions:*** Prior to service, reevaluate, repair, or replace components for which the acceptance criteria are not satisfied. Longer term corrective actions could consist of adjustment of operating parameters or selection of materials resistant to FAC. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable in addressing corrective actions.

8. ***Confirmation Process:*** Site quality assurance (QA) procedures, review and approval processes, and administrative controls are implemented in accordance with the requirements of 10 CFR Part 50, Appendix B. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable in addressing the confirmation process and administrative controls.

9. ***Administrative Controls:*** See Item 8, above.

10. ***Operating Experience:*** Wall-thinning problems in single-phase systems have occurred in feedwater and condensate systems (NRC IE Bulletin No. 87-01; NRC Information Notices [INs] 81-28, 92-35, 95-11) and in two-phase piping in extraction steam lines (NRC INs 89-53, 97-84) and moisture separation reheater and feedwater heater drains (NRC INs 89-53, 91-18, 93-21, 97-84). Operating experience shows that the present program, when properly implemented, is effective in managing FAC in high-energy carbon steel piping and components.

References

NRC Generic Letter 89-08, *Erosion/Corrosion-Induced Pipe Wall Thinning*, U.S. Nuclear Regulatory Commission, May 2, 1989.

NRC IE Bulletin 87-01, *Thinning of Pipe Walls in Nuclear Power Plants*, U.S. Nuclear Regulatory Commission, July 9, 1987.

NRC Information Notice 81-28, *Failure of Rockwell-Edward Main Steam Isolation Valves*, U.S. Nuclear Regulatory Commission, September 3, 1981.

NRC Information Notice 89-53, *Rupture of Extraction Steam Line on High Pressure Turbine*, U.S. Nuclear Regulatory Commission, June 13, 1989.

NRC Information Notice 91-18, *High-Energy Piping Failures Caused by Wall Thinning*, U.S. Nuclear Regulatory Commission, March 12, 1991.

NRC Information Notice 91-18, Supplement 1, *High-Energy Piping Failures Caused by Wall Thinning*, U.S. Nuclear Regulatory Commission, December 18, 1991.

NRC Information Notice 92-35, *Higher than Predicted Erosion/Corrosion in Unisolable Reactor Coolant Pressure Boundary Piping inside Containment at a Boiling Water Reactor*, U.S. Nuclear Regulatory Commission, May 6, 1992.

NRC Information Notice 93-21, *Summary of NRC Staff Observations Compiled during Engineering Audits or Inspections of Licensee Erosion/Corrosion Programs*, U.S. Nuclear Regulatory Commission, March 25, 1993.

NRC Information Notice 95-11, *Failure of Condensate Piping Because of Erosion/Corrosion at a Flow Straightening Device*, U.S. Nuclear Regulatory Commission, February 24, 1995.

NRC Information Notice 97-84, *Rupture in Extraction Steam Piping as a Result of Flow-Accelerated Corrosion*, U.S. Nuclear Regulatory Commission, December 11, 1997.

NSAC-202L-R2, *Recommendations for an Effective Flow Accelerated Corrosion Program*, Electric Power Research Institute, Palo Alto, CA, April 8, 1999.

NUREG-1344, *Erosion/Corrosion-Induced Pipe Wall Thinning in U.S. Nuclear Power Plants*, P. C. Wu, U.S. Nuclear Regulatory Commission, April 1989.

XI.M18 BOLTING INTEGRITY

Program Description

The program relies on recommendations for a comprehensive bolting integrity program, as delineated in NUREG-1339, and industry recommendations, as delineated in the Electric Power Research Institute (EPRI) NP-5769, with the exceptions noted in NUREG-1339 for safety related bolting. The program relies on industry recommendations for a comprehensive bolting maintenance, as delineated in the EPRI TR-104213 for pressure retaining bolting and structural bolting. The program generally includes periodic inspection of closure bolting for indication of loss of preload, cracking, and loss of material due to corrosion, rust, etc.

Evaluation and Technical Basis

1. ***Scope of Program:*** The program covers all bolting within the scope of license renewal including safety-related bolting, bolting for NSSS component supports, bolting for other pressure retaining components, and structural bolting. The program covers both greater than and smaller than 2-in. diameter bolting. The Nuclear Regulatory Commission (NRC) staff recommendations and guidelines for comprehensive bolting integrity programs that encompass all safety-related bolting are delineated in NUREG-1339. The industry's technical basis for the program for safety related bolting and guidelines for material selection and testing, bolting preload control, inservice inspection (ISI), plant operation and maintenance, and evaluation of the structural integrity of bolted joints, are outlined in EPRI NP-5769, with the exceptions noted in NUREG 1339. For other bolting, this information is set forth in EPRI TR-104213.

2. ***Preventive Actions:*** Selection of bolting material and the use of lubricants and sealants is in accordance with the guidelines of EPRI NP-5769 and the additional recommendations of NUREG-1339 to prevent or mitigate degradation and failure of safety-related bolting (see item 10, below). (NUREG-1339 takes exception to certain items in EPRI NP-5769, and recommends additional measures with regard to them.) Initial ISI of bolting for pressure retaining components includes a check of the bolt torque and uniformity of the gasket compression after assembly. It is noted that hot torquing of bolting is a leak preventive measure once the joint is brought to operating temperature and before or after it is pressurized. Hot torquing thus reestablishes preload before leak starts, but is ineffective in sealing a leak once it has begun.

3. ***Parameters Monitored/Inspected:*** The aging management program (AMP) monitors the effects of aging on the intended function of closure bolting, including loss of material, cracking, and loss of preload. High strength bolts (actual yield strength ≥ 150 ksi) used in NSSS component supports are monitored for cracking. Bolting for pressure retaining components is inspected for signs of leakage. Structural bolting is inspected for indication of potential problems including loss of coating integrity and obvious signs of corrosion, rust, etc.

4. ***Detection of Aging Effects:*** Inspection requirements are in accordance with the American Society of Mechanical Engineers (ASME) Section XI, Table IWB 2500-1 or IWC 2500-1 (1995 edition through the 1996 addenda) and the recommendations of EPRI NP-5769. For Class 1 components, Table IWB 2500-1, examination category B-G-1, for bolting greater than 2 in. in diameter, specifies volumetric examination of studs and bolts and visual VT-1 examination of surfaces of nuts, washers, bushings, and flanges. All high strength bolting used in NSSS component supports are to be inspected also to the requirements for Class 1

components, examination category B-G-1. Examination category B-G-2, for bolting 2 in. or smaller requires only visual VT-1 examination of surfaces of bolts, studs, and nuts. For Class 2 components, Table IWC 2500-1, examination category B-D, for bolting greater than 2 in. in diameter, requires volumetric examination of studs and bolts. Examination categories B-P or C-H require visual examination (IWA-5240) during system leakage testing of all pressure-retaining Class 1 and 2 components, according to Tables IWB 2500-1 and IWC 2500-1, respectively. In addition, degradation of the closure bolting due to crack initiation, loss of prestress, or loss of material due to corrosion of the closure bolting would result in leakage. The extent and schedule of inspections, in accordance with IWB 2500-1 or IWC 2500-1, assure detection of aging degradation before the loss of the intended function of the closure bolting. Structural bolting both inside and outside containment is inspected by visual inspection. Degradation of this bolting may be detected and measured either by removing the bolt, proof test by tension or torquing, by in situ ultrasonic tests, or hammer test. If this bolting is found corroded, a closer inspection is performed to assess extent of corrosion.

5. ***Monitoring and Trending:*** The inspection schedules of ASME Section XI are effective and ensure timely detection of cracks and leakage. If bolting for pressure retaining components (not covered by ASME Section XI) is reported to be leaking, then it may be inspected daily. If the leak rate does not increase, the inspection frequency may be decreased to weekly or biweekly.

6. ***Acceptance Criteria:*** Any indications in closure bolting are evaluated in accordance with IWB-3100 and acceptance standards of IWB-3400 and IWB-3500, or IWC-3100 and acceptance standards of IWC-3400 and IWC-3500. Indications of cracking in component support bolting warrant immediate replacement of the cracked bolt. For other pressure retaining components, a leak from a joint is immediately repaired if it is a major leak and causes adverse effect such as corrosion or contamination.

7. ***Corrective Actions:*** Repair and replacement is in conformance with IWB-4000 and guidelines and recommendations of EPRI NP-5769. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable in addressing corrective actions. Repair and replacement of other bolting including structural bolting is in conformance with the guidelines and recommendations of EPRI TR-104213.

8. ***Confirmation Process:*** Site quality assurance (QA) procedures, review and approval processes, and administrative controls are implemented in accordance with the requirements of 10 CFR Part 50, Appendix B. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable in addressing the confirmation process and administrative controls.

9. ***Administrative Controls:*** See item 8, above.

10. ***Operating Experience:*** Degradation of threaded fasteners in closures for the reactor coolant pressure boundary has occurred from boric acid corrosion, stress corrosion cracking, and fatigue loading (NRC IE Bulletin 82-02, NRC Generic Letter [GL] 91-17). Stress corrosion cracking has occurred in high strength bolts used for NSSS component supports. The bolting integrity programs developed and implemented in accordance with commitments made in response to NRC communications on bolting events have provided an effective means of ensuring bolting reliability. These programs are documented in EPRI NP-5769 and TR-104213 and represent industry consensus.

References

10 CFR 50.55a, *Codes and Standards*, Office of the Federal Register, National Archives and Records Administration, 2000.

ASME Section XI, *Rules for Inservice Inspection of Nuclear Power Plant Components*, ASME Boiler and Pressure Vessel Code, 1995 edition through the 1996 addenda, American Society of Mechanical Engineers, New York, NY.

EPRI NP-5769, *Degradation and Failure of Bolting in Nuclear Power Plants*, Volumes 1 and 2, Electric Power Research Institute, Palo Alto, CA, April 1988.

EPRI TR-104213, *Bolted Joint Maintenance & Application Guide,* Electric Power Research Institute, Palo Alto, CA, December 1995.

NRC Generic Letter 91-17, *Generic Safety Issue 79, "Bolting Degradation or Failure in Nuclear Power Plants,"* U.S. Nuclear Regulatory Commission, October 17, 1991.

NRC IE Bulletin No. 82-02, *Degradation of Threaded Fasteners in the Reactor Coolant Pressure Boundary of PWR Plants*, U.S. Nuclear Regulatory Commission, June 2, 1982.

NUREG-1339, *Resolution of Generic Safety Issue 29: Bolting Degradation or Failure in Nuclear Power Plants*, Richard E. Johnson, U.S. Nuclear Regulatory Commission, June 1990.

XI.M19 STEAM GENERATOR TUBE INTEGRITY

Program Description

Steam generator (SG) tubes have experienced tube degradation related to corrosion phenomena, such as primary water stress corrosion cracking (PWSCC), outside diameter stress corrosion cracking (ODSCC), intergranular attack (IGA), pitting, and wastage, along with other mechanically induced phenomena, such as denting, wear, impingement damage, and fatigue. Nondestructive examination (NDE) techniques are used to identify tubes that are defective and need to be removed from service or repaired in accordance with the guidelines of the plant technical specifications. In addition, operational leakage limits are included to ensure that, should substantial tube leakage develop, prompt action is taken to avoid rupture of the leaking tubes. These limits are included in plant technical specifications, such as standard technical specifications of NUREG-1430, Rev. 1, for Babcock & Wilcox pressurized water reactors (PWRs); NUREG-1431, Rev. 1, for Westinghouse PWRs; and NUREG-1432, Rev. 1, for Combustion Engineering PWRs.

The technical specifications specify SG inspection scope and frequency, and acceptance criteria for the plugging and repair of flawed tubes. The Nuclear Regulatory Commission (NRC) Regulatory Guide (RG) 1.121, “Bases for Plugging Degraded Steam Generator Tubes,” provides guidelines for determining the tube repair criteria and operational leakage limits. Acceptance criteria for the plugging and repair of flawed tubes are incorporated in the plant technical specifications.

However, plants may apply for changes in their technical specifications to provide an alternate regulatory basis for SG degradation management. The NRC has approved changes in the technical specification tube repair criteria at certain plants. Examples include the alternate voltage-based repair criteria of NRC Generic Letter (GL) 95-05 and certain sleeving process. In addition, all PWR licensees have committed voluntarily to a SG degradation management program described in the Nuclear Energy Institute (NEI) 97-06, “Steam Generator Program Guidelines.” This program references a number of industry guidelines and incorporates a balance of prevention, inspection, evaluation, repair, and leakage monitoring measures. These guidelines are currently under NRC review. The NEI 97-06 document (a) includes performance criteria that are intended to provide assurance that tube integrity is being maintained consistent with the plant’s licensing basis, and (b) provides guidance for monitoring and maintaining the tubes to provide assurance that the performance criteria are met at all times between scheduled inspections of the tubes. The NEI 97-06 program includes an assessment of degradation mechanisms that considers operating experience from similar SGs to identify degradation mechanisms and, for each mechanism, defines the inspection techniques, measurement uncertainty, as well as the sampling strategy. The industry guidelines provide criteria for the qualification of personnel, specific techniques, and the associated acquisition and analysis of data, including procedures, probe selection, analysis protocols, and reporting criteria. The performance criteria pertain to structural integrity, accident-induced leakage, and operational leakage. The SG monitoring program includes guidance on assessment of degradation mechanisms, inspection, tube integrity assessment, maintenance, plugging, repair, and leakage monitoring, as well as procedures for monitoring and controlling secondary-side and primary-side water chemistry. The water chemistry program for PWRs relies on monitoring and control of reactor water chemistry and secondary water chemistry.

As evaluated below, the plant technical specifications, incorporating NEI 97-06 as approved by the staff and any other alternate regulatory bases for SG degradation management that have been previously approved by the staff for that plant, are adequate to manage the effects of aging on the SG tubes. However, because NEI 97-06 is still under staff review, until the staff has approved NEI 97-06, the applicant's program should be reviewed on a plant-specific basis.

Evaluation and Technical Basis

1. ***Scope of Program:*** The scope of the program is specific to SG tubes. The program includes preventive measures to mitigate degradation related to corrosion phenomena; assessment of degradation mechanisms; inservice inspection (ISI) of steam generator tubes to detect degradation; evaluation and plugging or repair, as needed; and leakage monitoring to maintain the structural and leakage integrity of the pressure boundary. Tube inspection scope and frequency, plugging or repair, and leakage monitoring are in accordance with the plant technical specifications.

2. ***Preventive Actions:*** The program includes preventive measures to mitigate degradation related to corrosion phenomena. The guidelines in NEI 97-06 include foreign material exclusion as a means to inhibit fretting and wear degradation. The water chemistry program for PWRs relies on monitoring and control of reactor water chemistry based on the EPRI guidelines in TR-105714 for primary water chemistry and TR-102134 for secondary water chemistry. The program description and the evaluation and technical basis of monitoring and maintaining reactor water chemistry are presented in Chapter XI.M2, "Water Chemistry," of this report.

3. ***Parameters Monitored/Inspected:*** The inspection activities in the program detect flaws in tubing or degradation of secondary side internals needed to maintain tubing integrity. Flaws are removed based on technical specification repair criteria. Degradation of steam generator internals is evaluated for corrective actions.

4. ***Detection of Aging Effects:*** The inspection requirements in the technical specifications are intended to detect tube degradation (i.e., aging effects), if it should occur. The NEI 97-06 document, which is currently under NRC staff review, provides additional guidance on inspection programs to detect degradation. The intent of the inspection and repair criteria is to provide assurance of continued tube integrity between inspections.

5. ***Monitoring and Trending:*** Condition monitoring assessments are performed to determine whether structural and accident leakage criteria have been satisfied. Operational assessments are performed after inspections to verify that structural and leakage integrity are maintained during the operating interval until the next required inspection, which is selected in accordance with the technical specifications and staff approved NEI 97-06 guidelines. Comparison of the results of the condition monitoring assessment with the predictions of the previous operational assessment provides feedback for evaluation of the adequacy of the operational assessment and additional insights that can be incorporated into the next operational assessment.

6. ***Acceptance Criteria:*** Assessment of tube integrity and plugging or repair criteria of flawed tubes is in accordance with the plant technical specifications. The criteria for plugging or repairing SG tubes are based on NRC RG 1.121 or other criteria previously reviewed and approved by the staff and incorporated into the plant technical specifications. Some examples that are applicable under certain circumstances include P*, F*, L*, or NRC GL 95-05.

   For general and pitting corrosion, the acceptance criteria are in accordance with staff approved NEI 97-06 guidelines. Also, loose parts or foreign objects that are found are removed from the SGs unless it can be shown by evaluation that these objects do not cause unacceptable tube damage. The evaluation is to define an acceptable operating interval.

For Westinghouse steam generator tube plugs, limits for the life of the plug and correlations for estimating their life are contained in WCAP-12244 and WCAP-12245.

7. ***Corrective Actions:*** Tubes containing flaws that do not meet the acceptance criteria are plugged or repaired. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable in addressing corrective actions.

8. ***Confirmation Process:*** Site quality assurance (QA) procedures, review and approval processes, and administrative controls are implemented in accordance with the requirements of 10 CFR Part 50, Appendix B. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable in addressing the confirmation process and administrative controls.

9. ***Administrative Controls:*** See Item 8, above.

10. ***Operating Experience:*** Failures to detect some flaws, uncertainties in flaw sizing, inaccuracies in flaw locations, and the inability to detect some cracks at locations with dents have been reviewed in NRC Information Notice (IN) 97-88. Recent experience indicates the importance of performing a complete inspection by using appropriate techniques and equipment for the reliable detection of tube degradation and to provide assurance that new forms of degradation are detected. Implementation of the program provides reasonable assurance that SG tube integrity is maintained consistent with the plant's licensing basis for the period of extended operation. Experience with the condition and operational assessments required for plants that have implemented the alternate repair criteria in NRC GL 95-05 has shown that the predictions of the operational assessments have generally been consistent with the results of the subsequent condition monitoring assessments. In cases where discrepancies have been noted, adjustments have been made in the operational assessment models to improve agreement in subsequent assessments. In addition, NEI has prepared NEI 97-06 to incorporate lessons learned from plant operation experience and SG inspections and is under staff review.

References

EPRI TR-102134, *PWR Secondary Water Chemistry Guidelines: Revision 3*, Electric Power Research Institute, Palo Alto, CA, May 1993.

EPRI TR-105714, *PWR Primary Water Chemistry Guidelines: Revision 3*, Electric Power Research Institute, Palo Alto, CA, November 1995.

EPRI TR-107569, *PWR Steam Generator Examination Guidelines: Revision 5*, Electric Power Research Institute, Palo Alto, CA, September 1997.

NEI 97-06, Rev. 1, *Steam Generator Program Guidelines*, Nuclear Energy Institute, January 2000.

NRC Generic Letter 95-05, *Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside-Diameter Stress-Corrosion Cracking,* U.S. Nuclear Regulatory Commission, August 3, 1995.

NRC Information Notice, 97-88, *Experiences during Recent Steam Generator Inspections*, U.S. Nuclear Regulatory Commission, December 12, 1997.

NRC Regulatory Guide, 1.83, Rev. 1, *Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes*, U.S. Nuclear Regulatory Commission, July 1975.

NRC Regulatory Guide, 1.121, *Bases for Plugging Degraded PWR Steam Generator Tubes*, U.S. Nuclear Regulatory Commission, August 1976.

NUREG-1430, Rev. 1, *Standard Technical Specifications for Babcock and Wilcox Pressurized Water Reactors*, U.S. Nuclear Regulatory Commission, April 1995.

NUREG-1431, Rev. 1, *Standard Technical Specifications for Westinghouse Pressurized Water Reactors*, U.S. Nuclear Regulatory Commission, April 1995.

NUREG-1432, Rev. 1, *Standard Technical Specifications for Combustion Engineering Pressurized Water Reactors*, U.S. Nuclear Regulatory Commission, April 1995.

WCAP-12244 and WCAP-12245, *Steam Generator Tube Plug Integrity Summary Report*, Addendum 2 to Revision 3, Westinghouse Electric Corporation, PA, May 1991.

XI.M20 OPEN-CYCLE COOLING WATER SYSTEM

Program Description

The program relies on implementation of the recommendations of the Nuclear Regulatory Commission (NRC) Generic Letter (GL) 89-13 to ensure that the effects of aging on the open-cycle cooling water (OCCW) (or service water) system will be managed for the extended period of operation. The program includes surveillance and control techniques to manage aging effects caused by biofouling, corrosion, erosion, protective coating failures, and silting in the OCCW system or structures and components serviced by the OCCW system.

Evaluation and Technical Basis

1. ***Scope of Program:*** The program addresses the aging effects of material loss and fouling due to micro- or macro-organisms and various corrosion mechanisms. Because the characteristics of the service water system may be specific to each facility, the OCCW system is defined as a system or systems that transfer heat from safety-related systems, structures, and components (SSC) to the ultimate heat sink (UHS). If an intermediate system is used between the safety-related SSCs and the system rejecting heat to the UHS, that intermediate system performs the function of a service water system and is thus included in the scope of recommendations of NRC GL 89-13. The guidelines of NRC GL 89-13 include (a) surveillance and control of biofouling; (b) a test program to verify heat transfer capabilities; (c) routine inspection and a maintenance program to ensure that corrosion, erosion, protective coating failure, silting, and biofouling cannot degrade the performance of safety-related systems serviced by OCCW; (d) a system walkdown inspection to ensure compliance with the licensing basis; and (e) a review of maintenance, operating, and training practices and procedures.

2. ***Preventive Actions:*** The system components are constructed of appropriate materials and lined or coated to protect the underlying metal surfaces from being exposed to aggressive cooling water environments. Implementation of NRC GL 89-13 includes a condition and performance monitoring program; control or preventive measures, such as chemical treatment, whenever the potential for biological fouling species exists; or flushing of infrequently used systems. Treatment with chemicals mitigates microbiologically influenced corrosion (MIC) and buildup of macroscopic biological fouling species, such as blue mussels, oysters, or clams. Periodic flushing of the system removes accumulations of biofouling agents, corrosion products, and silt.

3. ***Parameters Monitored/Inspected:*** Adverse effects on system or component performance are caused by accumulations of biofouling agents, corrosion products, and silt. Cleanliness and material integrity of piping, components, heat exchangers, and their internal linings or coatings (when applicable) that are part of the OCCW system or that are cooled by the OCCW system are periodically inspected, monitored, or tested to ensure heat transfer capabilities. The program ensures (a) removal of accumulations of biofouling agents, corrosion products, and silt, and (b) detection of defective protective coatings and corroded OCCW system piping and components that could adversely affect performance of their intended safety functions.

4. ***Detection of Aging Effects:*** Inspections for biofouling, damaged coatings, and degraded material condition are conducted. Visual inspections are typically performed; however, nondestructive testing, such as ultrasonic testing, eddy current testing, and heat transfer capability testing, are effective methods to measure surface condition and the extent of wall

thinning associated with the service water system piping and components, when determined necessary.

5. ***Monitoring and Trending:*** Inspection scope, method (e.g., visual or nondestructive examination [NDE]), and testing frequencies are in accordance with the utility commitments under NRC GL 89-13. Testing and inspections are done annually and during refueling outages. Inspections or nondestructive testing will determine the extent of biofouling, the condition of the surface coating, the magnitude of localized pitting, and the amount of MIC, if applicable. Heat transfer testing results are documented in plant test procedures and are trended and reviewed by the appropriate group.

6. ***Acceptance Criteria:*** Biofouling is removed or reduced as part of the surveillance and control process. The program for managing biofouling and aggressive cooling water environments for OCCW systems is preventive. Acceptance criteria are based on effective cleaning of biological fouling organisms and maintenance of protective coatings or linings are emphasized.

7. ***Corrective Actions:*** Evaluations are performed for test or inspection results that do not satisfy established acceptance criteria and a problem or condition report is initiated to document the concern in accordance with plant administrative procedures. The corrective actions program ensures that the conditions adverse to quality are promptly corrected. If the deficiency is assessed to be significantly adverse to quality, the cause of the condition is determined, and an action plan is developed to preclude repetition. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable in addressing corrective actions.

8. ***Confirmation Process:*** Site quality assurance (QA) procedures, review and approval processes, and administrative controls are implemented in accordance with the requirements of 10 CFR Part 50, Appendix B. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable in addressing the confirmation process and administrative controls.

9. ***Administrative Controls:*** See Item 8, above.

10. ***Operating Experience:*** Significant microbiologically influenced corrosion (NRC Information Notice [IN] 85-30), failure of protective coatings (NRC IN 85-24), and fouling (NRC IN 81-21, IN 86-96) have been observed in a number of heat exchangers. The guidance of NRC GL 89-13 has been implemented for approximately 10 years and has been effective in managing aging effects due to biofouling, corrosion, erosion, protective coating failures, and silting in structures and components serviced by OCCW systems.

References

NRC Generic Letter 89-13, *Service Water System Problems Affecting Safety-Related Equipment*, U.S. Nuclear Regulatory Commission, July 18, 1989.

NRC Generic Letter 89-13, Supplement 1, *Service Water System Problems Affecting Safety-Related Equipment*, U.S. Nuclear Regulatory Commission, April 4, 1990.

NRC Information Notice 81-21, *Potential Loss of Direct Access to Ultimate Heat Sink*, U.S. Nuclear Regulatory Commission, July 21, 1981.

NRC Information Notice 85-24, *Failures of Protective Coatings in Pipes and Heat Exchangers*, U.S. Nuclear Regulatory Commission, March 26, 1985.

NRC Information Notice 85-30, *Microbiologically Induced Corrosion of Containment Service Water System*, U.S. Nuclear Regulatory Commission, April 19, 1985.

NRC Information Notice 86-96, *Heat Exchanger Fouling Can Cause Inadequate Operability of Service Water Systems*, U.S. Nuclear Regulatory Commission, November 20, 1986.

XI.M21 CLOSED-CYCLE COOLING WATER SYSTEM

Program Description

The program includes (a) preventive measures to minimize corrosion and (b) surveillance testing and inspection to monitor the effects of corrosion on the intended function of the component. The program relies on maintenance of system corrosion inhibitor concentrations within specified limits of Electric Power Research Institute [EPRI] TR-107396 to minimize corrosion. Surveillance testing and inspection in accordance with standards in EPRI TR-107396 for closed-cycle cooling water (CCCW) systems is performed to evaluate system and component performance. These measures will ensure that the CCCW system and components serviced by the CCCW system are performing their functions acceptably.

Evaluation and Technical Basis

1. ***Scope of Program:*** A CCCW system is defined as part of the service water system that is not subject to significant sources of contamination, in which water chemistry is controlled and in which heat is not directly rejected to a heat sink. The program described in this section applies only to such a system. If one or more of these conditions are not satisfied, the system is to be considered an open-cycle cooling water system. The staff notes that if the adequacy of cooling water chemistry control can not be confirmed, the system is treated as an open-cycle system as indicated in Action III of Generic Letter (GL) 89-13.

2. ***Preventive Actions:*** The program relies on the use of appropriate materials, lining, or coating to protect the underlying metal surfaces and maintenance of system corrosion inhibitor concentrations within specified limits of EPRI TR-107396 to minimize corrosion. The program includes monitoring and control of cooling water chemistry to minimize exposure to aggressive environments and application of corrosion inhibitor in the CCCW system to mitigate general, crevice, and pitting corrosion.

3. ***Parameters Monitored/Inspected:*** The aging management program (AMP) monitors the effects of corrosion by surveillance testing and inspection in accordance with standards in EPRI TR-107396 to evaluate system and component performance. For pumps, the parameters monitored include flow and discharge and suction pressures. For heat exchangers, the parameters monitored include flow, inlet and outlet temperatures, and differential pressure.

4. ***Detection of Aging Effects:*** Control of water chemistry does not preclude corrosion at locations of stagnant flow conditions or crevices. Degradation of a component due to corrosion would result in degradation of system or component performance. The extent and schedule of inspections and testing in accordance with EPRI TR-107396, assure detection of corrosion before the loss of intended function of the component. Performance and functional testing in accordance with EPRI TR-107396, ensures acceptable functioning of the CCCW system or components serviced by the CCCW system. For systems and components in continuous operation, performance adequacy is determined by monitoring data trends for evaluation of heat transfer fouling, pump wear characteristics, and branch flow changes. Components not in operation are periodically tested to ensure operability.

5. ***Monitoring and Trending:*** The frequency of sampling water chemistry varies and can occur on a continuous, daily, weekly, or as needed basis, as indicated by plant operating conditions. Per EPRI TR-107396, performance and functional tests are performed at least every 18 months to demonstrate system operability, and tests to evaluate heat removal

capability of the system and degradation of system components are performed every five years. The testing intervals may be adjusted on the basis of the results of the reliability analysis, type of service, frequency of operation, or age of components and systems.

6. ***Acceptance Criteria:*** Corrosion inhibitor concentrations are maintained within the limits specified in the EPRI water chemistry guidelines for CCCW. System and component performance test results are evaluated in accordance with the guidelines of EPRI TR-107396. Acceptance criteria and tolerances are also based on system design parameters and functions.

7. ***Corrective Actions:*** Corrosion inhibitor concentrations outside the allowable limits are returned to the acceptable range within the time period specified in the EPRI water chemistry guidelines for CCCW. If the system or component fails to perform adequately, corrective actions are taken in accordance with EPRI TR-107396. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable in addressing corrective actions.

8. ***Confirmation Process:*** Site quality assurance (QA) procedures, review and approval processes, and administrative controls are implemented in accordance with the requirements of 10 CFR Part 50, Appendix B. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable in addressing the confirmation process and administrative controls.

9. ***Administrative Controls:*** See Item 8, above.

10. ***Operating Experience:*** Degradation of closed-cycle cooling water systems due to corrosion product buildup (NRC Licensee Event Report [LER] 93-029-00) or through-wall cracks in supply lines (NRC LER 91-019-00) has been observed in operating plants. Accordingly, operating experience demonstrates the need for this program.

References

EPRI TR-107396, *Closed Cooling Water Chemistry Guidelines*, Electric Power Research Institute, Palo Alto, CA, November 1997.

NRC Generic Letter 89-13, *Service Water System Problems Affecting Safety-Related Equipment*, U.S. Nuclear Regulatory Commission, July 18, 1989.

NRC Generic Letter 89-13, Supplement 1, *Service Water System Problems Affecting Safety-Related Equipment*, U.S. Nuclear Regulatory Commission, April 4, 1990.

NRC Licensee Event Report LER #91-019-00, *Loss of Containment Integrity due to Crack in Component Cooling Water Piping*, October 26, 1991.

NRC Licensee Event Report LER #93-029-00, *Inoperable Check Valve in the Component Cooling System as a Result of a Build-Up of Corrosion Products between Valve Components*, December 13, 1993.

XI.M22 BORAFLEX MONITORING

Program Description

A Boraflex monitoring program for the actual Boraflex panels is implemented in the spent fuel racks to assure that no unexpected degradation of the Boraflex material would compromise the criticality analysis in support of the design of spent fuel storage racks. The applicable aging management program (AMP), based on manufacturer's recommendations, relies on periodic inspection, testing, monitoring, and analysis of the criticality design to assure that the required 5% subcriticality margin is maintained. The frequency of the inspection and testing depends on the condition of the Boraflex, with a maximum of five years. Certain accelerated samples are tested every two years. Results based on test coupons have been found to be unreliable in determining the degree to which the actual Boraflex panels have been degraded. Therefore, this AMP includes: (1) performing neutron attenuation testing, called blackness testing, to determine gap formation in Boraflex panels; (2) completing sampling and analysis for silica levels in the spent fuel pool water and trending the results by using the EPRI RACKLIFE predictive code or its equivalent on a monthly, quarterly, or annual basis (depending on Boraflex panel condition); and (3) measuring boron areal density by techniques such as the BADGER device. Corrective actions are initiated if the test results find that the 5% subcriticality margin cannot be maintained because of current or projected future Boraflex degradation.

Evaluation and Technical Basis

1. ***Scope of Program****:* The AMP manages the effects of aging on sheets of neutron-absorbing materials affixed to spent fuel racks. For Boraflex panels, gamma irradiation and long-term exposure to the wet pool environment cause shrinkage resulting in gap formation, gradual degradation of the polymer matrix, and the release of silica to the spent fuel storage pool water. This results in the loss of boron carbide in the neutron absorber sheets.

2. ***Preventive Actions:*** For Boraflex panels, monitoring silica levels in the storage pool water, measuring gap formation by blackness testing, periodically measuring boron areal density, and applying predictive codes, are performed. These actions ensure that degradation of the neutron-absorbing material is identified and corrected so the spent fuel storage racks will be capable of performing their intended functions during the period of extended operation, consistent with current licensing basis (CLB) design conditions.

3. ***Parameters Monitored/Inspected:*** The parameters monitored include physical conditions of the Boraflex panels, such as gap formation and decreased boron areal density, and the concentration of the silica in the spent fuel pool. These are conditions directly related to degradation of the Boraflex material. When Boraflex is subjected to gamma radiation and long-term exposure to the spent fuel pool environment, the silicon polymer matrix becomes degraded and silica filler and boron carbide are released into the spent fuel pool water. As indicated in the Nuclear Regulatory Commission (NRC) Information Notice (IN) 95-38 and NRC Generic Letter (GL) 96-04, the loss of boron carbide (washout) from Boraflex is characterized by slow dissolution of silica from the surface of the Boraflex and a gradual thinning of the material. Because Boraflex contains about 25% silica, 25% polydimethyl siloxane polymer, and 50% boron carbide, sampling and analysis of the presence of silica in the spent fuel pool provide an indication of depletion of boron carbide from Boraflex; however, the degree to which Boraflex has degraded is ascertained through measurement of the boron areal density.

4. ***Detection of Aging Effects:*** The amount of boron carbide released from the Boraflex panel is determined through direct measurement of boron areal density and correlated with the levels of silica present through the use of a predictive code. This is supplemented with detection of gaps through blackness testing and periodic verification of boron loss through areal density measurement techniques such as the BADGER device.

5. ***Monitoring and Trending:*** The periodic inspection measurements and analysis are to be compared to values of previous measurements and analysis to provide a continuing level of data for trend analysis.

6. ***Acceptance Criteria:*** The 5% subcriticality margin of the spent fuel racks is to be maintained for the period of extended operation.

7. ***Corrective Actions:*** Corrective actions are initiated if the test results find that the 5% subcriticality margin cannot be maintained because of the current or projected future degradation. Corrective actions consist of providing additional neutron-absorbing capacity by Boral or boron steel inserts, or other options which are available to maintain a subcriticality margin of 5%. As discussed in the appendix of this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable in addressing corrective actions.

8. ***Confirmation Process:*** Site quality assurance (QA) procedures, site review and approval processes, and administrative controls are implemented in accordance with the requirements of 10 CFR Part 50, Appendix B. As discussed in the appendix of this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable in addressing the confirmation process and administrative controls.

9. ***Administrative Controls:*** See item 8, above.

10. ***Operating Experience:*** The NRC IN 87-43 addresses the problems of development of tears and gaps (average 1-2 in., with the largest 4 in.) in Boraflex sheets due to gamma radiation-induced shrinkage of the material. The NRC IN 93-70 and 95-38 and NRC GL 96-04 address several cases of significant degradation of Boraflex test coupons due to accelerated dissolution of Boraflex caused by pool water flow through panel enclosures and high accumulated gamma dose. Two spent fuel rack cells with about 12 years of service have only 40% of the Boraflex remaining. In such cases, the Boraflex may be replaced by boron steel inserts or by a completely new rack system using Boral. Experience with boron steel is limited; however, the application of Boral for use in the spent fuel storage racks predates the manufacturing and use of Boraflex. The experience with Boraflex panels indicates that coupon surveillance programs are not reliable, therefore, measurement of boron areal density correlated, through a predictive code, with silica levels in the pool water and verified periodically, is performed during the period of extended operation. These monitoring programs provide assurance that degradation of Boraflex sheets is monitored, so that appropriate actions can be taken in a timely manner if significant loss of neutron-absorbing capability is occurring. These monitoring programs ensure that the Boraflex sheets will maintain their integrity and will be effective in performing its intended function.

References

BNL-NUREG-25582, *Corrosion Considerations in the Use of Boral in Spent Fuel Storage Pool Racks*, January 1979.

EPRI NP-6159, *An Assessment of Boraflex Performance in Spent Nuclear Fuel Storage Racks*, Electric Power Research Institute, Palo Alto, CA, December 14, 1988.

EPRI TR-101986, *Boraflex Test Results and Evaluation*, Electric Power Research Institute, Palo Alto, CA, March 1, 1993.

EPRI TR-103300, *Guidelines for Boraflex Use in Spent-Fuel Storage Racks*, Electric Power Research Institute, Palo Alto, CA, December 1, 1993.

NRC Generic Letter 96-04, *Boraflex Degradation in Spent Fuel Pool Storage Racks*, U.S. Nuclear Regulatory Commission, June 26, 1996.

NRC Information Notice 87-43, *Gaps in Neutron Absorbing Material in High Density Spent Fuel Storage Racks,* U.S. Nuclear Regulatory Commission, September 8, 1987.

NRC Information Notice 93-70, *Degradation of Boraflex Neutron Absorber Coupons*, U.S. Nuclear Regulatory Commission, September 10, 1993.

NRC Information Notice 95-38, *Degradation of Boraflex Neutron Absorber in Spent Fuel Storage Racks*, U.S. Nuclear Regulatory Commission, September 8, 1995.

NRC Regulatory Guide 1.26, Rev. 3, *Quality Group Classifications and Standards for Water, Steam, and Radioactive-Waste-Containing Components of Nuclear Power Plants (for Comment),* U.S. Nuclear Regulatory Commission, February 1976.

XI.M23 INSPECTION OF OVERHEAD HEAVY LOAD AND LIGHT LOAD (RELATED TO REFUELING) HANDLING SYSTEMS

Program Description

Most commercial nuclear facilities have between 50 and 100 cranes. Many are industrial grade cranes which meet the requirements of 29 CFR Volume XVII, Part 1910, and Section 1910.179. Most are not within the scope of 10 CFR 54.4, and therefore are not required to be part of the integrated plant assessment (IPA).

Normally, fewer than 10 cranes fall within the scope of 10 CFR 54.4. These cranes comply with the Maintenance Rule requirements provided in 10 CFR 50.65. The Nuclear Regulatory Commission Regulatory Guide (RG) 1.160 provides guidance for monitoring the effectiveness of maintenance at nuclear power plants.

The program demonstrates that testing and monitoring programs have been implemented and have ensured that the structures, systems, and components of these cranes are capable of sustaining their rated loads. This is their intended function during the period of extended operation. It is noted that many of the systems and components of these cranes perform an intended function with moving parts or with a change in configuration, or subject to replacement based on qualified life. In these instances, these types of crane systems and components are not within the scope of this aging management program (AMP). This program is primarily concerned with structural components that make up the bridge and trolley. NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," provides specific guidance on the control of overhead heavy load cranes.

Evaluation and Technical Basis

1. ***Scope of Program:*** The program manages the effects of general corrosion on the crane and trolley structural components for those cranes that are within the scope of 10 CFR 54.4, and the effects of wear on the rails in the rail system.

2. ***Preventive Actions:*** No preventive actions are identified. The crane program is an inspection program.

3. ***Parameters Monitored/Inspected:*** The program evaluates the effectiveness of the maintenance monitoring program and the effects of past and future usage on the structural reliability of cranes. The number and magnitude of lifts made by the crane are also reviewed.

4. ***Detection of Aging Effect:*** Crane rails and structural components are visually inspected on a routine basis for degradation. Functional tests are also performed to assure their integrity.

5. ***Monitoring and Trending:*** Monitoring and trending are not required as part of the crane inspection program.

6. ***Acceptance Criteria:*** Any significant visual indication of loss of material due to corrosion or wear are evaluated according to applicable industry standards and good industry practice. The crane may also have been designed to a specific Service Class as defined in the EOCI Specification #61 (or later revisions), or CMAA Specification #70 (or later revisions), or CMAA Specification #74 (or later revisions). The specification that was applicable at the time the crane was manufactured is used.

7. ***Corrective Actions:*** Site corrective actions program, quality assurance (QA) procedures, site review and approval process, and administrative controls are implemented in accordance with the requirements of 10 CFR Part 50, Appendix B. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable in addressing the corrective actions, confirmation process, and administrative controls.

8. ***Confirmation Process:*** See Item 7, above.

9. ***Administrative Controls:*** See Item 7, above.

10. ***Operating Experience:*** Because of the requirements for monitoring the effectiveness of maintenance at nuclear power plants provided in 10 CFR 50.65, there has been no history of corrosion-related degradation that has impaired cranes. Likewise, because cranes have not been operated beyond their design lifetime, there have been no significant fatigue-related structural failures.

References

10 CFR 50.65, *Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants*, Office of the Federal Register, National Archives and Records Administration, January 1997.

Crane Manufactures Association of America, Inc., CMAA Specification No. 70, *Specifications for Electric Overhead Traveling Cranes*, 1970 (or later revisions).

Crane Manufactures Association of America, Inc., CMAA Specification No. 74, *Specifications for Top Running and Under Running Single Girder Electric Overhead Traveling Cranes*, 1974 (or later revisions).

Electric Overhead Crane Institute, Inc., EOCI Specification No. 61, *Specifications for Electric Overhead Traveling Cranes*, 1961 (or later revisions).

NUREG-0612, *Control of Heavy Loads at Nuclear Power Plants*, U.S. Nuclear Regulatory Commission, 1980.

NRC Regulatory Guide 1.160, Rev. 2, *Monitoring the Effectiveness of Maintenance at Nuclear Power Plants*, U.S. Nuclear Regulatory Commission, March 1997.

XI.M24 COMPRESSED AIR MONITORING

Program Description

The program consists of inspection, monitoring, and testing of the entire system, including (a) frequent leak testing of valves, piping, and other system components, especially those made of carbon steel; and (b) preventive monitoring that checks air quality at various locations in the system to ensure that oil, water, rust, dirt, and other contaminants are kept within the specified limits. The aging management program (AMP) provides for timely corrective actions to ensure that the system is operating within specified limits.

The AMP is based on results of the plant owners response to the Nuclear Regulatory Commission (NRC) Generic Letter (GL) 88-14, augmented by previous NRC Information Notices IN 81-38, IN 87-28, and IN 87-28 S1, and by the Institute of Nuclear Power Operations Significant Operating Experience Report (INPO SOER) 88-01. The NRC GL 88-14, issued after several years of study of problems and failures of instrument air systems, recommends each holder of an operating license to perform an extensive design and operations review and verification of its instrument air system. The GL 88-14 also recommends the licensees to describe their program for maintaining proper instrument air quality. The AMP also incorporates provisions conforming to the guidance of the Electric Power Research Institute (EPRI) NP-7079, issued in 1990, to assist utilities in identifying and correcting system problems in the instrument air system and to enable them to maintain required industry safety standards. Subsequent to these initial actions by all plant licensees to implement an improved AMP, some utilities decided to replace their instrument air system with newer models and types of equipment. The EPRI then issued TR-108147, which addresses maintenance of the latest compressors and other instrument air system equipment currently in use at those plants. The American Society of Mechanical Engineers operations and maintenance standards and guides (ASME OM-S/G-1998, Part 17) provides additional guidance to the maintenance of the instrument air system by offering recommended test methods, test intervals, parameters to be measured and evaluated, acceptance criteria, corrective actions, and records requirements.

Evaluation and Technical Basis

1. ***Scope of Program:*** The program manages the effects of corrosion and the presence of unacceptable levels of contaminants on the intended function of the compressed air system. The AMP includes frequent leak testing of valves, piping, and other system components, especially those made of carbon steel, and a preventive maintenance program to check air quality at several locations in the system.

2. ***Preventive Actions:*** The system air quality is monitored and maintained in accordance with the plant owner’s testing and inspection plans, which are designed to ensure that the system and equipment meet specified operability requirements. These requirements are prepared from consideration of manufacturer's recommendations for individual components and guidelines based on ASME OM-S/G-1998, Part 17; ISA-S7.0.01-1996; EPRI NP-7079; and EPRI TR-108147. The preventive maintenance program addresses various aspects of the inoperability of air-operated components due to corrosion and the presence of oil, water, rust, and other contaminants.

3. ***Parameters Monitored/Inspected:*** Inservice inspection (ISI) and testing is performed to verify proper air quality and confirm that maintenance practices, emergency procedures, and training are adequate to ensure that the intended function of the air system is maintained.

4. ***Detection of Aging Effects:*** Guidelines in EPRI NP-7079, EPRI TR-108147, and ASME OM-S/G-1998, Part 17, ensure timely detection of degradation of the compressed air system function. Degradation of the piping and any equipment would become evident by observation of excessive corrosion, by the discovery of unacceptable leakage rates, and by failure of the system or any item of equipment to meet specified performance limits.

5. ***Monitoring and Trending:*** Effects of corrosion and the presence of contaminants are monitored by visual inspection and periodic system and component tests, including leak rate tests on the system and on individual items of equipment. These tests verify proper operation by comparing measured values of performance with specified performance limits. Test data are analyzed and compared to data from previous tests to provide for timely detection of aging effects.

6. ***Acceptance Criteria:*** Acceptance criteria is established for the system and for individual equipment that contain specific limits or acceptance ranges based on design basis conditions and/or equipment vendor specifications. The testing results are analyzed to verify that the design and performance of the system is in accordance with its intended function.

7. ***Corrective Actions:*** Corrective actions are taken if any parameters are out of acceptable ranges, such as moisture content in the system air. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable in addressing corrective actions.

8. ***Confirmation Process:*** The site corrective actions program, quality assurance (QA) procedures, site review and approval process, and administrative controls are implemented in accordance with the requirements of 10 CFR Part 50, Appendix B. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable in addressing the confirmation process and administrative controls.

9. ***Administrative Controls:*** See Item 8, above.

10. ***Operating Experience:*** Potentially significant safety-related problems pertaining to air systems have been documented in NRC IN 81-38, IN 87-28, IN 87-28 S1 and license event report (LER) 50-237/94-005-3. Some of the systems that have been significantly degraded or have failed due to the problems in the air system include the decay heat removal, auxiliary feedwater, main steam isolation, containment isolation, and fuel pool seal system. As a result of NRC GL 88-14 and consideration of INPO SOER 88-01, EPRI NP-7079, and EPRI TR-108147, performance of air systems has improved significantly.

References

ASME OM-S/G-1998, Part 17, *Performance Testing of Instrument Air Systems Information Notice Light-Water Reactor Power Plants*, 1ISA-S7.0.1-1996, “Quality Standard for Instrument Air,” American Society of Mechanical Engineers, New York, NY, 1998.

EPRI NP-7079, *Instrument Air System: A Guide for Power Plant Maintenance Personnel*, Electric Power Research Institute, Palo Alto, CA., December 1990.

EPRI/NMAC TR-108147, *Compressor and Instrument Air System Maintenance Guide: Revision to NP-7079*, Electric Power Research Institute, Palo Alto, CA., March 1998.

INPO SOER 88-01, *Instrument Air System Failures*, May 18, 1988.

NRC Generic Letter 88-14, *Instrument Air Supply Problems Affecting Safety-Related Equipment*, U.S. Nuclear Regulatory Commission, August 8, 1988.

NRC Information Notice 81-38, *Potentially Significant Equipment Failures Resulting from Contamination of Air-Operated Systems*, U.S. Nuclear Regulatory Commission, December 17, 1981.

NRC Information Notice 87-28, *Air Systems Problems at U.S. Light Water Reactors*, U.S. Nuclear Regulatory Commission, June 22, 1987.

NRC Information Notice 87-28, Supplement 1, *Air Systems Problems at U.S. Light Water Reactors*, U.S. Nuclear Regulatory Commission, December 28, 1987.

NRC Licensee Event Report LER 50-237/94-005-3, *Manual Reactor Scram due to Loss of Instrument Air Resulting from Air Receiver Pipe Failure Caused by Improper Installation of Threaded Pipe during Initial Construction*, U.S. Nuclear Regulatory Commission, April 23, 1997.

XI.M25 BWR REACTOR WATER CLEANUP SYSTEM

Program Description

The program includes inservice inspection (ISI) and monitoring and control of reactor coolant water chemistry to manage the effects of stress corrosion cracking (SCC) or intergranular stress corrosion cracking (IGSCC) on the intended function of austenitic stainless steel (SS) piping in the reactor water cleanup (RWCU) system. Based on the Nuclear Regulatory Commission (NRC) criteria related to inspection guidelines for RWCU piping welds outboard of the second isolation valve, the program includes the measures delineated in NUREG-0313, Rev. 2, and NRC Generic Letter (GL) 88-01. Coolant water chemistry is monitored and maintained in accordance with the Electric Power Research Institute (EPRI) guidelines in boiling water reactor vessel and internals project (BWRVIP)-29 (TR-103515) to minimize the potential of crack initiation and growth due to SCC or IGSCC.

Evaluation and Technical Basis

1. ***Scope of Program:*** Based on the NRC letter (September 15, 1995) on the screening criteria related to inspection guidelines for RWCU piping welds outboard of the second isolation valve, the program includes the measures delineated in NUREG-0313, Rev. 2, and NRC GL 88-01 to monitor SCC or IGSCC and its effects on the intended function of austenitic SS piping. The screening criteria include:

   a. Satisfactory completion of all actions requested in NRC GL 89-10,

   b. No detection of IGSCC in RWCU welds inboard of the second isolation valves (ongoing inspection in accordance with the guidance in NRC GL 88-01), and

   c. No detection of IGSCC in RWCU welds outboard of the second isolation valves after inspecting a minimum of 10% of the susceptible piping.

   No IGSCC inspection is recommended for plants that meet all three criteria or that meet criterion (a) and piping is made of material that is resistant to IGSCC.

2. ***Preventive Actions:*** The comprehensive program outlined in NUREG-0313 and NRC GL 88-01 addresses improvements in all three elements that, in combination, cause SCC or IGSCC. These elements are a susceptible (sensitized) material, a significant tensile stress, and an aggressive environment. The program delineated in NUREG-0313 and NRC GL 88-01 includes recommendations regarding selection of materials that are resistant to sensitization, use of special processes that reduce residual tensile stresses, and monitoring and maintenance of coolant chemistry. The resistant materials are used for new and replacement components and include low-carbon grades of austenitic SS and weld metal, with a maximum carbon of 0.035 wt.% and a minimum ferrite of 7.5% in weld metal and cast austenitic stainless steel (CASS). Inconel 82 is the only commonly used nickel-base weld metal considered to be resistant to SCC; other nickel-alloys, such as Alloy 600, are evaluated on an individual basis. Special processes are used for existing as well as new and replacement components. These processes include solution heat treatment, heat sink welding, induction heating, and mechanical stress improvement.

   The program delineated in NUREG-0313 and NRC GL 88-01 varies depending on the plant-specific reactor water chemistry to mitigate SCC or IGSCC.

3. ***Parameters Monitored/Inspected:*** The aging management program (AMP) monitors SCC or IGSCC of austenitic SS piping by detection and sizing of cracks by implementing the inspection guidelines delineated in the NRC screening criteria for the RWCU piping outboard of isolation valves. The following schedules are followed:

   Schedule A: No inspection is required for plants that meet all three criteria set forth above, or if they meet only criterion (a). Piping is made of material that is resistant to IGSCC, as described above in preventive actions.

   Schedule B: For plants that meet only criterion (a): Inspect at least 2% of the welds or two welds every refueling outage, whichever sample is larger.

   Schedule C: For plants that do not meet criterion (a): Inspect at least 10% of the welds every refueling outage.

4. ***Detection of Aging Effects:*** The extent, method, and schedule of the inspection and test techniques delineated in the NRC inspection criteria for RWCU piping and NRC GL 88-01 are designed to maintain structural integrity and to detect aging effects before the loss of intended function of austenitic SS piping and fittings. Guidelines for the inspection schedule, methods, personnel, sample expansion, and leak detection guidelines are based on the guidelines of NRC GL 88-01.

   The NRC GL 88-01 recommends that the detailed inspection procedure, equipment, and examination personnel be qualified by a formal program approved by the NRC. Inspection can reveal crack initiation and growth and leakage of coolant. The extent and frequency of inspections recommended by the program are based on the condition of each weld (e.g., whether the weldments were made from IGSCC-resistant material, whether a stress improvement process was applied to a weldment to reduce the residual stresses, and how the weld was repaired if it had been cracked).

5. ***Monitoring and Trending:*** The extent and schedule for inspection in accordance with the recommendations of NRC GL 88-01 provide timely detection of cracks and leakage of coolant. Based on inspection results, NRC GL 88-01 provides guidelines for additional samples of welds to be inspected when one or more cracked welds are found in a weld category.

6. ***Acceptance Criteria:*** The NRC GL 88-01 recommends that any indication detected be evaluated in accordance with the requirements of ASME Section XI, Subsection IWB-3640 (1995 edition through the 1996 addenda).

7. ***Corrective Actions*****:** The guidance for weld overlay repair, stress improvement, or replacement is provided in NRC GL 88-01. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable in addressing corrective actions.

8. ***Confirmation Process:*** Site quality assurance (QA) procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of 10 CFR Part 50, Appendix B. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable in addressing the confirmation process and administrative controls.

9. ***Administrative Controls:*** See Item 8, above.

10. ***Operating Experience:*** The IGSCC has occurred in small- and large-diameter boiling water reactor (BWR) piping made of austenitic SSs or nickel alloys. The comprehensive program outlined in NRC GL 88-01 and NUREG-0313 addresses improvements in all elements that cause SCC or IGSCC (e.g., susceptible material, significant tensile stress, and an aggressive environment) and is effective in managing IGSCC in austenitic SS piping in the RWCU system.

References

ASME Section XI, *Rules for Inservice Inspection of Nuclear Power Plant Components*, ASME Boiler and Pressure Vessel Code, 1995 edition through the 1996 addenda, American Society of Mechanical Engineers, New York, NY.

BWRVIP-29 (EPRI TR-103515), *BWR Vessel and Internals Project*, *BWR Water Chemistry Guidelines-1993 Revision, Normal and Hydrogen Water Chemistry,* Electric Power Research Institute, Palo Alto, CA, February 1994.

Letter from Joseph W. Shea, U.S. Nuclear Regulatory Commission, to George A. Hunter, Jr., PECO Energy Company, *Reactor Water Cleanup (RWCU) System Weld Inspections at Peach Bottom Atomic Power Station, Units 2 and 3 (TAC Nos. M92442 and M92443),* September 15, 1995.

NRC Generic Letter 88-01, *NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping,* U.S. Nuclear Regulatory Commission, January 25, 1988.

NUREG-0313, Rev. 2, *Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping,* W. S. Hazelton and W. H. Koo, U.S. Nuclear Regulatory Commission, 1988.

XI.M26 FIRE PROTECTION

Program Description

For operating plants, the fire protection aging management program (AMP) includes a fire barrier inspection program and a diesel-driven fire pump inspection program. The fire barrier inspection program requires periodic visual inspection of fire barrier penetration seals, fire barrier walls, ceilings, and floors, and periodic visual inspection and functional tests of fire rated doors to ensure that their operability is maintained. The diesel-driven fire pump inspection program requires that the pump be periodically tested to ensure that the fuel supply line can perform the intended function. The AMP also includes periodic inspection and test of halon/carbon dioxide fire suppression system.

Evaluation and Technical Basis

1. ***Scope of Program:*** For operating plants, the AMP manages the aging effects on the intended function of the penetration seals, fire barrier walls, ceilings, and floors, and all fire rated doors (automatic or manual) that perform a fire barrier function. It also manages the aging effects on the intended function of the fuel supply line. The AMP also includes management of the aging effects on the intended function of the halon/carbon dioxide fire suppression system.

2. ***Preventive Actions:*** For operating plants, the fire hazard analysis assesses the fire potential and fire hazard in all plant areas. It also specifies measures for fire prevention, fire detection, fire suppression, and fire containment and alternative shutdown capability for each fire area containing structures, systems, and components important to safety.

3. ***Parameters Monitored/Inspected:*** Visual inspection of 10% of each type of penetration seal is performed during walkdowns carried out at least once every refueling outage. These inspections examine any sign of degradation such as cracking, seal separation from walls and components, separation of layers of material, rupture and puncture of seals which are directly caused by increased hardness and shrinkage of seal material due to weathering. Visual inspection of the fire barrier walls, ceilings, and floors examines any sign of degradation such as cracking, spalling, and loss of material caused by freeze-thaw, chemical attack, and reaction with aggregates. Hollow metal fire doors are visually inspected at least once bi-monthly for holes in the skin of the door. Fire door clearances are also checked at least once bi-monthly as part of an inspection program. Function tests of fire doors are performed daily, weekly, or monthly (which maybe plant specific) to verify the operability of automatic hold-open, release, closing mechanisms, and latches.

   The diesel-driven fire pump is under observation during performance tests such as flow and discharge tests, sequential starting capability tests, and controller function tests for detecting any degradation of the fuel supply line.

   Periodic visual inspection and function test at least once every six months examines the signs of degradation of the halon/carbon dioxide fire suppression system. The suppression agent charge pressure is monitored in the test. Material conditions that may affect the performance of the system, such as corrosion, mechanical damage, or damage to dampers, are observed during these tests. Inspections performed at least once every month verify that the extinguishing agent supply valves are open and the system is in automatic mode.

4. ***Detection of Aging Effects:*** Visual inspection of penetration seals detects cracking, seal separation from walls and components, and rupture and puncture of seals. Visual inspection (VT-1 or equivalent) of 10% of each type of seal in walkdowns is performed at least once every refueling outage. If any sign of degradation is detected within that 10%, the scope of the inspection and frequency is expanded to ensure timely detection of increased hardness and shrinkage of the penetration seal before the loss of the component intended function. Visual inspection (VT-1 or equivalent) of the fire barrier walls, ceilings, and floors performed in walkdown at least once every refueling outage ensures timely detection for concrete cracking, spalling, and loss of material. Visual inspection (VT-3 or equivalent) detects any sign of degradation of the fire door such as wear and missing parts. Function tests promptly detect deficiencies in operational conditions. Periodic visual inspection and function tests detect degradation of the fire doors before there is a loss of intended function.

   Periodic tests performed at least once every refueling outage, such as flow and discharge tests, sequential starting capability tests, and controller function tests performed on diesel-driven fire pump ensure fuel supply line performance. The performance tests detect degradation of the fuel supply lines before the loss of the component intended function.

   In the test of the halon/carbon dioxide fire suppression system, the suppression agent charge pressure is verified to be within in the normal band. Visual inspection detects any sign of degradation, such as corrosion, mechanical damage, or damage to dampers. The periodic function test and inspection performed at least once every six months detects degradation of the halon/carbon dioxide fire suppression system before the loss of the component intended function. The monthly inspection ensures that the extinguishing agent supply valves are open and the system is in automatic mode.

5. ***Monitoring and Trending:*** The aging effects of weathering on fire barrier penetration seals are detectable by visual inspection and, based on operating experience, visual inspections performed at least once every refueling outage to detect any sign of degradation of fire barrier penetration seals prior to loss of the intended function.

   Concrete cracking, spalling, and loss of material are detectable by visual inspection and, based on operating experience, visual inspection performed at least once every refueling outage detects any sign of degradation of the fire barrier walls, ceilings, and floors before there is a loss of the intended function. Wear, missing parts, or holes in the fire door are detectable by visual inspection and, based on operating experience, the visual inspection and function test performed bi-monthly which detects degradation of the fire doors prior to loss of the intended function.

   The performance of the fire pump is monitored during the periodic test to detect any degradation in the fuel supply lines. Periodic testing provides data (e.g., pressure) for trending necessary.

   The performance of the halon/carbon dioxide fire suppression system is monitored during the periodic test to detect any degradation in the system. These periodic tests provide data necessary for trending.

6. ***Acceptance Criteria:*** Inspection results are acceptable if there are no visual indications of cracking, separation of seals from walls and components, separation of layers of material, or ruptures or punctures of seals, no visual indications of concrete cracking, spalling and loss of material of fire barrier walls, ceilings, and floors, no visual indications of missing parts, holes, and wear and no deficiencies in the functional tests of fire doors. No corrosion

is acceptable in the fuel supply line for the diesel-driven fire pump. Also, any signs of corrosion and mechanical damage of the halon/carbon dioxide fire suppression system are not acceptable.

7. ***Corrective Actions:*** For fire protection structures and components identified within scope that are subject to an aging management review for license renewal, the applicant is to expand the scope of the 10 CFR Part 50, Appendix B, program to include these in-scope structures and components to address corrective actions, confirmation process, and administrative controls for aging management during the period of extended operation. This commitment is documented in the final safety analysis report (FSAR) supplement in accordance with 10 CFR 54.21(d). As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address corrective actions, confirmation process, and administrative controls.

8. ***Confirmation Process:*** See Item 7, above.

9. ***Administrative Controls:*** See Item 7, above.

10. ***Operating Experience:*** Silicone foam fire barrier penetration seals have experienced splits, shrinkage, voids, lack of fill, and other failure modes (IN 88-56, IN 94-28, and IN 97-70). Degradation of electrical racing way fire barrier such as small holes, cracking, and unfilled seals are found on routine walkdown (IN 91-47 and GL 92-08). Fire doors have experienced wear of the hinges and handles. Operating experience with the use of this AMP has shown that no corrosion-related problem has been reported for the fuel supply line, pump casing of the diesel-driven fire pump, and the halon/carbon dioxide suppression system. No significant aging related problems have been reported of fire protection systems, emergency breathing and auxiliary equipment, and communication equipment.

References

NRC Generic Letter 92-08, *Thermo-Lag 330-1 Fire Barrier*, December 17, 1992.

NRC Information Notice 88-56, *Potential Problems with Silicone Foam Fire Barrier Penetration Seals*, August 14, 1988.

NRC Information Notice 91-47, *Failure of Thermo-Lag Fire Barrier Material to Pass Fire Endurance Test*, August 6, 1991.

NRC Information Notice 94-28, *Potential problems with Fire-Barrier Penetration Seals*, April 5, 1994.

NRC Information Notice 97-70, *Potential problems with Fire Barrier Penetration Seals*, September 19, 1997.

XI.M27 FIRE WATER SYSTEM

Program Description

This aging management program applies to water-based fire protection systems that consist of sprinklers, nozzles, fittings, valves, hydrants, hose stations, standpipes, water storage tanks, and aboveground and underground piping and components that are tested in accordance with the applicable National Fire Protection Association (NFPA) codes and standards. Such testing assures the minimum functionality of the systems. Also, these systems are normally maintained at required operating pressure and monitored such that loss of system pressure is immediately detected and corrective actions initiated. In addition to NFPA codes and standards, which do not currently contain programs to manage aging, portions of the fire protection sprinkler system, which are not routinely subjected to flow, are to be subjected to full flow tests at the maximum design flow and pressure before the period of extended operation (and at not more than 5-year intervals thereafter). In addition, a sample of sprinkler heads is to be inspected by using the guidance of NFPA 25, Section 2.3.3.1. This NFPA section states that "where sprinklers have been in place for 50 years, they shall be replaced or representative samples from one or more sample areas shall be submitted to a recognized testing laboratory for field service testing." It also contains guidance to perform this sampling every 10 years after the initial field service testing. Finally, portions of fire protection suppression piping located aboveground and exposed to water are disassembled and visually inspected internally once every refueling outage. The purpose of the full flow testing and internal visual inspections is to ensure that corrosion, microbiological influenced corrosion (MIC), or biofouling aging effects are managed such that the system function is maintained.

Evaluation and Technical Basis

1. ***Scope of Program:*** The aging management program focuses on managing loss of material due to corrosion, MIC, or biofouling of carbon steel and cast-iron components in fire protection systems exposed to water. Hose station and standpipe are considered as piping in the AMP.

2. ***Preventive Actions:*** To ensure no significant corrosion, MIC, or biofouling has occurred in water-based fire protection systems, periodic flushing, system performance testing, and inspections are conducted.

3. ***Parameters Monitored/Inspected:*** Loss of material due to corrosion and biofouling could reduce wall thickness of the fire protection piping system and result in system failure. Therefore, the parameters monitored are the system's ability to maintain pressure and internal system corrosion conditions. The NRC GL 89-13 recommends periodic flow testing of infrequently used loops of the fire water system at the maximum design flow to ensure that the system maintains its intended function.

4. ***Detection of Aging Effects:*** Fire protection system testing is performed to assure required pressures. Internal inspections of aboveground fire protection piping and the smaller diameter fire suppression piping are performed on system components (when they are disassembled) to identify evidence of loss of material due to corrosion. Repair and replacement actions are initiated as necessary. Continuous system pressure monitoring, periodic system flow testing performed, and internal inspections of aboveground piping are effective means to ensure that corrosion and biofouling are not occurring and the system's intended function is maintained. In addition, general requirements of existing fire protection programs include testing and maintenance of fire detection and suppression systems and

surveillance procedures to ensure that fire detectors, as well as fire suppression systems and components, are operable.

Visual inspection of yard fire hydrants performed once every six months ensures timely detection of signs of degradation, such as corrosion. Fire hydrant hose hydrostatic tests, gasket inspections, and fire hydrant flow tests, performed annually, ensure that fire hydrants can perform their intended function and provide opportunities for degradation to be detected before a loss of intended function can occur.

Sprinkler systems are inspected once every refueling outage to ensure that signs of degradation, such as corrosion, are detected in a timely manner.

5. ***Monitoring and Trending:*** System discharge pressure is monitored continuously. Results of system performance testing are monitored and trended as specified by the NFPA codes and standards. Degradation identified by internal inspection is evaluated.

6. ***Acceptance Criteria:*** The acceptance criteria are (a) the ability of a fire protection system to maintain required pressure, (b) no unacceptable signs of degradation observed during visual assessment of internal system conditions, and (c) that no biofouling exists in the sprinkler systems that could cause corrosion in the sprinkler heads.

7. ***Corrective Actions:*** For fire water systems and components identified within scope that are subject to an aging management review for license renewal, the applicant is to expand the scope of the 10 CFR Part 50, Appendix B, program to include these in-scope systems and components to address corrective actions, confirmation process, and administrative controls for aging management during the period of extended operation. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address corrective actions, confirmation process, and administrative controls.

8. **Confirmation Process:** See Item 7, above.

9. **Administrative Controls:** See Item 7, above.

10. ***Operating Experience:*** Water-based fire protection systems designed, inspected, tested and maintained in accordance with the NFPA minimum standards have demonstrated reliable performance.

References

NFPA 25: Inspection, Testing and Maintenance of Water-Based Fire Protection Systems, 1998 Edition.

NRC Generic Letter 89-13, *Service Water System Problems Affecting Safety-Related Equipment*, July 18, 1989.

XI.M28 BURIED PIPING AND TANKS SURVEILLANCE

Program Description

The program includes surveillance and preventive measures to mitigate corrosion by protecting the external surface of buried carbon steel piping and tanks. Surveillance and preventive measures are in accordance with standard industry practice, based on NACE Standards RP-0285-95 and RP-0169-96, and include external coatings, wrappings, and cathodic protection systems.

Evaluation and Technical Basis

1. ***Scope of Program:*** The program relies on preventive measures, such as coating, wrapping, and cathodic protection, and surveillance, based on NACE Standard RP-0285-95 and NACE Standard RP-0169-96, to manage the effects of corrosion on the intended function of buried tanks and piping, respectively.

2. ***Preventive Actions:*** In accordance with industry practice, underground piping and tanks are coated during installation with a protective coating system, such as coal tar enamel with a fiberglass wrap and a kraft paper outer wrap, a polyolifin tape coating, or a fusion bonded epoxy coating to protect the piping from contacting the aggressive soil environment. A cathodic protection system is used to mitigate corrosion where pinholes in the coating allow the piping or components to be in contact with the aggressive soil environment. The cathodic protection imposes a current from an anode onto the pipe or tank to stop corrosion from occurring at defects in the coating.

3. ***Parameters Monitored/Inspected:*** The effectiveness of the coatings and cathodic protection system, per standard industry practice, is determined by measuring coating conductance, by surveying pipe-to-soil potential, and by conducting bell hole examinations to visually examine the condition of the coating.

4. ***Detection of Aging Effects:*** Coatings and wrapping can be damaged during installation or while in service and the cathodic protection system is relied upon to avoid any corrosion at the damaged locations. Degradation of the coatings and wrapping during service will result in the requirement for more current from the cathodic protection rectifier in order to maintain the proper cathodic protect potentials. Any increase in current requirements is an indication of coating and wrapping degradation. A close interval pipe-to-soil potential survey can be used to locate the locations where degradation has occurred.

5. ***Monitoring and Trending:*** Monitoring the coating conductance versus time or the current requirement versus time provide an indication of the condition of the coating and cathodic protection system when compared to predetermined values.

6. ***Acceptance Criteria:*** In accordance with accepted industry practice, per NACE Standard RP-0285-95 and NACE Standard RP-0169-96, the assessment of the condition of the coating and cathodic protection system is to be conducted on an annual basis and compared to predetermined values.

7. ***Corrective Actions:*** The site corrective actions program, quality assurance (QA) procedures, site review and approval process, and administrative controls are implemented in accordance with the requirements of 10 CFR Part 50, Appendix B. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B,

acceptable in addressing the corrective actions, confirmation process, and administrative controls.

8. ***Confirmation Process:*** See Item 7, above.

9. ***Administrative Controls:*** See Item 7, above.

10. ***Operating Experience:*** Corrosion pits from the outside diameter have been discovered in buried piping with far less than 60 years of operation. Buried pipe that is coated and cathodically protected is unaffected after 60 years of service. Accordingly, operating experience from application of the NACE standards on non-nuclear systems demonstrates the effectiveness of this program.

References

NACE Standard RP-0169-96, *Control of External Corrosion on Underground or Submerged Metallic Piping Systems*, 1996.

NACE Standard RP-0285-95, *Corrosion Control of Underground Storage Tank Systems by Cathodic Protection,* Approved March 1985, revised February 1995.

XI.M29 ABOVEGROUND CARBON STEEL TANKS

Program Description

The program includes preventive measures to mitigate corrosion by protecting the external surface of carbon steel tanks with paint or coatings in accordance with standard industry practice. The program also relies on periodic system walkdowns to monitor degradation of the protective paint or coating. However, for storage tanks supported on earthen or concrete foundations, corrosion may occur at inaccessible locations, such as the tank bottom. Accordingly, verification of the effectiveness of the program is to be performed to ensure that significant degradation in inaccessible locations is not occurring and the component intended function will be maintained during the extended period of operation. For reasons set forth below, an acceptable verification program consists of thickness measurement of the tank bottom surface.

Evaluation and Technical Basis

1. ***Scope of Program***: The program consists of (a) preventive measures to mitigate corrosion by protecting the external surfaces of carbon steel tanks protected with paint or coatings and (b) periodic system walkdowns to manage the effects of corrosion on the intended function of these tanks. Plant walkdowns cover the entire outer surface of the tank up to its surface in contact with soil or concrete.

2. ***Preventive Actions:*** In accordance with industry practice, tanks are coated with protective paint or coating to mitigate corrosion by protecting the external surface of the tank from environmental exposure. Sealant or caulking at the interface edge between the tank and concrete or earthen foundation mitigates corrosion of the bottom surface of the tank by preventing water and moisture from penetrating the interface, which would lead to corrosion of the bottom surface.

3. ***Parameters Monitored/Inspected:*** The aging management program (AMP) utilizes periodic plant system walkdowns to monitor degradation of coatings, sealants, and caulking because it is a condition directly related to the potential loss of materials.

4. ***Detection of Aging Effects:*** Degradation of exterior carbon steel surfaces cannot occur without degradation of paint or coatings on the outer surface and of sealant and caulking at the interface between the component and concrete. Periodic system walkdowns to confirm that the paint, coating, sealant, and caulking are intact is an effective method to manage the effects of corrosion on the external surface of the component. However, corrosion may occur at inaccessible locations, such as the tank bottom surface, and thickness measurement of the tank bottom is to be taken to ensure that significant degradation is not occurring and the component intended function will be maintained during the extended period of operation.

5. ***Monitoring and Trending:*** The effects of corrosion of the aboveground external surface are detectable by visual techniques. Based on operating experience, plant system walkdowns during each outage provide for timely detection of aging effects. The effects of corrosion of the underground external surface are detectable by thickness measurement of the tank bottom and are monitored and trended if significant material loss is detected.

6. ***Acceptance Criteria:*** Any degradation of paint, coating, sealant, and caulking is reported and will require further evaluation. Degradation consists of cracking, flaking, or peeling of

paint or coatings, and drying, cracking or missing sealant and caulking. Thickness measurements of the tank bottom are evaluated against the design thickness and corrosion allowance.

7. ***Corrective Actions:*** The site corrective actions program, quality assurance (QA) procedures, site review and approval process, and administrative controls are implemented in accordance with 10 CFR Part 50, Appendix B. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable in addressing the corrective actions, confirmation process, and administrative controls.

8. ***Confirmation Process:*** See Item 7, above.

9. ***Administrative Controls:*** See Item 7, above.

10. ***Operating Experience:*** Coating degradation, such as flaking and peeling, has occurred in safety-related systems and structures (Nuclear Regulatory Commission [NRC] Generic Letter [GL] 98-04). Corrosion damage near the concrete-metal interface and sand-metal interface has been reported in metal containments (NRC Information Notice [IN] 89-79, Supplement 1, and NRC IN 86-99, Supplement 1).

References

NRC Generic Letter 98-04, *Potential for Degradation of the Emergency Core Cooling System and the Containment Spray System after a Loss-of-Coolant Accident Because of Construction and Protective Coating Deficiencies and Foreign Material in Containment*, U.S. Nuclear Regulatory Commission, July 14, 1998.

NRC Information Notice 86-99, *Degradation of Steel Containments*, U.S. Nuclear Regulatory Commission, December 8, 1986.

NRC Information Notice 86-99, Supplement 1, *Degradation of Steel Containments*, U.S. Nuclear Regulatory Commission, February 14, 1991.

NRC Information Notice 89-79, *Degraded Coatings and Corrosion of Steel Containment Vessel*, U.S. Nuclear Regulatory Commission, December 1, 1989.

NRC Information Notice 89-79, Supplement 1, *Degraded Coatings and Corrosion of Steel Containment Vessel*, U.S. Nuclear Regulatory Commission, June 29, 1990.

XI.M30 FUEL OIL CHEMISTRY

Program Description

The program includes (a) surveillance and maintenance procedures to mitigate corrosion and (b) measures to verify the effectiveness of an aging management program (AMP) and confirm the absence of an aging effect. Fuel oil quality is maintained by monitoring and controlling fuel oil contamination in accordance with the guidelines of the American Society for Testing Materials (ASTM) Standards D 1796, D 2276, D 2709, and D 4057. Exposure to fuel oil contaminants, such as water and microbiological organisms, is minimized by periodic draining or cleaning of tanks and by verifying the quality of new oil before its introduction into the storage tanks. However, corrosion may occur at locations in which contaminants may accumulate, such as tank bottoms. Accordingly, the effectiveness of the program is verified to ensure that significant degradation is not occurring and the component intended function will be maintained during the extended period of operation. Thickness measurement of tank bottom surfaces is an acceptable verification program.

Evaluation and Technical Basis

1. ***Scope of Program:*** The program is focused on managing the conditions that cause general, pitting, and microbiologically influenced corrosion (MIC) of the diesel fuel tank internal surfaces. The program serves to reduce the potential of exposure of the tank internal surface to fuel oil contaminated with water and microbiological organisms.

2. ***Preventive Actions:*** The quality of fuel oil is maintained by additions of biocides to minimize biological activity, stabilizers to prevent biological breakdown of the diesel fuel, and corrosion inhibitors to mitigate corrosion. Periodic cleaning of a tank allows removal of sediments, and periodic draining of water collected at the bottom of a tank minimizes the amount of water and the length of contact time. Accordingly, these measures are effective in mitigating corrosion inside diesel fuel oil tanks. Coatings, if used, prevent or mitigate corrosion by protecting the internal surfaces of the tank from contact with water and microbiological organisms.

3. ***Parameters Monitored/Inspected:*** The AMP monitors fuel oil quality and the levels of water and microbiological organisms in the fuel oil, which cause the loss of material of the tank internal surfaces. The ASTM Standard D 4057 is used for guidance on oil sampling. The ASTM Standards D 1796 and D 2709 are used for determination of water and sediment contamination in diesel fuel. For determination of particulates, *modified* ASTM D 2276, Method A, is used. The modification consists of using a filter with a pore size of 3.0 μm, instead of 0.8 μm. These are the principal parameters relevant to tank structural integrity.

4. ***Detection of Aging Effects:*** Degradation of the diesel fuel oil tank cannot occur without exposure of the tank internal surfaces to contaminants in the fuel oil, such as water and microbiological organisms. Compliance with diesel fuel oil standards in item 3, above, and periodic multilevel sampling provide assurance that fuel oil contaminants are below acceptable levels. Internal surfaces of tanks that are drained for cleaning are visually inspected to detect potential degradation. However, corrosion may occur at locations in which contaminants may accumulate, such as a tank bottom, and an ultrasonic thickness measurement of the tank bottom surface ensures that significant degradation is not occurring.

5. ***Monitoring and Trending:*** Water and biological activity or particulate contamination concentrations are monitored and trended at least quarterly. Based on industry operating experience, quarterly sampling and analysis of fuel oil provide for timely detection of conditions conducive to corrosion of the internal surface of the diesel fuel oil tank before the potential loss of its intended function.

6. ***Acceptance Criteria:*** The ASTM Standard D 4057 is used for guidance on oil sampling. The ASTM Standards D 1796 and D 2709 are used for guidance on the determination of water and sediment contamiination in diesel fuel. Modified ASTM D 2276, Method A is used for determination of particulates. The modification consists of using a filter with a pore size of 3.0 μm, instead of 0.8 μm.

7. ***Corrective Actions:*** Specific corrective actions are implemented in accordance with the plant quality assurance (QA) program. For example, corrective actions are taken to prevent recurrence when the specified limits for fuel oil standards are exceeded or when water is drained during periodic surveillance. Also, when the presence of biological activity is confirmed, a biocide is added to fuel oil. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable in addressing corrective actions.

8. ***Confirmation Process:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with the requirements of 10 CFR Part 50, Appendix B. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable in addressing the confirmation process and administrative controls.

9. ***Administrative Controls:*** See Item 8, above.

10. ***Operating Experience:*** The operating experience at some plants has included identification of water in the fuel, particulate contamination, and biological fouling. However, no instances of fuel oil system component failures attributed to contamination have been identified.

References

ASTM D 1796-97, *Standard Test Method for Water and Sediment in Fuel Oils by the Centrifuge Method*, American Society for Testing Materials, West Conshohocken, PA.

ASTM D 2276-00, *Standard Test Method for Particulate Contaminant in Aviation Fuel by Line Sampling,* American Society for Testing Materials, West Conshohocken, PA.

ASTM D 2709-96, *Standard Test Method for Water and Sediment in Middle Distillate Fuels by Centrifuge*, American Society for Testing Materials, West Conshohocken, PA.

ASTM D 4057-95(2000), *Standard Practice for Manual Sampling of Petroleum and Petroleum Products*, American Society for Testing Materials, West Conshohocken, PA.

XI.M31 REACTOR VESSEL SURVEILLANCE

Program Description

The Code of Federal Regulations, 10 CFR Part 50, Appendix H, requires that peak neutron fluence at the end of the design life of the vessel will not exceed 10^{17} n/cm^2 (E >1MeV), or that reactor vessel beltline materials be monitored by a surveillance program to meet the American Society for Testing and Materials (ASTM) E 185 Standard. However, the surveillance program in ASTM E 185 is based on plant operation during the current license term, and additional surveillance capsules may be needed for the period of extended operation. Alternatively, an integrated surveillance program for the period of extended operation may be considered for a set of reactors that have similar design and operating features in accordance with 10 CFR Part 50, Appendix H, Paragraph II.C. Additional surveillance capsules may also be needed for the period of extended operation for this alternative.

The existing reactor vessel material surveillance program provides sufficient material data and dosimetry to monitor irradiation embrittlement at the end of the period of extended operation, and to determine the need for operating restrictions on the inlet temperature, neutron spectrum, and neutron flux. If surveillance capsules are not withdrawn during the period of extended operation, operating restrictions are to be established to ensure that the plant is operated under the conditions to which the surveillance capsules were exposed.

An acceptable reactor vessel surveillance program consists of the following:

1. The extent of reactor vessel embrittlement for upper-shelf energy and pressure-temperature limits for 60 years is projected in accordance with the Nuclear Regulatory Commission (NRC) Regulatory Guide (RG) 1.99, Rev. 2, “Radiation Embrittlement of Reactor Vessel Materials.” When using NRC RG 1.99, Rev. 2, an applicant has a choice of the following:

a. Neutron Embrittlement Using Chemistry Tables

An applicant may use the tables in NRC RG 1.99, Rev. 2, to project the extent of reactor vessel neutron embrittlement for the period of extended operation based on material chemistry and neutron fluence. This is described as Regulatory Position 1 in the RG.

b. Neutron Embrittlement Using Surveillance Data

When credible surveillance data are available, the extent of reactor vessel neutron embrittlement for the period of extended operation may be projected according to Regulatory Position 2 in NRC RG 1.99, Rev. 2, based on best fit of the surveillance data. The credible data could be collected during the current operating term. The applicant may have a plant-specific program or an integrated surveillance program during the period of extended operation to collect additional data.

2. An applicant that determines embrittlement by using the NRC RG 1.99, Rev. 2, tables (see item 1[a], above) uses the applicable limitations in Regulatory Position 1.3 of the RG. The limits are based on material properties, temperature, material chemistry, and fluence.

3. An applicant that determines embrittlement by using surveillance data (see item 1[b], above) defines the applicable bounds of the data, such as cold leg operating temperature and neutron fluence. These bounds are specific for the referenced surveillance data. For example, the plant-specific data could be collected within a smaller temperature range than that in the RG.

4. All pulled and tested capsules, unless discarded before August 31, 2000, are placed in storage. (Note: These specimens are saved for future reconstitution use, in case the surveillance program is reestablished.)

5. If an applicant has a surveillance program that consists of capsules with a projected fluence of less than the 60-year fluence at the end of 40 years, at least one capsule is to remain in the reactor vessel and is tested during the period of extended operation. The applicant may either delay withdrawal of the last capsule or withdraw a standby capsule during the period of extended operation to monitor the effects of long-term exposure to neutron irradiation.

6. If an applicant has a surveillance program that consists of capsules with a projected fluence exceeding the 60-year fluence at the end of 40 years, the applicant withdraws one capsule at an outage in which the capsule receives a neutron fluence equivalent to the 60-year fluence and tests the capsule in accordance with the requirements of ASTM E 185. Any capsules that are left in the reactor vessel provide meaningful metallurgical data (i.e., the capsule fluence does not significantly exceed the vessel fluence at an equivalent of 60 years). For example, in a reactor with a lead factor of three, after 20 years the capsule test specimens would have received a neutron exposure equivalent to what the reactor vessel would see in 60 years; thus, the capsule is to be removed since further exposure would not provide meaningful metallurgical data. Other standby capsules are removed and placed in storage. These standby capsules (and archived test specimens available for reconstitution) would be available for reinsertion into the reactor if additional license renewals are sought (e.g., 80 years of operation). If all surveillance capsules have been removed, operating restrictions are to be established to ensure that the plant is operated under conditions to which the surveillance capsules were exposed and the exposure conditions of the reactor vessel are monitored to ensure that they continue to be consistent with those used to project the effects of embrittlement to the end of license. If the reactor vessel exposure conditions (neutron flux, spectrum, irradiation temperature, etc.) are altered, then the basis for the projection to 60 years is reviewed; and, if deemed appropriate, an active surveillance program is re-instituted. Any changes to the reactor vessel exposure conditions and the potential need to re-institute a vessel surveillance program is discussed with the NRC staff prior to changing the plant's licensing basis.

7. Applicants without in-vessel capsules use alternative dosimetry to monitor neutron fluence during the period of extended operation, as part of the aging management program (AMP) for reactor vessel neutron embrittlement.

8. The applicant may choose to demonstrate that the materials in the inlet, outlet, and safety injection nozzles are not controlling, so that such materials need not be added to the material surveillance program for the license renewal term.

The reactor vessel monitoring program provides that, if future plant operations exceed the limitations or bounds specified in item 2 or 3, above (as applicable), such as operating at a lower cold leg temperature or higher fluence, the impact of plant operation changes on the extent of reactor vessel embrittlement will be evaluated and the NRC will be notified. An applicant without capsules in its reactor vessel is to propose reestablishing the reactor vessel surveillance program to assess the extent of embrittlement. This program will consist of (1) capsules from item 6, above; (2) reconstitution of specimens from item 4, above; and/or (3) capsules made from any available archival materials; or (4) some combination of the three previous options. This program could be a plant-specific program or an integrated surveillance program.

Evaluation and Technical Basis

Reactor vessel surveillance programs are plant specific, depending on matters such as the composition of limiting materials, availability of surveillance capsules, and projected fluence levels. In accordance with 10 CFR Part 50, Appendix H, an applicant submits its proposed withdrawal schedule for approval prior to implementation. Thus, further staff evaluation is required for license renewal.

References

10 CFR Part 50, Appendix H, *Reactor Vessel Material Surveillance Program Requirements*, Office of the Federal Register, National Archives and Records Administration, 2000.

ASTM E-185, *Standard Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels*, American Society for Testing Materials, Philadelphia, PA.

NRC Regulatory Guide 1.99, Rev. 2, *Radiation Embrittlement of Reactor Vessel Materials,* U.S. Nuclear Regulatory Commission.

XI.M32 ONE-TIME INSPECTION

Program Description

The program includes measures to verify the effectiveness of an aging management program (AMP) and confirm the absence of an aging effect. For example, for structures and components that rely on an AMP, such as water chemistry control, this program verifies the effectiveness of the AMP by confirming that unacceptable degradation is not occurring and the intended function of a component will be maintained during the extended period of operation. One-time inspection is needed to address concerns for the potential long incubation period for certain aging effects on structures and components. There are cases where either (a) an aging effect is not expected to occur but there is insufficient data to completely rule it out, or (b) an aging effect is expected to progress very slowly. For these cases, there is to be confirmation that either the aging effect is indeed not occurring, or the aging effect is occurring very slowly as not to affect the component or structure intended function. A one-time inspection of the subject component or structure is an acceptable option for this verification. One-time inspection is to provide additional assurance that either aging is not occurring or the evidence of aging is so insignificant that an aging management program is not warranted. For example, for structures and components, such as Class 1 piping with a diameter less than nominal pipe size (NPS) 4 inch that do not receive volumetric examination during inservice inspection, the program confirms that crack initiation and growth due to stress corrosion cracking (SCC) or cyclic loading is not occurring and, therefore, there is no need to manage an aging related degradation for the period of extended operation.

The elements of the program include (a) determination of the sample size based on an assessment of materials of fabrication, environment, plausible aging effects, and operating experience; (b) identification of the inspection locations in the system or component based on the aging effect; (c) determination of the examination technique, including acceptance criteria that would be effective in managing the aging effect for which the component is examined; and (d) evaluation of the need for follow-up examinations to monitor the progression of any aging degradation.

When evidence of an aging effect is revealed by a one-time inspection, the routine evaluation of the inspection results would identify appropriate corrective actions.

As set forth below, an acceptable verification program may consist of a one-time inspection of selected components and susceptible locations in the system. An alternative acceptable program may include routine maintenance or a review of repair records to confirm that these components have been inspected for aging degradation and significant aging degradation has not occurred and thereby verify the effectiveness of existing AMPs. One-time inspection, or any other action or program, is to be reviewed by the staff on a plant-specific basis.

Evaluation and Technical Basis

1. ***Scope of Program:*** The program includes measures to verify that unacceptable degradation is not occurring, thereby validating the effectiveness of existing AMPs or confirming that there is no need to manage aging-related degradation for the period of extended operation. The structures and components for which one-time inspection is to verify the effectiveness of the AMPs (e.g., water chemistry control, etc.) have been identified in the Generic Aging Lessons Learned (GALL) report. Examples include small bore piping in the reactor coolant system or the feedwater system components in boiling water reactors (BWRs) and pressurized water reactors (PWRs).

2. ***Preventive Actions:*** One-time inspection is an inspection activity independent of methods to mitigate or prevent degradation.

3. ***Parameters Monitored/Inspected:*** The program monitors parameters directly related to the degradation of a component. Inspection is performed in accordance with the requirements of the American Society of Mechanical Engineers (ASME) Code and 10 CFR 50, Appendix B, by using a variety of nondestructive examination (NDE) methods, including visual, volumetric, and surface techniques.

4. ***Detection of Aging Effects:*** The inspection includes a representative sample of the system population, and, where practical, focus on the bounding or lead components most susceptible to aging due to time in service, severity of operating conditions, and lowest design margin. For small-bore piping, actual inspection locations are based on physical accessibility, exposure levels, NDE techniques, and locations identified in Nuclear Regulatory Commission (NRC) Information Notice (IN) 97-46.

   Combinations of NDE, including visual, ultrasonic, and surface techniques, are performed by qualified personnel following procedures consistent with the ASME Code and 10 CFR 50, Appendix B. For small-bore piping less than NPS 4 in., including pipe, fittings, and branch connections, a plant-specific destructive examination of replaced piping due to plant modifications or NDE that permits inspection of the inside surfaces of the piping is to be conducted to ensure that cracking has not occurred. Follow-up of unacceptable inspection findings includes expansion of the inspection sample size and locations.

   The inspection and test techniques prescribed by the program verify any aging effects because these techniques, used by qualified personnel, have been proven effective and consistent with staff expectations. With respect to inspection timing, the one-time inspection is to be completed before the end of the current operating license. The applicant may schedule the inspection in such a way as to minimize the impact on plant operations. However, the inspection is not to be scheduled too early in the current operating term, which could raise questions regarding continued absence of aging effects prior to and near the extended period of operation.

5. ***Monitoring and Trending:*** One-time inspection does not provide specific guidance on monitoring and trending. However, evaluation of the appropriateness of the techniques and timing of the one-time inspection improve with the accumulation of plant-specific and industry-wide experience.

6. ***Acceptance Criteria*:** Any indication or relevant conditions of degradation detected are evaluated. The ultrasonic thickness measurements are to be compared to predetermined limits, such as design minimum wall thickness.

7. ***Corrective Actions:*** Site quality assurance (QA) procedures, review and approval processes, and administrative controls are implemented in accordance with the requirements of 10 CFR Part 50, Appendix B. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable in addressing the corrective actions, confirmation process, and administrative controls.

8. ***Confirmation Process:*** See Item 7, above.

9. ***Administrative Controls:*** See Item 7, above.

10. ***Operating Experience:*** One-time inspection is a new program to be applied by the applicant. The elements that comprise these inspections (e.g., the scope of the inspections and inspection techniques) are consistent with years of industry practice and staff expectations.

References

10 CFR 50.55a, *Codes and Standards*, Office of the Federal Register, National Archives and Records Administration, 2000.

ASME Section XI, *Rules for Inservice Inspection of Nuclear Power Plant Components*, ASME Boiler and Pressure Vessel Code, 1995 edition through the 1996 addenda, American Society of Mechanical Engineers, New York, NY.

NRC Information Notice 97-46, *Unisolable Crack in High-Pressure Injection Piping*, U.S. Nuclear Regulatory Commission, July 9, 1997.

XI.M33 SELECTIVE LEACHING OF MATERIALS

Program Description

The program for selective leaching of materials ensures the integrity of the components made of cast iron, bronze, brass, and other alloys exposed to a raw water, brackish water, treated water, or groundwater environment that may lead to selective leaching of one of the metal components. The aging management program (AMP) includes a one-time visual inspection and hardness measurement of selected components that may be susceptible to selective leaching to determine whether loss of materials due to selective leaching is occurring, and whether the process will affect the ability of the components to perform their intended function for the period of extended operation.

Evaluation and Technical Basis

1. ***Scope of Program:*** This AMP determines the acceptability of the components that may be susceptible to selective leaching and assess their ability to perform the intended function during the period of extended operation. These components include piping, valve bodies and bonnets, pump casings, and heat exchanger components. The materials of construction for these components may include cast iron, brass, bronze, or aluminum-bronze. These components may be exposed to a raw water, treated water, or groundwater environment. The AMP includes a one-time hardness measurement of a selected set of components to determine whether loss of material due to selective leaching is not occurring for the period of extended operation.

   The selective leaching process involves the preferential removal of one of the alloying elements from the material, which leads to the enrichment of the remaining alloying elements. Dezincification (loss of zinc from brass) and graphitization (removal of iron from cast iron) are examples of such a process. Susceptible materials, high temperatures, stagnant-flow conditions, and corrosive environment such as acidic solutions, for example, for brasses with high zinc content, and dissolved oxygen, are conducive to selective leaching.

2. ***Preventive Actions:*** The one-time visual inspection and hardness measurement is an inspection/verification program; thus, there is no preventive action. However, it is noted that monitoring of water chemistry to control pH and concentration of corrosive contaminants, and treatment with hydrazine to minimize dissolved oxygen in water are effective in reducing selective leaching.

3. ***Parameters Monitored/Inspected:*** The visual inspection and hardness measurement is to be a one-time inspection. Because selective leaching is a slow acting corrosion process, this measurement is performed just before the beginning of the license renewal period. Follow-up of unacceptable inspection findings includes expansion of the inspection sample size and location.

4. ***Detection of Aging Effects:*** The one-time visual inspection and hardness measurement includes close examination of a select set of components to determine whether selective leaching has occurred and whether the resulting loss of strength and/or material will affect the intended functions of these components during the period of extended operation. Selective leaching generally does not cause changes in dimensions and is difficult to detect. However, in certain brasses it causes plug-type dezincification, which can be detected by visual inspection. One acceptable procedure is to visually inspect the susceptible

components closely and conduct Brinell Hardness testing on the inside surfaces of the selected set of components to determine if selective leaching has occurred. If it is occurring, an engineering evaluation is initiated to determine acceptability of the affected components for further service.

5. ***Monitoring and Trending:*** There is no monitoring and trending for the one-time visual inspection and hardness measurement.

6. ***Acceptance Criteria:*** Identification of selective leaching will define the need for further engineering evaluation before the affected components can be qualified for further service. If necessary, the evaluation will include a root cause analysis.

7. ***Corrective Actions:*** Evaluations are performed for test or inspection results that do not satisfy established acceptance criteria. The corrective actions program ensures that conditions adverse to quality are promptly corrected. If the deficiency is assessed to be significantly adverse to quality, the cause of the condition is determined and an action plan is developed to preclude repetition. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable in addressing corrective actions.

8. ***Confirmation Process:*** Site quality assurance (QA) procedures, review and approval processes, and administrative controls are implemented in accordance with the requirements of 10 CFR Part 50, Appendix B. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable in addressing the confirmation process and administrative controls.

9. ***Administrative Controls:*** See Item 8, above.

10. ***Operating Experience:*** One-time inspection is a new program to be applied by the applicant. The elements that comprise these inspections (e.g., the scope of the inspections and inspection techniques) are consistent with years of industry practice and staff expectations.

References

NRC Safety Evaluation Report Related to the License Renewal of Calvert Cliffs Nuclear Power Plant, Units 1 and 2, NUREG-1705, December 1999.

NRC Safety Evaluation Report Related to the License Renewal of Oconee Nuclear Station, Units 1, 2, and 3, NUREG-1723, March 2000.

XI.M34 BURIED PIPING AND TANKS INSPECTION

Program Description

The program includes (a) preventive measures to mitigate corrosion, and (b) periodic inspection to manage the effects of corrosion on the pressure-retaining capacity of buried carbon steel piping and tanks. Preventive measures are in accordance with standard industry practice for maintaining external coatings and wrappings. Buried piping and tanks are inspected when they are excavated during maintenance and when a pipe is dug up and inspected for any reason.

As evaluated below, this is an acceptable option to manage buried components, except for the program element/attributes of detection of aging effects (regarding inspection frequency) and operating experience. Thus, the staff further evaluates an applicant's inspection frequency and operating experience with buried components.

Evaluation and Technical Basis

1. ***Scope of Program:*** The program relies on preventive measures such as coating and wrapping and periodic inspection for loss of material caused by corrosion of the external surface of buried carbon steel piping and tanks. Loss of material in these components, which may be exposed to aggressive soil environment, is caused by general, pitting, and crevice corrosion, and microbiologically influenced corrosion (MIC). Periodic inspections are performed when the components are excavated for maintenance or for any other reason. The scope of the program covers buried components that are within the scope of license renewal for the plant.

2. ***Preventive Actions:*** In accordance with industry practice, underground piping and tanks are coated during installation with a protective coating system, such as coal tar enamel with a fiberglass wrap and a kraft paper outer wrap, a polyolifin tape coating, or a fusion bonded epoxy coating to protect the piping from contacting the aggressive soil environment.

3. ***Parameters Monitored/Inspected:*** The program monitors parameters such as coating and wrapping integrity that are directly related to corrosion damage of the external surface of buried carbon steel piping and tanks. Coatings and wrappings are inspected by visual techniques. Any evidence of damaged wrapping or coating defects, such as coating perforation, holidays, or other damage, is an indicator of possible corrosion damage to the external surface of piping and tanks.

4. ***Detection of Aging Effects:*** Periodic inspection of susceptible locations to confirm that coating and wrapping are intact, is an effective method to ensure that corrosion of external surfaces has not occurred and the intended function is maintained. Buried piping and tanks are inspected when they are excavated during maintenance. The inspections are performed in areas with the highest likelihood of corrosion problems, and in areas with a history of corrosion problems. However, because the inspection frequency is plant specific and also depends on the plant operating experience, the applicant's proposed inspection frequency is to be further evaluated for the extended period of operation.

5. ***Monitoring and Trending:*** Results of previous inspections are used to identify susceptible locations.

6. ***Acceptance Criteria:*** Any coating and wrapping degradations are reported and evaluated according to site corrective actions procedures.

7. ***Corrective Actions:*** The site corrective actions program, quality assurance (QA) procedures, site review and approval process, and administrative controls are implemented in accordance with the requirements of 10 CFR Part 50, Appendix B. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable in addressing the corrective actions, confirmation process, and administrative controls.

8. ***Confirmation Process:*** See Item 7, above.

9. ***Administrative Controls:*** See Item 7, above.

10. ***Operating Experience:*** Operating experience shows that the program described here is effective in managing corrosion of external surfaces of buried carbon steel components. However, because the inspection frequency is plant specific and also depends on the plant operating experience, the applicant's plant-specific operating experience is further evaluated for the extended period of operation.

References

None.