## Proposed Changes to NUREG-1801 Volume II Aging Management Programs

General Comments

For details and markups related to these comments, see individual aging management programs (AMPs). Additions are identified by double underline; deleted text is noted by a strike-through.

G.1 Programs include actions that are beyond the control of the program.

**Discussion:** Certain programs include design considerations that you either have or not. These actions create unnecessary exceptions even in cases where the program is adequate to address the aging effects with or without the exception.

**Recommendation:** Revise NUREG-1801 to delete these actions from the program.

This affects the following programs.

XI.M5, BWR Feedwater Nozzle

XI.M6, BWR Control Rod Drive Return Line Nozzle

XI.M10, Boric Acid Corrosion

XI.M20, Open-Cycle Cooling Water System

G.2 Several AMPs have a specific ASME Code year in the description of the AMP or in one of the ten elements.

**Discussion:** The NUREG-1801 AMPs that credit the ASME Code ISI inspections specifically include the Code year (e.g., 2001 edition including the 2002 and 2003 Addenda), as endorsed by the NRC in 10 CFR 50.55a. The current NRC process for review of updated versions of the ASME Code for endorsement under 10 CFR 50.55a includes a review to determine if the new version of the ASME Code meets the 10 CFR 54 requirements to be an acceptable AMP. NUREG-1801 Rev. 0 credited the ISI inspection plans in accordance with ASME Code 1995 edition through 1996 addenda. The NRC has accepted other editions of the ASME Code that were the basis for licensees ISI plans. The use of ASME editions earlier than those specified in NUREG-1801 has been accepted by the NRC as documented in the SERs for LRAs.

**Recommendation:** Revise NUREG-1801 to state that any edition of the ASME code that has been endorsed by the NRC under 10 CFR 50.55a is an acceptable AMP and is consistent with NUREG-1801. This would eliminate the need for the NRC to revise NUREG-1801 every time the NRC updates 10 CFR 50.55a to endorse a later edition of the ASME Code.

This affects the following programs.

XI.M1, ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD

XI.M3, Reactor Head Closure Bolts

XI.M4, BWR Vessel ID Attachment Welds

XI.M5, BWR Feedwater Nozzles

XI.M6, BWR Control Rod Drive Return Line Nozzle

XI.M8, BWR Penetrations

XI.M9, BWR Vessel Internals

XI.M25, BWR Reactor Water Cleanup System (The text and the reference is to 2001 edition of the ASME Code, but both should refer to 1986 edition.)

XI.S1, ASME Section XI, Subsection IWE (The details in attribute 3 are based on the 1992 edition of the ASME code, and may not be accurate for other editions.)

XI.S2, ASME Section XI, Subsection IWL

XI.S3, ASME Section XI, Subsection IWF

G.3 Several AMPs have a specific ASME Code year listed in the references.

**Discussion:** Although the ASME Code edition is not in the body of the AMP, it is listed in the references (e.g., 2001 edition including the 2002 and 2003 Addenda); or references in the NUREG-1801 attributes to specific sections of the code are taken from a specific edition of the ASME Code.

**Recommendation:** Revise NUREG-1801 to state that any edition of the ASME code that has been endorsed by the NRC under 10 CFR 50.55a is an acceptable AMP and is consistent with NUREG-1801. This would eliminate the need for the NRC to revise NUREG-1801 every time the NRC updates 10 CFR 50.55a to endorse a later edition of the ASME Code. The revision should state that later Code editions than those listed in the references are acceptable.

This affects the following programs.

XI.M7, BWR Stress Corrosion Cracking (reference to ASME Code in element 6 indicates a specific code edition.)

XI.M12, Thermal Aging Embrittlement of CASS (Reference to specific sections of the ASME Code in attributes 6 and 7 indicate a specific code edition.)

XI.M13, Thermal Aging and Neutron Irradiation Embrittlement of CASS (Reference to specific sections of the ASME Code in attributes 6 and 7 indicate a specific code edition.)

XI.M18, Bolting Integrity (The text correctly references the 1995 edition of the ASME Code, but the reference section lists the 2001 edition of the ASME Code)

XI.M32, One-Time Inspection (References identify a specific code edition)

XI.M35, One-Time Inspection of ASME Code Class 1 Small Bore-Piping (References identify a specific code edition)

G.4 Several AMPs reference a specific year or revision to an industry standard.

**Discussion:** This issue is similar to G.2 and G.3 in that industry standards, just like 10 CFR 50.55a, will change with time; but this issue is different in that the NRC does not review and approve or endorse these standards. There are two different ways in which this issue can be resolved. The first is for standards that are revised based on actual operating experience using an industry consensus approval process. Use of later revisions to these standards should be acceptable as AMPs for a LRA. The NRC always has the option not to accept an AMP based on a later revision of the industry standard if the change is not acceptable. The second is when a plant is committed to a specific revision of the standard (e.g., NUMARC-1801 for steam generator tube inspection). For these, the AMP should permit use of the version of the standard to which the plant is committed.

**Recommendation:** Revise NUREG-1801 to state that later revisions of these standards are acceptable as references for an AMP. The NRC should clearly state in the NUREG-1801 that the use of later revisions to these standards does not constitute an exception to NUREG-1801.

This affects the following programs.

XI.M17, Flow-Accelerated Corrosion

XI.M19, Steam Generator Tube Integrity

XI.M21, Closed-Cycle Cooling Water System

XI.M35, One-Time Inspection of ASME Code Class 1 Small Bore-Piping

XI.S6, Structures Monitoring Program (ACI Code)

XI.S7, RG 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants (ACI Code)

XI.S8, Protective Coating Monitoring and Maintenance Program

G.5 Several AMPs are applied with an expanded scope to include components, materials, environments, or aging effects not covered in the NUREG-1801 AMP.

**Discussion:** These AMPs as described in the NUREG-1801 do not include common components and material-environment combinations for the applicable SSCs in nuclear power plants, but the AMPs still can be used to adequately manage aging. Often, these material-environment combinations are covered by NUREG-1801, but are managed by similar activities in different AMPs.

**Recommendation:** Revise the NUREG-1801 AMPs to include common material-environment-aging-effects combinations that have been included in previous LRAs.

**Recommendation:** Revise NUREG-1801 to so that AMPs are not limited to specific components, such as XI.M30, which is written only for diesel fuel oil tanks but is equally applicable to other components in a fuel oil system.

This affects the following programs.

XI.M20, Open-Cycle Cooling Water System (NUREG-1801 does not include aging effects for concrete)

XI.M29, Aboveground Steel Tanks (This AMP has been used for other materials, e.g., aluminum; exceptions also taken for frequency)

XI.M30, Fuel Oil Chemistry (The scope of the AMP is only for fuel oil storage tanks, but it is also used for other fuel oil system components)

XI.M34, Buried Piping and Tanks Inspections (Only for steel tanks and piping, does not include stainless steel, aluminum, titanium, or other materials; credits coatings, which may not be used on some materials, such as stainless steel; specifies external inspections when UT from inside has been used frequently)

XI.M36, External Surfaces Monitoring (NUREG-1801 AMP is limited to steel components; AMP is used for other materials, including elastomers, with corresponding aging effects)

XI.M38, Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components (NUREG-1801 AMP is limited to steel components; AMP is used for other materials, including elastomers, with corresponding aging effects and inspection techniques)

G.6 AMPs have been accepted with different inspection methods (XI.M30 Fuel Oil Chemistry and XI.M33 Selective Leaching of Materials).

**Discussion:** NUREG-1801 credits the most recent industry standards for fuel oil chemistry. Many plants have specific requirements in the plant Technical Specifications for testing of fuel oil. These requirements have proven to be adequate for the plant life to maintain fuel oil condition, and will also be adequate during the period of extended operation.

**Recommendation:** Revise XI.M30 to credit fuel oil chemistry programs that are included in the plant Technical Specifications.

**Recommendation:** Revise XI.M33 to permit use of additional methods that have been accepted in license renewal SERs, such as scraping and chipping, and laboratory examinations.

This affects the following programs.

XI.M30, Fuel Oil Chemistry

XI.M33, Selective Leaching of Materials

**X.M1 Metal Fatigue**

X.M1-1 The scope states preventive actions. This is not correct. This is a monitoring program that includes no preventive actions.

X.M1-2 Under Element 4, the program also specifies periodic update of fatigue calculations. Typically, updates are performed on an as-needed basis if an allowable number of cycles is approached.

X.M1-3 In Element 6, change "acceptance criteria involves" to "acceptance criteria include…"

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## X.M1 METAL FATIGUE OF REACTOR COOLANT PRESSURE BOUNDARY

### Program Description

In order not to exceed the design limit on fatigue usage, the aging management program (AMP) monitors and tracks the number of critical thermal and pressure transients for the selected reactor coolant system components.

The AMP addresses the effects of the coolant environment on component fatigue life by assessing the impact of the reactor coolant environment on a sample of critical components for the plant. Examples of critical components are identified in NUREG/CR-6260. The sample of critical components can be evaluated by applying environmental life correction factors to the existing ASME Code fatigue analyses. Formulae for calculating the environmental life correction factors are contained in NUREG/CR-6583 for carbon and low-alloy steels and in NUREG/CR-5704 for austenitic stainless steels.

As evaluated below, this is an acceptable option for managing metal fatigue for the reactor coolant pressure boundary, considering environmental effects. Thus, no further evaluation is recommended for license renewal if the applicant selects this option under 10 CFR 54.21(c)(1)(iii) to evaluate metal fatigue for the reactor coolant pressure boundary.

### Evaluation and Technical Basis

1. ***Scope of Program:*** The program monitors and tracks the number of critical thermal and pressure transients for the selected reactor coolant system components. For a sample of components, the program includes fatigue usage calculations that consider the effects of the reactor water environment. ~~includes preventive measures to mitigate~~This ensures the design limit on fatigue usage is not exceeded, thus limiting fatigue cracking of metal components of the reactor coolant pressure boundary caused by anticipated cyclic strains in the material.

2. ***Preventive Actions:*** ~~Maintaining the fatigue usage factor below the design code limit and considering the effect of the reactor water environment, as described under the program description, will provide adequate margin against fatigue cracking of reactor coolant system components due to anticipated cyclic strains.~~ The program consists of monitoring and tracking and provides no preventive action.

3. ***Parameters Monitored/Inspected:*** The program monitors all plant transients that cause cyclic strains, which are significant contributors to the fatigue usage factor. The number of

plant transients that cause significant fatigue usage for each critical reactor coolant pressure boundary component is to be monitored. Alternatively, more detailed local monitoring of the plant transient may be used to compute the actual fatigue usage for each transient.

4. ***Detection of Aging Effects:*** The program provides for ~~periodic~~ updates of the fatigue usage calculations on an as-needed basis if an allowable number of cycles is approached.

5. ***Monitoring and Trending:*** The program monitors a sample of high fatigue usage locations. This sample is to include the locations identified in NUREG/CR-6260, as minimum, or propose alternatives based on plant configuration.

6. ***Acceptance Criteria:*** The acceptance criteria ~~involves~~ include maintaining the fatigue usage below the design code limit considering environmental fatigue effects as described under the program description.

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## X.S1 Concrete Containment Tendon Prestress

X.S1-1 Exceptions to inspections are routinely taken. NUREG-1801 describes the assessment of the adequacy of prestressing force of containment prestressed tendons, and refers to criteria in RG 1.35.1, but some plants use criteria and methods in ASME XI, IWL as incorporated in 10 CFR 50.55a to calculate tendon prestress. Revise NUREG-1801 to allow the use of either RG 1.35.1 or ASME Code to calculate tendon prestress.

X.S1-2. The details of the program should be included in the ten elements. According to SRP, the ten elements should define an acceptable program. However, much of the detail of this AMP is included only in the program description. The program description should be a high level summary description and the ten elements should define the program. Text in *italics* indicates that it was relocated.

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### X.S1 CONCRETE CONTAINMENT TENDON PRESTRESS

**Program Description**

In order to ensure the adequacy of prestressing forces in prestressed concrete containment tendons during the extended period of operation, an applicant shall develop an aging management program (AMP) under 10 CFR 54.21(c)(1)(iii).

The AMP consists of an assessment of the results of inspections performed in accordance with the requirements of Subsection IWL of the ASME Section XI Code, as supplemented by the requirements of 10 CFR 50.55a(b)(2)(ix) or (viii) in the later amendment of the regulation. The assessment related to the adequacy of the prestressing force will consist of the establishment of (1) acceptance criteria and (2) trend lines. ~~The acceptance criteria will normally consist of predicted lower limit (PLL) and the minimum required prestressing force, also called minimum required value (MRV). NRC Regulatory Guide 1.35.1 provides guidance for calculating PLL and MRV. The trend line represents the trend of prestressing forces based on the actual measured forces. NRC Information Notice IN 99-10 provides guidance for constructing the trend line. The goal is to keep the trend line above the PLL because, as a result of any inspection performed in accordance with ASME Section XI, Subsection IWL, if the trend line crosses the PLL, the existing prestress in the containment tendon could go below the MRV soon after the inspection and would not meet the requirements of 10 CFR 50.55a(b)(2)(ix)(B) or 10 CFR 50.55a(b)(2)(viii)(B).~~

As evaluated below, this time limited aging analysis (TLAA) is an acceptable option to manage containment tendon prestress force, except for the program element/attribute regarding operating experience. Thus, it is recommended that the staff should further evaluate an applicant's operating experience related to the containment tendon prestress force.

The AMP related to the adequacy of prestressing force for containments with grouted tendons will be reviewed on a case-by-case basis.

**Evaluation and Technical Basis**

1. ***Scope of Program:*** The program addresses the assessment of containment tendon prestressing force when an applicant chooses to perform the containment prestress force TLAA using 10 CFR 54.21(c)(1)(iii).

2. ***Preventive Actions:*** Maintaining the prestress above the minimum required value (MRV), as described under ~~program description above~~acceptance criteria below, will ensure that the structural and functional adequacy of the containment are maintained.

3. ***Parameters Monitored:*** The parameters to be monitored are the containment tendon prestressing forces in accordance with requirements specified in Subsection IWL of Section XI of the ASME Code, as incorporated by reference in 10 CFR 50.55a.

4. ***Detection of Aging Effects:*** The loss of containment tendon prestressing forces is detected by the program.

5. ***Monitoring and Trending:*** The estimated and measured prestressing forces are plotted against time and the predicted lower limit (PLL), MRV, and trending lines developed for the period of extended operation. *NRC Regulatory Guide 1.35.1 provides guidance for calculating PLL and MRV. The trend line represents the trend of prestressing forces based on the actual measured forces. NRC Information Notice IN 99-10 provides guidance for constructing the trend line.*

6. ***Acceptance Criteria:*** The prestressing force trend lines indicate that existing prestressing forces in the containment tendon would not be below the MRVs prior to the next scheduled inspection, as required by 10 CFR 50.55a(b)(2)(ix)(B) or 10 CFR 50.55a(b)(2)(viii)(B). *The acceptance criteria will normally consist of* the ~~*predicted lower limit*~~ *(PLL*~~*)*~~ *and the minimum required prestressing force, also called* ~~*minimum required value (*~~the *MRV*~~*)*~~*. The goal is to keep the trend line above the PLL because, as a result of any inspection performed in accordance with ASME Section XI, Subsection IWL, if the trend line crosses the PLL, the existing prestress in the containment tendon could go below the MRV soon after the inspection and would not meet the requirements of 10 CFR 50.55a(b)(2)(ix)(B) or 10 CFR 50.55a(b)(2)(viii)(B).*

7. ***Corrective Actions:*** If acceptance criteria are not met, then either systematic retensioning of tendons or a reanalysis of the containment is warranted to ensure the design adequacy of the containment. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the corrective actions.

8. ***Confirmation Process:*** As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the confirmation process.

9. ***Administrative Controls:*** As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the administrative controls.

10. ***Operating Experience:*** The program incorporates the relevant operating experience that has occurred at the applicant's plant as well as at other plants. The applicable portions of the experience with prestressing systems described in NRC Information Notice 99-10 could be useful for the purpose. However, tendon operating experience could be different at plants with prestressed concrete containments. The difference could be due to the prestressing system design (e.g., button-headed, wedge, or swaged anchorages), environment, and type of reactor (i.e., PWR and BWR). Thus, the applicant's plant-specific operating experience should be further evaluated for license renewal.

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## XI.M1 ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD

XI.M1-1. In Element 4, Category B-E is now completely covered by category B-P and is no longer in the code. Thus, discussion of category B-E should be removed from Element 4.

XI.M1-2. Element 5, "Monitoring and Trending" states in part, "The areas containing such flaw indications and relevant conditions are reexamined during the next three inspection periods of IWB-2110 for Class 1 components, IWC-2410 for Class 2 components, and IWD-2410 for Class 3 components." NUREG-1801 apparently has a typographical error in stating that Subsection IWB-2***1***10 specifies the inspection periods for Class 1 components. Subsection IWB-2***4***10 contains this information.

XI.M1-3. See General Comment G-2. Revise footnote to state that any edition of the ASME code endorsed by NRC in 10 CFR 55a is acceptable.

XI.M1-4. Subsections IWB-4000, IWC-4000, IWD-4000, IWB-7000, IWC-7000 and IWD-7000 do not exist in ASME Section XI 2001 edition including the 2002 and 2003 addenda as they were incorporated into Subsection IWA-4000 in a previous edition of the code. This affects Element #7.

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

**Program Description**

Title 10 of 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 2001 edition[1] including the 2002 and 2003 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

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[1] An applicant may rely on a different version of the ASME Code~~, but should justify such use. An applicant may wish to refer to the SOC for an update of 10 CFR § 50.55a to justify use of a more recent edition of the Code~~. An edition of the ASME Code that has been endorsed by the NRC under 10 CFR 50.55a is acceptable and is considered consistent with NUREG-1801.

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:*** Operation within the limits prescribed in the Technical Specifications.

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 cracking, 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 specifyies 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 this 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, including the uncertainties inherent in delivering and executing an 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 provided such relief is submitted under the provisions of 10 CFR 50.55a and approved by the staff.

The ASME Section XI examination categories used in this report are given below.

**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 surface.

~~*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 inch (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; 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-K 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 eliminated 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. wall thickness for piping >NPS 4 in. or for piping welds >1/5 in. 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 eliminated 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 conditions or relevant conditions of degradation are evaluated in accordance with IWB-3100, IWC-3100, or IWD-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-21410 for Class 1 components, IWC-2410 for Class 2 components, and IWD-2410 for Class 3 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.

7. ***Corrective Actions:*** Repair and replacement will be in conformance with ASME Section XI, Subsection IWA-4000, "Repair/Replacement Activities," or an alternative approved under the provisions of 10 CFR 50.55a. ~~For Class 1, 2, and 3, respectively, repair is performed 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 to address the corrective actions.

….

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 Bulletin 80-13, NRC Information Notice [IN] 95-17, NRC Generic Letter [GL] 94-03, and NUREG-1544). Cracking 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) caused by 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 been observed in plants in the United States. Cracking due to thermal and mechanical loading has occurred in high-pressure injection and safety injection piping (NRC IN 97-46 and NRC Bulletin 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).

## XI.M3 Reactor Head Closure Studs

XI.M3-1. Title should be all capitals.

XI.M3-2. Element 4 "detection of aging effects" states in part: "…surface and volumetric examination of studs when removed." However, since ASME Code Edition 2001 including the 2002 and 2003 addenda permits either surface or volumetric examination when studs are removed, it appears that this phrase should have been changed to "surface or volumetric examination of studs when removed" when the ASME code version cited in NUREG-1801 was changed.

XI.M3-3 See General Comment G-2. Revise to state that any edition of the ASME code that has been endorsed by NRC under 10 CFR 50.55a is acceptable.

XI.M3-4 Subsections IWB-4000 and IWB-7000 do not exist in ASME Section XI 2001 edition including the 2002 and 2003 addenda as they were incorporated into Subsection IWA-4000 in a previous edition of the code. This affects Element #7.

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## 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 (2001 edition[2] including the 2002 and 2003 Addenda), Table IWB 2500-1, and (b) preventive measures to mitigate cracking.

### Evaluation and Technical Basis

….

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 cracking, loss of material due to corrosion or wear, and leakage of coolant.

   ….

   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. 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~~or 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-

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[2] An applicant may rely on a different version of the ASME Code~~, but should justify such use. An applicant may wish to refer to the SOC for an update of 10 CFR § 50.55a to justify use of a more recent edition of the Code~~. An edition of the ASME Code that has been endorsed by the NRC under 10 CFR 50.55a is acceptable and is considered consistent with NUREG-1801.

retaining components specifies visual VT-2 examination of all pressure-retaining boundary components during the system leakage test and the system hydrostatic test.

…

7. ***Corrective Actions:*** Repair and replacement will be in conformance with ASME Section XI, Subsection IWA-4000, “Repair/Replacement Activities,” or an alternative approved under the provisions of 10 CFR 50.55a.~~Repair and replacement are performed 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 to address the corrective actions.

## XI.M4 BWR Vessel ID Attachment Welds

XI.M4-1 See General Comment G-2. Revise to state that any edition of the ASME code that has been endorsed by NRC under 10 CFR 50.55a is acceptable.

XI.M4-2 Subsections IWB-4000 and IWB-7000 do not exist in ASME Section XI 2001 edition including the 2002 and 2003 addenda as they were incorporated into Subsection IWA-4000 in a previous edition of the code. This affects Element #7.

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## 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

…

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 (2001 edition[3] including the 2002 and 2003 Addenda). An applicant may use the guidelines of BWRVIP-62 for inspection relief for vessel internal components with hydrogen water chemistry provided that such relief is submitted under the provisions of 10 CFR 50.55a and approved by the staff.

…

7. ***Corrective Actions:*** Repair and replacement procedures are equivalent to those requirements in the ASME Section XI. Repair and replacement will be in conformance with ASME Section XI, Subsection IWA-4000, “Repair/Replacement Activities,” or an alternative approved under the provisions of 10 CFR 50.55a. ~~Repair is performed 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.

…

3 An applicant may rely on a different version of the ASME Code~~, but should justify such use. An applicant may wish to refer to the SOC for an update of 10 CFR § 50.55a to justify use of a more recent edition of the Code~~. An edition of the ASME Code that has been endorsed by the NRC under 10 CFR 50.55a is acceptable and is considered consistent with NUREG-1801.

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**<u>XI.M5 BWR Feedwater Nozzle</u>**

XI.M5-1 See General Comment G-1. Program includes design considerations.

XI.M5-2 See General Comment G-2. Revise to state that any edition of the ASME code that has been endorsed by NRC under 10 CFR 50.55a is acceptable.

XI.M5-3 Subsections IWB-4000 and IWB-7000 do not exist in ASME Section XI 2001 edition including the 2002 and 2003 addenda as they were incorporated into Subsection IWA-4000 in a previous edition of the code. This affects Element #7.

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## 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 (2001 edition[4] including the 2002 and 2003 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 the boiling water reactor (BWR) feedwater nozzle.

<u>Systems modifications to mitigate cracking may have been made, 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. However, these modifications and changes are not part of the aging management program.</u>

### Evaluation and Technical Basis

…

1. ***Scope of Program:*** The program includes enhanced ISI to monitor the effects of cracking 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.~~<u>This program is a monitoring program and has no preventive actions.</u>

…

7. ***Corrective Actions:*** ~~Repair is performed in conformance with IWB-4000 and replacement in accordance with IWB-7000~~ <u>Repair and replacement are in conformance with ASME Section XI, Subsection IWA-4000, “Repair/Replacement Activities,” or an alternative approved under the provisions of 10 CFR 50.55a</u>. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the corrective actions.

…

4 An applicant may rely on a different version of the ASME Code~~, but should justify such use. An applicant may wish to refer to the SOC for an update of 10 CFR § 50.55a to justify use of a more recent edition of the Code~~. An edition of the ASME Code that has been endorsed by the NRC under 10 CFR 50.55a is acceptable and is considered consistent with NUREG-1801.

## XI.M6 BWR Control Rod Drive Return Line Nozzle

XI.M6-1. See General Comment G.1.
BWR CRD return line program specifies plant modifications that are not governed by the site program. They are design features that are not likely to be changed by the program.

XI.M6-2 See General Comment G-2. Revise to state that any edition of the ASME code that has been endorsed by NRC under 10 CFR 50.55a is acceptable.

XI.M6-3 Subsections IWB-4000 and IWB-7000 do not exist in ASME Section XI 2001 edition including the 2002 and 2003 addenda as they were incorporated into Subsection IWA-4000 in a previous edition of the code. This affects Element #7.

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## 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 (2001 edition[5] including the 2002 and 2003 Addenda) and the recommendations of NUREG-0619, and (b) ~~system modifications and~~ maintenance programs to mitigate cracking. System modifications may have been made to mitigate cracking, such as rerouting the CRDRL to a system that connects to the reactor vessel or cutting and capping the CRDRL nozzle without rerouting. The program specifies periodic liquid penetrant and ultrasonic inspection of critical regions of the 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 cracking 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 is ~~also~~ accomplished by confirmation of proper return flow capability~~,~~ and two-pump operation~~, and cutting and capping the CRDRL nozzle without rerouting~~.

…

7. ***Corrective Actions:*** ~~Repair is performed in conformance with IWB-4000 and replacement in accordance with IWB-7000~~ Repair and replacement are in conformance with ASME Section XI, Subsection IWA-4000, "Repair/Replacement Activities," or an alternative approved under the provisions of 10 CFR 50.55a. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the corrective actions.

…

5 An applicant may rely on a different version of the ASME Code~~, but should justify such use. An applicant may wish to refer to the SOC for an update of 10 CFR § 50.55a to justify use of a more recent edition of the Code~~. An edition of the ASME Code that has been endorsed by the NRC under 10 CFR 50.55a is acceptable and is considered consistent with NUREG-1801.

**<u>XI.M7 BWR Stress Corrosion Cracking</u>**

XI.M7-1 See General Comment G-3. NUREG-1801 lists both ASME 1986 and 2001 in the references. The text correctly calls out 1986 edition of ASME Code. The 1986 edition of the ASME Code is the code edition referenced by GL 88-01. In the list of references, the reference for the 2001 edition should be deleted.

XI.M7-2 Based on the NRC teleconference of April 14, 2008, use of later code versions does not constitute an exception to NUREG-1801 since the NRC staff evaluates later editions of the ASME Code for their adequacy for license renewal as part of the 10 CFR 50.55a rulemakings. The guidance from this teleconference should be incorporated into this AMP.

XI.M7-3 Subsections IWB-4000, IWB-7000, IWC-4000, IWC-7000, IWD-4000 and IWD-7000 do not exist in ASME Section XI 2001 edition including the 2002 and 2003 addenda as they were incorporated into Subsection IWA-4000 in a previous edition of the code. This affects Element #7.

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## 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) and nickel based alloy components is delineated in NUREG-0313, Rev. 2, and Nuclear Regulatory Commission (NRC) Generic Letter (GL) 88-01 and its Supplement 1. The material includes base metal and welds. 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 to the inspection scope in the GL 88-01 program.

### Evaluation and Technical Basis

…

6. ***Acceptance Criteria:*** As recommended in NRC GL 88-01, any indication detected is evaluated in accordance with ASME Section XI, IWB-3600 of Section XI of the 1986 Edition of the ASME Boiler and Pressure Vessel Code 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<u>. Repair and replacement will be in conformance with ASME Section XI, Subsection IWA-4000, “Repair/Replacement Activities,” or an alternative approved under the provisions of 10 CFR 50.55a.</u>~~; 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 to address the corrective actions.

…

**References**

10 CFR Part 50, Appendix B, Quality Assurance Criteria for Nuclear Power Plants, Office of the Federal Register, National Archives and Records Administration, 2005.

10 CFR 50.55a, Codes and Standards, Office of the Federal Register, National Archives and Records Administration, 2005.

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, 2001 edition including the 2002 and 2003 Addenda, 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, 1986 edition, 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-59, Evaluation of Crack Growth in BWR Nickel-Base Austenitic Alloys in RPV Internals, (EPRI TR-108710), 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), 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), 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.

## XI.M8 BWR Penetrations

XI.M8-1. Under element 5, text reads: "Inspections scheduled in accordance with IWB-2400 and approved BWRVIP-48 or BWRVIP-27 provide timely detection of cracks. The scope of examination and reinspection must be expanded beyond the baseline inspection if flaws are detected." BWRVIP-48 should be BWRVIP-49.

XI.M8-2. Under element 7, text reads: "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, corrective actions." BWRVIP-48 should be BWRVIP-49.

XI.M8-3 See General Comment G-2. Revise to state that any edition of the ASME code that has been endorsed by NRC under 10 CFR 50.55a is acceptable.

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### 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. BWRVIP-49 provides guidelines for instrument penetrations, and BWRVIP-27 addresses the standby liquid control (SLC) system nozzle or housing.

**Evaluation and Technical Basis**

…

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 (2001 edition[6] including the 2002 and 2003 Addenda). An applicant may use the guidelines of BWRVIP-62 for inspection relief for vessel internal components with hydrogen water chemistry, provided that such relief is submitted under the provisions of 10 CFR 50.55a and approved by the staff.

…

6 An applicant may rely on a different version of the ASME Code~~, but should justify such use. An applicant may wish to refer to the SOC for an update of 10 CFR § 50.55a to justify use of a more recent edition of the Code~~. An edition of the ASME Code that has been endorsed by the NRC under 10 CFR 50.55a is acceptable and is considered consistent with NUREG-1801.

…

5. ***Monitoring and Trending:*** Inspections scheduled in accordance with IWB-2400 and approved BWRVIP-4~~8~~9 or BWRVIP-27 provide timely detection of cracks. The scope of examination and reinspection must be expanded beyond the baseline inspection if flaws are detected.

...

7. ***Corrective Actions:*** Repair and replacement procedures in staff-approved BWRVIP-57 and BWRVIP-53 are equivalent to those required 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-4~~8~~9, 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.

**<u>XI.M9 BWR Vessel Internals</u>**

XI.M9-1 See General Comment G-2. Revise to state that any edition of the ASME code that has been endorsed by NRC under 10 CFR 50.55a is acceptable.

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## 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

…

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 (2001 edition[7] including the 2002 and 2003 Addenda). An applicant may use the guidelines of BWRVIP-62 for inspection relief for vessel internal components with hydrogen water chemistry provided such relief is submitted under the provisions of 10 CFR 50.55a and approved by the staff.

…

7 An applicant may rely on a different version of the ASME Code~~, but should justify such use. An applicant may wish to refer to the SOC for an update of 10 CFR § 50.55a to justify use of a more recent edition of the Code~~. <u>An edition of the ASME Code that has been endorsed by the NRC under 10 CFR 50.55a is acceptable and is considered consistent with NUREG-1801</u>.

**XI.M10 Boric Acid Corrosion**

XI.M10-1 Element 2 - See General Comment G-1. Program includes design considerations.

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## XI.M10 BORIC ACID CORROSION

### Program Description

The program relies in part on implementation of recommendations in 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. Potential improvements to boric acid corrosion programs have been identified as a result of recent operating experience with cracking of certain nickel alloy pressure boundary components (NRC Regulatory Issue Summary 2003-013).

Borated water leakage from piping and components that are outside the scope of the program established in response to GL 88-05 may affect structures and components that are subject to aging management review. Therefore, the scope of the monitoring and inspections of this program includes all components that contain borated water that are in proximity to structures and components that are subject to aging management review. The scope of the evaluations, assessments and corrective actions include all observed leakage sources and the affected structures and components.

Borated water leakage may be discovered by activities other than those established specifically to detect such leakage. Therefore, the program includes provisions for triggering evaluations and assessments when leakage is discovered by other activities.

### Evaluation and Technical Basis

…

2. ***Preventative 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.~~

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**<u>XI.M12 Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS)</u>**

XI.M12-1 See General Comment G-3. Reference to specific sections of the ASME Code in Elements 6 and 7 indicate a specific code edition. Element 6 is accurate but is susceptible to being affected by future revisions of the Code. In Element 7, repair and replacement are now both in IWA-4000, IWB-4000, and IWC-4000.

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## 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.…

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

…

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 submerged arc welds (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~~ <u>and replacement are</u> performed in conformance with IWA-4000 and IWB-4000 or IWC-4000~~, and replacement 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 to address the corrective actions.

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**<u>XI.M13 Thermal Aging And Neutron Irradiation Embrittlement Of Cast Austenitic Stainless Steel (CASS)</u>**

XI.M13-1 See General Comment G-3. Reference to specific sections of the ASME Code in Elements 6 and 7 indicate a specific code edition. Element 6 is accurate but is susceptible to being affected by future revisions of the Code. In Element 7, repair and replacement are now both in IWA-4000, IWB-4000, and IWC-4000.

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## 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 and neutron irradiation embrittlement 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

…

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~~ <u>and replacement are</u> performed in conformance with IWA-4000 and IWB-4000~~, and replacement 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 to address the corrective actions.

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## XI.M17 Flow-Accelerated Corrosion

XI.M17-1. Element 4 states that ultrasonic and radiographic testing is used to detect wall thinning. Some programs use ultrasonic only. There is no technical basis for requiring that radiographic testing is included in the program in addition to ultrasonic testing. Change "Ultrasonic and radiographic testing…" to "Ultrasonic or radiographic testing…".

XI.M17-2. See General Comment G-4. Reference to specific revision of an industry standard in Element 1 should be revised to allow later revisions.

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### 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. NSAC-202L-R2 (April 1999 or later revisions) 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.

…

4. ***Detection of Aging Effects:*** Degradation of piping and components occurs by wall thinning. The inspection program delineated in NSAC-202L-R2 consists of identification of susceptible locations as indicated by operating conditions or special considerations. Ultrasonic ~~and~~ or 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.

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**XI.M18 Bolting Integrity**

XI.M18-1. Element 1 includes "The staff's recommendations and guidelines for comprehensive bolting integrity programs that encompass all safety-related bolting are delineated in NUREG-1339, which include the criteria established in the 1995 edition through the 1996 addenda of ASME Code Section XI. The industry's technical basis for the program for safety-related bolting and guidelines for material selection and testing, bolting preload control, 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." This should be removed from the scope and added to the general description. This passage includes general statements of fact that do not add to the definition of the program scope. Applicants have called for plant specific exception if they are not under the stated version of the Code. This creates an unnecessary exception. Describing the content of industry documents should not be included under the scope of the program. This is not in accordance with SRP, Appendix A.1.

XI.M18-2. Element 5, Monitoring and trending, states, "If bolting connections 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 biweekly or weekly."
This is causing some plants to unnecessarily cite exception to the AMP. Inspection frequency is determined through evaluation under site-specific corrective action programs. A prescriptive frequency should not be specified without knowledge of the circumstances surrounding the leak and its potential consequences.

XI.M18-3 See General Comment G-3. The text correctly references the 1995 edition of the ASME Code, but the reference section lists the 2001 edition of the ASME Code. Change the reference to 1995 edition of the ASME Code, through 1996 addenda. This ASME Code edition does not change with changes to a plant's ISI plan or changes to 10 CFR 50.55a.

XI.M18-4 This program includes structural bolting and permits the use of XI.S6, Structures Monitoring Program or XI.S3, ASME Section XI Subsection IWF. Torque/tension testing specified by this program for structural bolting (Structural and ASME Class 1 Component Supports) are not required by the SMP. XI.S3, ASME Section XI Subsection IWF inspection requirements apply to structural bolting for ASME Class 1 Component Supports. Structural bolting should be deleted from XI.M18. This change will require revision of NUREG-1801 lines III.B1.1-3 and IIIB1.1-4 for high strength bolting for NSSS component supports.

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**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 comprehensive bolting maintenance, as delineated in 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. The program also includes preventive measures to preclude or minimize loss of preload and cracking.

Other aging management programs, such as XI.M1, “ASME Section XI Inservice Inspection (ISI) Subsections IWB, IW C, and IWD,” ~~and~~ XI.S3, “ASME Section XI Subsection IWF” also manage inspection of safety-related bolting and supplement this bolting integrity program. XI.S6, “Structures Monitoring,” or XI.S3, “ASME Section XI Subsection IWF” is used for managing structural bolting.

The staff’s recommendations and guidelines for comprehensive bolting integrity programs that encompass all safety-related bolting are delineated in NUREG-1339, which include the criteria established in the 1995 edition through the 1996 addenda of ASME Code Section XI. The industry’s technical basis for the program for safety-related bolting and guidelines for material selection and testing, bolting preload control, 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.

**Evaluation and Technical Basis**

1. ***Scope of Program:*** This program covers bolting within the scope of license renewal, including: 1) safety-related bolting, 2) bolting for nuclear steam supply system (NSSS) component supports, and 3) bolting for other pressure retaining components, including nonsafety-related bolting~~, and 4) structural bolting (actual measured yield strength ³ 150 ksi)~~. The aging management of reactor head closure studs is addressed by XI.M3~~,~~ and is not included in this program.~~ The staff’s recommendations and guidelines for comprehensive bolting integrity programs that encompass all safety-related bolting are delineated in NUREG-1339, which include the criteria established in the 1995 edition through the 1996 addenda of ASME Code Section XI. The industry’s technical basis for the program for safety-related bolting and guidelines for material selection and testing, bolting preload control, 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.~~

…

3. ***Parameters Monitored/Inspected:*** This program monitors the effects of aging on the intended function of bolting. Specifically, bolting for safety-related pressure retaining components is inspected for leakage, loss of material, cracking, and loss of preload/loss of prestress. Bolting for other pressure retaining components is inspected for signs of leakage.

   ~~High strength bolts (actual yield strength >150 ksi) used in NSSS component supports are monitored for cracking. Structural bolts and fasteners are inspected for indication of potential problems including loss of material, cracking, loss of coating integrity, and obvious signs of corrosion, rust, etc.~~

4. ***Detection of Aging Effects:*** Inspection requirements are in accordance with the ASME Section XI, Tables IWB 2500-1, IWC 2500-1 and IWD 2500-1 editions endorsed in 10 CFR

50.55a(b)(2) and the recommendations of EPRI NP-5769. For Class 1 components, Table IWB 2500-1, Examination Category B-G-1, for bolts greater than 2-inches in diameter, specifies volumetric examination of studs and bolts and visual VT-1 examination of surfaces of nuts, washers, bushings, and flanges. Examination Category B-G-2, for bolts 2-inches 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 C-D, for bolts greater than 2-inches in diameter, requires volumetric examination of studs and bolts. Examination Categories B-P, C-H, and D-B require visual examination (IWA-5240) during system leakage testing of all pressure-retaining Class 1, 2 and 3 components, according to Tables IWB 2500-1, IWC 2500-1, and IWD 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 Tables IWB 2500-1, IWC 2500-1, and IWD 2500-1, combined with periodic system walkdowns, assure detection of leakage before the leakage becomes excessive.

For other pressure retaining bolting, periodic system walkdowns assure detection of leakage before the leakage becomes excessive.

~~High strength structural bolts and fasteners (actual yield strength 150 ksi) for NSSS component supports, may be subject to stress corrosion cracking (SCC). For this type of high strength structural bolts that are potentially subjected to SCC, in sizes greater than 1-inch nominal diameter, volumetric examination comparable to that of Examination Category B-G-1 is required in addition to visual examination. This requirement may be waived with adequate plant-specific justification. Structural bolts and fasteners (actual yield strength < 150 ksi) both inside and outside containment are inspected by visual inspection (e.g., Structures Monitoring Program or equivalent). In addition to visual and volumetric examination, degradation of these bolts and fasteners may be detected and measured by removing the bolt/fastener, a proof test by tension or torquing, in situ ultrasonic tests, or a hammer test. If these bolts and fasteners are found cracked and/or corroded, a closer inspection is performed to assess extent of corrosion. An appropriate technique is selected on the basis of the bolting application and the applicable code.~~

5. ***Monitoring and Trending:*** The inspection schedules of ASME Section XI are effective and ensure timely detection of applicable aging effects. If a bolting connection~~s~~ 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 biweekly or weekly.~~ evaluation under the corrective action program determines appropriate corrective actions.

6. ***Acceptance Criteria:*** Any indications of aging effects in ASME pressure retaining bolting are evaluated in accordance with Section XI of the ASME Code. For other pressure retaining bolting and~~,~~ NSSS component support bolting ~~and structural bolting~~, indications of aging should be dispositioned in accordance with the corrective action process.

7. ***Corrective Actions:*** Replacement of ASME pressure retaining bolting is performed in accordance with appropriate requirements of Section XI of the ASME Code, as subject to the additional guidelines and recommendations of EPRI NP-5769. Replacement of other pressure retaining bolting (i.e., non-Class 1 bolting) ~~and disposition of degraded structural bolting~~ is performed in accordance with the guidelines and recommendations of EPRI TR-104213. Replacement of NSSS component support bolting is performed in accordance with EPRI NP-5769. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the corrective actions.

…

**References**

10 CFR Part 50, Appendix B, Quality Assurance Criteria for Nuclear Power Plants, Office of the Federal Register, National Archives and Records Administration, 2005.

10 CFR 50.55a, Codes and Standards, Office of the Federal Register, National Archives and Records Administration, 2005.

ASME Section XI, Rules for Inservice Inspection of Nuclear Power Plant Components, ASME Boiler and Pressure Vessel Code, ~~2001~~ 1995 edition ~~including the 2002 and 2003~~ through 1996 Addenda, American Society of Mechanical Engineers, New York, NY.

…

## XI.M19 Steam Generator Tube Integrity

XI.M19-1 See General Comment G-4. Reference to specific revision of an industry standard should be revised to allow later revisions or revisions to which a plant is committed.

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### XI.M19 STEAM GENERATOR TUBE INTEGRITY

#### Program Description

The steam generator tube integrity program is applicable to managing the aging of steam generator tubes, plugs, sleeves and tube supports.

The steam generator tube integrity program is applicable to managing the aging of steam generator tubes, plugs, sleeves and tube supports. Mill annealed alloy 600 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. The dominant degradation mode at this time for thermally treated alloy 600 and 690 tubes is wear. Nondestructive examination (NDE) techniques are used to inspect all tubing materials and sleeves to identify tubes with degradation that may need to be removed from service or repaired in accordance with plant technical specifications. In addition, operational leakage limits are included to ensure that, should substantial tube leakage develop, prompt action is taken. 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, frequency, and acceptance criteria for the plugging and repair of flawed tubes. 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 plant technical specifications. In addition to flaw acceptance (or plugging/repair) criteria, the technical specifications also specify acceptable tube repair methods (e.g., plugging and/or sleeving). Plants may also apply for changes in their technical specifications to provide an alternate repair criteria for SG degradation management.

In addition to plant technical specifications, 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~~.~~,” Rev. 1 (or later revision). This program references a number of industry guidelines and incorporates a balance of prevention, inspection, evaluation, repair, and leakage monitoring measures. 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. Steam generator tube integrity can be affected by degradation of SG plugs, sleeves and tube supports. Therefore, these components are also addressed by this aging management program.

…

**Evaluation and Technical Basis**

1. ***Scope of Program:*** The scope of the program is specific to SG tubes, plugs, sleeves and tube supports. The program includes preventive measures to mitigate degradation related to corrosion phenomena, assessment of degradation mechanisms, inservice inspection (ISI) of steam generator tubes, plugs, sleeves, and tube supports 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 and sleeve inspection scope and frequency, plugging or repair, and leakage monitoring are in accordance with the plant technical specifications and the licensee's SG degradation management program implemented in accordance with NEI 97-06. Plug inspection scope and frequency, plugging or repair, and leakage monitoring are in accordance with the licensee's SG degradation management program implemented in accordance with NEI 97-06. Lastly, tube support plate inspection scope and frequency are in accordance with the licensee's SG degradation management program implemented in accordance with NEI 97-06 as well as the program described in the licensee's response to GL 97-06.

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 wear degradation. The water chemistry program for PWRs relies on monitoring and control of ~~reactor~~ water chemistry based on ~~the~~ EPRI guidelines~~ in TR-05714 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.

**…**

**References**

…

NEI 97-06, Rev. 1, Steam Generator Program Guidelines, Nuclear Energy Institute, January 2001.

**<u>XI.M20 Open-Cycle Cooling Water System</u>**

XI.M20-1. See General Comment G-1. OCCW System Program implies that all components are lined or coated. Many components are not lined. This is a function of the design and is not within control of the program. OCCW system programs are essentially the same regardless of linings and coatings. If a component is lined, inspections monitor the condition of the lining. If not lined, inspections monitor the condition of the base material. Only a change to Element 2 is needed.

XI.M20-2 Condition and performance monitoring is not a preventive action. Delete from Preventive Actions.

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## 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 walk down 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 <u>may be</u> 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, elastomers, 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.

…

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.

…

**XI.M21 Closed-Cycle Cooling Water System**

XI.M21-1. Program description of the AMP references EPRI report TR-107396 for testing. Testing is not well-defined in EPRI report and does not manage effects of aging except for active components. This is inconsistent with other aging management programs that typically do not rely on flow measurement for managing the effects of aging. Flow is provided by active components that are subject to testing under the maintenance rule as explained in the statement of considerations for the license renewal rule.

XI.M21-2 See General Comment G-4. Reference to specific revision of an industry standard should be revised to allow later revisions or revisions to which a plant is committed.

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## XI.M21 CLOSED-CYCLE COOLING WATER SYSTEM

### Program Description

The program includes (a) preventive measures to minimize corrosion and stress corrosion cracking (SCC) and (b) ~~testing and~~ inspection to monitor the effects of corrosion and SCC on the intended function of the component. The program relies on maintenance of system corrosion inhibitor concentrations within the specified limits of Electric Power Research Institute (EPRI) TR-107396 to minimize corrosion and SCC. (Later revisions of this EPRI guideline are acceptable.) ~~Non-chemistry monitoring techniques such as testing and i~~Inspections in accordance with guidance in EPRI TR-107396 for closed-cycle cooling water (CCCW) systems provide one acceptable method to ~~evaluate system and component performance~~confirm effectiveness of chemistry controls. These measures will ensure that the intended functions of the CCCW system and components serviced by the CCCW system are not compromised by aging.

### Evaluation and Technical Basis

…

2. ***Preventive Actions:*** The program relies on the use of appropriate materials, lining, or coating to protect the underlying metal surfaces and maintains system corrosion inhibitor concentrations within the specified limits of EPRI TR-107396 to minimize corrosion and SCC. 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 as well as SCC.

3. ***Parameters Monitored/Inspected:*** The program monitors chemistry parameters in accordance with EPRI TR-107396 for CCCW systems. The aging management program ~~monitors~~ inspects for the effects of corrosion and SCC ~~by testing and inspection~~ in accordance with guidance in EPRI TR- 107396 to evaluate system and component condition. ~~For pumps, the parameters monitored include flow, discharge pressures, and suction pressures. For heat exchangers, the parameters monitored include flow, inlet and outlet temperatures, and differential pressure.~~Inspections monitor surface condition.

…

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 and the type of chemical treatment. In accordance with EPRI TR-107396, internal visual inspections ~~and performance/functional tests are to be performed periodically to demonstrate system operability and~~ confirm the effectiveness of the program. Tests to evaluate heat removal capability of the system and degradation of system components may also be used. The testing intervals should be established based on plant-specific considerations such as system conditions, trending, and past operating experience, and may be adjusted based on the results of a reliability analysis, type of service, frequency of operation, or age of components and systems.

…

## XI.M22 Boraflex Monitoring

XI.M22-1 The same actions are listed under preventive actions and under detection of aging effects. This is not consistent with the SRP guidance on how to describe AMP elements. Detection of aging effects for boraflex degradation should not be the same as preventive actions.

XI.M22-2 This AMP specifies blackness testing and inspection of coupons. However, some plants do not do blackness testing but use BADGER testing. Also, some plants do not have test coupons. Revise the Boraflex Monitoring AMP to allow either blackness testing or BADGER testing, depending on the testing actually performed at the plant. Also, revise this AMP to delete the requirement for coupon testing unless coupon testing is part of the overall testing program implemented at the plant to monitor the condition of Boraflex.

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### 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~~2) either (a) performing neutron attenuation testing, called blackness testing, to determine gap formation in Boraflex panels; or (b) 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

…

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.~~The program includes no preventive actions.

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 with a predictive code. This ~~is~~ can be supplemented with detection of gaps through blackness testing and periodic verification of boron loss through areal density measurement techniques such as the BADGER device.

…

**XI.M23 Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems**

XI.M23-1 Detection of Aging Effects and Acceptance Criteria specify visual inspection for crane rails and girders. Parameters monitored/inspected should specify what parameters are to be monitored consistent with these sections.

XI.M23-2 Parameters Monitored/Inspected states, "The program evaluates the effectiveness of the maintenance monitoring program and the effects of past and future usage on the structural reliability of cranes." Nothing else in the program addresses this. It is not clear what this is intended to mean. This sentence should be deleted and corresponding words in the program description should be revised.

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## 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.

~~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 m~~Many of the systems and components of these cranes perform an intended function with moving parts or with a change in configuration, or are 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

…

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.~~ Surface condition is monitored to ensure that loss of material is not occurring due to corrosion or wear.

…

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**<u>XI.M24 Compressed Air Monitoring</u>**

XI.M24-1 Parameters Monitored states "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." Emergency procedures and training don't seem appropriate for aging management programs. This is unlike any other AMP in NUREG-1801. This section fails to identify parameters monitored.

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## XI.M24 COMPRESSED AIR MONITORING

### Program Description

The program consists of inspection, monitoring, and testing of the entire system. This includes (a) frequent leak testing of valves, piping, and other system components, especially those made of carbon steel and stainless 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 owner's response to 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 components. The EPRI then issued TR-108147, which addresses maintenance of the latest compressors and other instrument air system components 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

…

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. <u>Inspections monitor surface condition for signs of loss of material, such as corrosion.</u>

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**<u>XI.M25 BWR Reactor Water Cleanup System</u>**

XI.M25-1 See General Comment G-2. (The text and the reference is to 2001 edition of the ASME Code, but both should refer to 1986 edition.) The 1986 edition of the ASME Code is the code edition referenced by GL 88-01. This ASME Code edition does not change with changes to a plant's ISI plan or changes to 10 CFR 50.55a. However, use of a later Code should be acceptable.

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## 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 cracking due to SCC or IGSCC.

### Evaluation and Technical Basis

…

**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 (~~2001~~ <u>1986</u> edition~~[8] including the 2002 and 2003 Addenda~~). <u>Use of a later NRC-approved version of the ASME Code is acceptable.</u>

…

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~~8 An applicant may rely on a different version of the ASME Code, but should justify such use. An applicant may wish to refer to the SOC for an update of 10 CFR § 50.55a to justify use of a more recent edition of the Code.~~

### References

…

ASME Section XI, *Rules for Inservice Inspection of Nuclear Power Plant Components*, ASME Boiler and Pressure Vessel Code, ~~2001~~ <u>1986</u> edition~~ including the 2002 and 2003 Addenda~~, American Society of Mechanical Engineers, New York, NY.

**<u>XI.M26 Fire Protection</u>**

XI.M26-1. Fire barrier penetration seal inspection states approximately 10%. The intent is unclear. Is 7% approximately 10%? Seems the real concern should be a statistically valid sample size, which with numbers of seals in the range of thousands will be much less than 10%. As written, it is too vague. Exceptions have been required even though programs meet established regulatory requirements. AMP acceptability should be based on compliance with existing current term requirements for fire seals.

XI.M26-2. Acceptance criteria attribute says no corrosion is acceptable in the fuel supply line. The program has no activity that can confirm this acceptance criterion. This part of the acceptance criteria attribute should be deleted.

XI.M26-3 Element 4, "Detection of Aging Effects," states that Halon/CO2 system visual inspection detects any sign of degradation, such as corrosion, mechanical damage, or damage to dampers. Mechanical damage and damage to dampers are not aging effects. While existing programs likely do include inspections for damage, this should not be included in the NUREG-1801 aging management program.

XI.M26-4 Operating experience element includes "electrical racing way fire barrier..". The correct terminology is "electrical raceway fire barrier…".

XI.M26-5 Halon and $CO_2$ systems are highly reliable and current testing practices (either in the Fire Protection Plan or plant Technical Specifications) have proven to be adequate for the current license term. The frequencies for inspections specified in NUREG-1801 for Halon and $CO_2$ system testing are much shorter than often practiced. More frequent testing will not increase reliability of these systems.
Revise Element 4 to allow frequencies based on either of two sources. The first is the plant's Technical Specifications. The second would be industry standards for Fire Protection. The NUREG-1801 should not include the frequency specifically, but should state that the frequency in the Technical Specifications or Fire Protection standard is acceptable. The reference should not be to a specific revision of the standard, allowing for changes based on operating experience through an industry consensus process.

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## 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 testing of the halon/carbon dioxide ($CO_2$) fire suppression system.

**Evaluation and Technical Basis**

…

3. ***Parameters Monitored/Inspected:*** Visual inspection ~~of approximately 10%~~ of each type of penetration seal is performed during walkdowns carried out at least once every refueling outage per existing plant technical requirements. 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. Fire-rated doors are visually inspected on a plant-specific interval to verify the integrity of door surfaces and for clearances. The plant-specific inspection intervals are to be determined by engineering evaluation to detect degradation of the fire doors prior to the loss of intended function.

   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 detection of any degradation of the fuel supply line. The periodic visual inspection and function tests are ~~is~~ performed ~~at least once every six months~~ periodically to examine the signs of degradation of the halon/$CO_2$ fire suppression system. Material conditions that may affect the performance of the system, such as corrosion, mechanical damage, or damage to dampers, are observed during these tests.

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 by fire protection qualified inspectors ~~of approximately 10%~~ of each type of seal in walkdowns is performed at least once every refueling cycle per existing plant technical requirements. If any sign of degradation is detected within that sample, the scope of the inspection is expanded to include additional seals. Visual inspection by fire protection qualified inspectors of the fire barrier walls, ceilings, and floors, performed in walkdowns at least once every refueling outage ensures timely detection of concrete cracking, spalling, and loss of material. Visual inspection by fire protection qualified inspectors detects any sign of degradation of the fire door such as wear and missing parts. 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.

   Visual inspections of the halon/$CO_2$ fire suppression system detect any sign of added 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/$CO_2$ fire suppression system before the loss of the component intended function. Frequency of this inspection is dictated by the site Technical Specifications or industry standards for Fire Protection.

   …

6. ***Acceptance Criteria:*** Inspection results are acceptable if there are no visual indications (outside those allowed by approved penetration seal configurations) 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, a~~Any signs of corrosion and mechanical damage of the halon/$CO_2$ fire suppression system are not acceptable.

…

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 race~~ing ~~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.

**<u>XI. M27 Fire Water System</u>**

XI.M27-1. Element 4 specifies AMP activities for fire hose and gaskets. Table 2.1-3 of SRP indicates such components are not subject to aging management review.

XI.M27-2. The program description states, "A sample of sprinkler heads is to be inspected by using the guidance of NFPA 25, "Inspection, Testing and Maintenance of Water-Based Fire Protection Systems" (1998 Edition), Section 2-3.1.1, or NFPA 25 (2002 Edition), Section 5.3.1.1.1. This NFPA section states '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." This is different from Detection of Aging Effects. It's not clear what is expected. NUREG-1801 says inspection; the NFPA quote says field service testing.

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## XI. M27 FIRE WATER SYSTEM

### Program Description

This aging management program (AMP) 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.

A sample of sprinkler heads is to be ~~inspected~~ <u>tested</u> by using the guidance of NFPA 25 "Inspection, Testing and Maintenance of Water-Based Fire Protection Systems" (1998 Edition), Section 2- 3.1.1, or NFPA 25 (2002 Edition), Section 5.3.1.1.1. This NFPA section states "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.

The fire protection system piping is to be subjected to required flow testing in accordance with guidance in NFPA 25 to verify design pressure or evaluated for wall thickness (e.g., nonintrusive volumetric testing or plant maintenance visual inspections) to ensure that aging effects are managed and that wall thickness is within acceptable limits. These inspections are performed before the end of the current operating term and at plant-specific intervals thereafter during the period of extended operation. The plant-specific inspection intervals are to be determined by engineering evaluation of the fire protection piping to ensure that degradation will be detected before the loss of intended function. The purpose of the full flow testing and wall thickness evaluations is to ensure that corrosion, MIC, or biofouling is managed such that the system function is maintained.

### Evaluation and Technical Basis

…

4. ***Detection of Aging Effects:*** Fire protection system testing is performed to assure that the system functions by maintaining required operating pressures. Wall thickness evaluations of fire protection piping are performed on system components using non-intrusive techniques (e.g., volumetric testing) to identify evidence of loss of material due to corrosion. These inspections are performed before the end of the current operating term and at plant-specific intervals thereafter during the period of extended operation. As an alternative to non-intrusive testing, the plant maintenance process may include a visual inspection of the internal surface of the fire protection piping upon each entry to the system for routine or corrective maintenance, as long as it can be demonstrated that inspections are performed (based on past maintenance history) on a representative number of locations on a reasonable basis. These inspections must be capable of evaluating (1) wall thickness to ensure against catastrophic failure and (2) the inner diameter of the piping as it applies to the design flow of the fire protection system. If the environmental and material conditions that exist on the interior surface of the below grade fire protection piping are similar to the conditions that exist within the above grade fire protection piping, the results of the inspections of the above grade fire protection piping can be extrapolated to evaluate the condition of below grade fire protection piping. If not, additional inspection activities are needed to ensure that the intended function of below grade fire protection piping will be maintained consistent with the current licensing basis for the period of extended operation. Continuous system pressure monitoring, system flow testing, and wall thickness evaluations of piping are effective means to ensure that corrosion and biofouling are not occurring and the system's intended function is maintained.

   General requirements of existing fire protection programs include testing and maintenance of fire detection and protection systems and surveillance procedures to ensure that fire detectors, as well as fire protection systems and components are operable.

   Visual inspection of yard fire hydrants performed annually in accordance with NFPA 25 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 heads are ~~inspected~~ tested before the end of the 50-year sprinkler head service life and at 10-year intervals thereafter during the extended period of operation to ensure that signs of degradation, such as corrosion, are detected in a timely manner.

## XI. M29 Aboveground Steel Tanks

XI.M29-1. This AMP discusses system or plant walkdowns. This is actually crediting visual inspections that may or may not be done in conjunction with system walkdowns. A more correct terminology is to say inspections.

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## XI. M29 ABOVEGROUND STEEL TANKS

### Program Description

The program includes preventive measures to mitigate corrosion by protecting the external surface of steel tanks with paint or coatings in accordance with standard industry practice. The program also relies on periodic ~~system walkdowns~~ inspections 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~~ inspections to manage the effects of corrosion on the intended function of these tanks. ~~Plant walkdowns~~Inspections 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 AMP utilizes periodic ~~plant system walkdowns~~inspections to monitor surface condition for degradation of coatings, sealants, and caulking because it is a condition directly related to the potential loss of material~~s~~.

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~~ inspections 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, thus, 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~~inspections 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.

…

**XI.M30 Fuel Oil Chemistry**

XI.M30-1. Element 1, Scope of Program, says it is limited to tanks, while many other components refer to the same program.

XI.M30-2. Element 2, Preventive Actions, says biocides are added. Corrective actions say add biocides if biological activity is detected. This is inconsistent. Preventive Actions should be revised to say that biocides are added, if necessary, based on indications of biological activity.

XI.M30-3. Element 3, Parameters Monitored/Inspected, says the AMP monitors fuel oil quality and the levels of water and microbiological organisms in the fuel oil. Levels of water and microbiological organisms are clearly parameters to be monitored and are related to the applicable effects of aging. Fuel oil quality, on the other hand, is rather ambiguous and facets of this may be primarily related to quality for combustion and not to the effects of aging.

XI.M30-4. Elements 3 (unrevised) states, “For determination of particulates, modified ASTM D2276, Method A, is used. The modification consists of using a filter with a pore size of 3.0 mm, instead of 0.8 mm. These are the principal parameters relevant to tank structural integrity.” Element 6 (unrevised), Acceptance Criteria, repeats the reference to the modified ASTM.
The modification is likely based on a plant-specific program that is not a requirement. Many plants take exception because they use the more conservative unmodified process. NUREG-1801 should allow the more conservative approach. In fact, the statement that “these are the principal parameters relevant to tank structural integrity” would seem to apply to the levels of water and microbiological organisms rather than particulates.

XI.M30-5. The discussion of methods under parameters monitored (Element 3) should be under detection of aging effects (Element 4) in accordance with the SRP, Appendix A.1. Testing conducted ASTM D1796 gives quantitative results, whereas D2709 testing gives only pass-fail results. Either method would be acceptable for detecting aging effects.

XI.M30-6. Element 6 references standards that provide guidance for how to do sampling. This guidance belongs under Detection of Aging Effects. The referenced standards do not provide acceptance criteria related to the parameters monitored under the program. The discussion of parameters monitored is largely a repeat of the methods for detection of aging effects. Acceptance criteria should be provided.

XI.M30-7 NUREG-1801 is based on standards in Standard Tech Specs. Some plants do not use Standard Tech Spec, but have Tech Spec requirements based on different ASTM standards than in NUREG-1801. Revise to allow standards based on a plant’s Tech Specs.

## 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 insignificance of an aging effect. Fuel oil quality is maintained by monitoring and controlling fuel oil contamination in accordance with the plant's technical specifications ~~and~~ or the guidelines of the American Society for Testing Materials (ASTM) Standards D 1796, D 2276, D 2709, D6217, and D 4057. ~~Some plants' technical specifications may be based on different ASTM standards.~~ 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's 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 of components exposed to fuel oil in accordance with the plant's technical specifications ( i.e., NUREG- 1430, NUREG-1431, NUREG-1432, NUREG-1433) on fuel oil purity ~~and~~ or the guidelines of ASTM Standards D1796, D2276, D2709, D6217, and D4057. The program serves to reduce the potential of SSC 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. Biocides are added if necessary based on indications of biological activity. 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 of components exposed to fuel oil. ~~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 mm, instead of 0.8 mm.~~ These are ~~the~~ principal parameters relevant to ~~tank structural~~fuel oil system integrity.

4. ***Detection of Aging Effects:*** Degradation of ~~the~~ components exposed to 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. ASTM Standard D 4057 is used for guidance on oil sampling. ASTM Standards D 1796 or ~~and~~ D 2709 are used for determination of water and sediment contamination in diesel fuel. For determination of particulates, ASTM D 2276,

Method A, is acceptable using a filter with a pore size of 3.0 mm or 0.8 mm. Use of ASTM standards particular to a plant's technical specifications is acceptable. Compliance with these diesel fuel oil standards ~~in item 3, above,~~ and periodic multilevel sampling provide assurance that fuel oil contaminants are below unacceptable 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.

…

**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 contamination in diesel fuel. ASTM D 6217 and Modified D 2276, Method A are used for guidance for determination of particulates. The modification to D 2276 consists of using a filter with a pore size of 3.0 μm, instead of 0.8 μm.~~ Quality of the fuel oil will be maintained in accordance with the plant's technical specifications.

…

**XI.M31 Reactor Vessel Surveillance**

XI.M31-1. Item 4 specifies that pulled and tested capsules are placed in storage. This creates an unnecessary expense that may not be warranted if a plant has ample capsules remaining for future use.

XI.M31-2. Item 7 states, "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." Since XI.M31 is the AMP for reactor vessel neutron embrittlement, the last part of this sentence should be removed to avoid confusion.

XI.M31-3. This program should be converted into the 10-element format of NUREG-1801. Text in italics has been relocated in the discussion. Changes to this text are shown as usual.

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## 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 > 1 MeV), 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.

All capsules in the reactor vessel that are removed and tested must meet the test procedures and reporting requirements of ASTM E 185-82, to the extent practicable, for the configuration of the specimens in the capsule. Any changes to the capsule withdrawal schedule, including spare capsules, must be approved by the Nuclear Regulatory Commission (NRC) prior to implementation. Untested capsules placed in storage must be maintained for future insertion.

~~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 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 is 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 because 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. 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 program is 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.

1. ***Scope of Program:*** The program includes all reactor vessel beltline materials as defined by 10 CFR 50 Appendix G, Section II.F. *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.*

2. ***Preventive Actions:*** The program is a surveillance program; no preventive actions are identified.

3. ***Parameters Monitored/Inspected:*** The program monitors reduction of fracture toughness of reactor vessel beltline materials due to neutron irradiation embrittlement.

4. ***Detection of Aging Effects:*** R~~r~~*eactor vessel beltline materials* will *be monitored by a surveillance program* in which surveillance capsules are withdrawn from the reactor vessel and tested ~~*to meet*~~in accordance with *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.*

~~*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 because 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. 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.*

~~*4.*~~ If a plant does not have ample capsules remaining for future use, ~~*A*~~*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.)*

~~*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*~~*.*

**5. *Monitoring and Trending:*** ~~1.~~ *The extent of reactor vessel embrittlement for upper-shelf energy and pressure-temperature limits for 60 years is projected in accordance with the 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 is 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.*

*The reactor vessel monitoring program provides that, if future plant operations exceed these 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~~ capsules with a projected fluence exceeding the 60-year fluence at the end of 40 years;* (2) other standby capsules; *(~~2~~3) reconstitution of ~~specimens from item 4, above~~*pulled and tested specimens previously placed in storage*; and/or (~~3~~4) capsules made from any available archival materials; or (~~4~~5) some combination of the ~~three~~*four *previous options. This program could be a plant-specific program or an integrated surveillance program.*

**6.** ***Acceptance Criteria:*** Reactor vessel embrittlement projections will comply with 10 CFR 50 Appendix G limits through the period of extended operation. $RT_{NDT}$ for material in the beltline will remain below screening criterion using end-of-life fluence.

Acceptable pressure-temperature curves for heatup and cooldown of the unit will be maintained in Technical Specifications. The operational EFPY shall not exceed the Technical Specification limits for the pressure-temperature curves.

**7.** ***Corrective Actions:*** Results of surveillance capsule testing will be incorporated into site operating limitations. Affected documents may include pressure-temperature (P-T) curves in the technical specifications, UFSAR, and operating procedures. If changes to the P-T curves are required, a licensing basis change shall be prepared and the revised curves shall be submitted to the NRC. A submittal is only required if the existing P-T curves are determined to be non-conservative.

If embrittlement projections drop below 50 ft-lbs, the margins of safety against fracture will be demonstrated to be equivalent to those of Appendix G of ASME Section XI.

If $RT_{NDT}$ for material in the beltline is projected to exceed the screening criterion using end of life fluence, a site may implement flux reduction programs that are reasonably practicable to avoid exceeding this criterion. If no reasonably practicable flux reduction program will avoid exceeding the screening criteria, the site will submit a safety analysis to determine actions to prevent potential failure of the reactor vessel as a result of postulated events if continued operation beyond the screening criterion is allowed.

If a capsule is not withdrawn as scheduled, the NRC will be notified and a revised withdrawal schedule will be submitted to the NRC.

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 to address the corrective actions, confirmation process, and administrative controls.

**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 to address the corrective actions, confirmation process, and administrative controls.

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

**10.** ***Operating Experience:*** *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.*

**References**

10 CFR Part 50, Appendix H, Reactor Vessel Material Surveillance Program Requirements, Office of the Federal Register, National Archives and Records Administration, 2005.

ASTM E-185, Standard Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels, American Society for Testing Materials, Philadelphia, PA. (Versions of ASTM E-185 to be used for the various aspects of the reactor vessel surveillance program are as specified in 10 CFR Part 50, Appendix.)

NRC Regulatory Guide 1.99, Rev. 2, Radiation Embrittlement of Reactor Vessel Materials, U.S. Nuclear Regulatory Commission, May 1988.

**XI.M32 One-Time Inspection**

XI.M32-1. Referring to table under Detection of Aging Effects, parameter monitored for loss of material is "wall thickness". This is certainly an indication of loss of material, but surface condition is a leading indicator of loss of material due to corrosion mechanisms. Surface condition can be monitored with the visual testing methods listed in the table, but visual is not always an effective technique to assess wall thickness. Surface condition should be listed under parameters monitored for those cases where visual inspection is recommended.

XI.M32-2 See General Comment G-3. Specific ASME code year is listed only in references. State that any edition of the ASME code endorsed by the NRC in 10CFR50.55a is acceptable.

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**XI.M32 ONE-TIME INSPECTION**

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4. ***Detection of Aging Effects:*** The inspection includes a representative sample of the system population, and, where practical, focuses on the bounding or lead components most susceptible to aging due to time in service, severity of operating conditions, and lowest design margin.

   The program will rely on established NDE techniques, including visual, ultrasonic, and surface techniques that are performed by qualified personnel following procedures consistent with the ASME Code and 10 CFR Part 50, Appendix B.

   The inspection and test techniques will have a demonstrated history of effectiveness in detecting the aging effect of concern. Typically, the one time inspections should be performed as indicated in the following table.

**Examples of Parameters Monitored or Inspected**
**And Aging Effect for Specific Structure or Component[9]**

| Aging Effect | Aging Mechanism | Parameter Monitored | Inspection Method[10] |
|---|---|---|---|
| Loss of Material | Crevice Corrosion | <u>Surface Condition</u><br>Wall Thickness | Visual (VT-1 or equivalent) and/or Volumetric (RT or UT) |
| Loss of Material | Galvanic Corrosion | <u>Surface Condition</u><br>Wall Thickness | Visual (VT-3 or equivalent) and/or Volumetric (RT or UT) |
| Loss of Material | General Corrosion | <u>Surface Condition</u><br>Wall Thickness | Visual (VT-3 or equivalent) and/or Volumetric (RT or UT) |

| Aging Effect | Aging Mechanism | Parameter Monitored | Inspection Method[10] |
|---|---|---|---|
| Loss of Material | MIC | Surface Condition Wall Thickness | Visual (VT-3 or equivalent) and/or Volumetric (RT or UT) |
| Loss of Material | Pitting Corrosion | Surface Condition Wall Thickness | Visual (VT-1 or equivalent) and/or Volumetric (RT or UT) |
| Loss of Material | Erosion | Surface Condition Wall Thickness | Visual (VT-3 or equivalent) and/or Volumetric (RT or UT) |
| Loss of Heat Transfer | Fouling | Surface Condition Tube Fouling | Visual (VT-3 or equivalent) or Enhanced VT-1 for CASS |
| Cracking | SCC or Cyclic Loading | Surface Condition Cracks | Enhanced Visual (VT-1 or equivalent) and/or Volumetric (RT or UT) |
| Loss of Preload | Thermal Effects, Gasket Creep and Self-loosening | Loosening of Components | Visual (VT-3 or equivalent) |

9 The examples provided in the table may not be appropriate for all relevant situations. If the applicant chooses to use an alternative to the recommendations in this table, a technical justification should be provided as an exception to this AMP. This exception should list the AMR line item component, examination technique, acceptance criteria, evaluation standard and a description of the justification.

10 Visual inspection may be used only when the inspection methodology examines the surface potentially experiencing the aging effect.

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**References**

10 CFR Part 50, Appendix B, Quality Assurance Criteria for Nuclear Power Plants, Office of the Federal Register, National Archives and Records Administration, 2005.

10 CFR 50.55a, Codes and Standards, Office of the Federal Register, National Archives and Records Administration, 2005.

ASME Section XI, Rules for Inservice Inspection of Nuclear Power Plant Components, ASME Boiler and Pressure Vessel Code, 2001 edition including the 2002 and 2003 Addenda, American Society of Mechanical Engineers, New York, NY  An applicant may rely on a different version of the ASME Code. An edition of the ASME Code that has been endorsed by the NRC under 10 CFR 50.55a is acceptable and is considered consistent with NUREG-1801.

## XI.M33 SELECTIVE LEACHING OF MATERIALS

XI.M33-1. Element 3, Parameters Monitored/Inspected, does not define the parameters monitored. It simply repeats elements from Detection of Aging Effects and discusses corrective actions. The entire program description should be revised to conform to the SRP description of what to include in the 10 elements.

XI.M33-2 NUREG-1801 specifies hardness testing, but other techniques have been accepted. Revise program to permit use of additional methods that have been accepted in license renewal SERs, such as scraping and chipping, and laboratory examinations.

XI.M33-3 Operating experience states that this is a new program. That is true for most applicants, but will not necessarily always be true in the future. This statement is unnecessary in the operating experience discussion. Delete first sentence of Operating Experience.

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## 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. Procedures other than hardness testing, such as laboratory examinations or scraping and chipping, are acceptable.

### Evaluation and Technical Basis

1. ***Scope of Program:*** This AMP determines the acceptability of the components that may be susceptible to selective leaching and assesses 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 visual inspection and hardness measurement (or other accepted method) of a selected set of sample 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 (or other accepted method) 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.~~Surface condition is monitored for indications of selective leaching, including dezincification. Hardness of the surface is monitored. Other accepted methods may be used in lieu of hardness testing or visual inspection.

4. ***Detection of Aging Effects:*** The one-time visual inspection and hardness measurement (or other accepted method) 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. Because selective leaching is a slow acting corrosion process, this measurement is performed ~~just before the beginning of the license renewal~~ within a ten-year period prior to entering the period of extended operation.

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~~require engineering evaluation before the affected components can be ~~qualified~~ accepted 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. Follow-up of unacceptable inspection findings includes expansion of the inspection sample size. 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 to address the corrective actions.

…

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

**<u>XI.M34 Buried Piping And Tanks Inspection</u>**

XI.M34-1. Scope states the program covers buried components that are within the scope of license renewal for the plant. The AMP should not define the scope in terms of specific components. That should be left to the AMR results. If a better program is selected to manage an aging effect, that should not create an automatic exception because the AMP scope definition is too restrictive. Example is fuel oil monitoring program that does periodic UT from inside of a tank to detect loss of material on outside. This is more effective than opportunistic external inspection under M34.

XI.M34-2 The NUREG-1801 program is only for steel tanks and piping, does not include stainless steel, aluminum, titanium, or other materials; credits coatings, which may not be used on some materials, such as stainless steel; specifies external inspections when UT from inside has been used frequently.

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## 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 ~~steel~~ piping and tanks. Gray cast iron~~, which is included under the definition of steel,~~ is also subject to a loss of material due to selective leaching, which is an aging effect managed under Chapter XI.M33, "Selective Leaching of Materials."

Preventive measures<u>, when used,</u> 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~~<u>for any reason</u>.

This program is ~~an~~ acceptable ~~option~~ to manage buried piping and tanks, except further evaluation is required for the program element/attributes of detection of aging effects (regarding inspection frequency) and operating experience.

### Evaluation and Technical Basis

1. ***Scope of Program:*** The program relies on preventive measures<u>,</u> such as coating~~,~~ <u>and</u> wrapping<u>,</u> and periodic inspection for loss of material caused by corrosion of the external surface of buried ~~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 i~~<u>I</u>nspections 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 <u>of susceptible materials</u> 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. Selecting materials that are resistant to the soil environment is an effective preventive action.

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 ~~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.

**…**

6. ***Acceptance Criteria:*** Any coating and wrapping degradation~~s~~ ~~are~~is reported and evaluated according to site corrective actions procedures.

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**XI.M35 One-Time Inspection of ASME Code Class 1 Small-Bore Piping**

XI.M35-1. See General Comment G-3 (specific ASME Code year listed in the references). Add reference to specific Code year to Program Description and then allow for later approved revisions.

XI.M35-2. See General Comment G-4 (specific year or revision to an industry standard or NRC standard is referenced in Element 1).

XI.M35-3. The NUREG-1801 Program Description for Program XI.M35 describes the program to include piping "less than or equal to NPS 4" with a reference to ASME Section XI, Table IWB-2500-1, Examination Category BJ. However, volumetric examinations are already required for piping equal to NPS 4" according to ASME Code. These examinations are included in the Inservice Inspection Program. Consistent with the Code, NUREG-1801 Item IV.C2-1 applies the One-Time Inspection of ASME Code Class 1 Small Bore Piping Program (XI.M35) only to Class 1 piping less than NPS 4". Based on this, we conclude that it is not the intent of NUREG-1801 Program XI.M35 to include NPS 4" pipe, and the One-Time Inspection – Small Bore Piping Program should include only small bore Class 1 piping < NPS 4". This affects Element #1.

XI.M35-4 Element 1 update EPRI report reference to EPRI Report 112657, "Revised Risk-informed Inservice Inspection Evaluation Procedure". EPRI Report 1000701 referenced by NUREG-1801 is an interim document. Guidelines for identifying piping susceptible to potential effects of thermal stratification or turbulent penetration that are provided in EPRI Report 1000701 are also provided in EPRI Report 112657.

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## XI.M35 ONE-TIME INSPECTION OF ASME CODE CLASS 1 SMALL-BORE PIPING

### Program Description

This program is applicable to small-bore ASME Code Class 1 piping and systems less than ~~or equal to~~ 4 inches nominal pipe size (NPS 4), which includes pipes, fittings, and branch connections. This program is based on ASME Section XI, 2001 edition[1] including the 2002 and 2003 Addenda, but later revisions approved by NRC are acceptable. According to Table IWB-2500-1, Examination Category B-J Item No. B9.21 of the current ASME code, for small-bore Class 1 piping, a surface examination should be included for piping less than or equal to NPS 4 and greater than or equal to NPS 1. Also, Examination Category B-P requires system leakage and hydrostatic tests. However, the staff believes that, for a one-time inspection to detect cracking resulting from thermal and mechanical loading or intergranular stress corrosion, the inspection should be a volumetric examination. This is to provide additional assurance that either aging of small-bore ASME Code Class 1 piping is not occurring or the aging is insignificant, such that an aging management program (AMP) is not warranted. This program is applicable only to plants that have not experienced cracking of ASME Code Class 1 small-bore piping resulting from stress corrosion or thermal and mechanical loading. Should evidence of significant aging be revealed by a one-time inspection or previous operating experience, periodic inspection will be proposed, as managed by a plant-specific AMP.

### Evaluation and Technical Basis

1. ***Scope of Program:*** This program is a one-time inspection of a sample of ASME Code Class 1 piping less than ~~or equal to~~ NPS 4. The program includes measures to verify that degradation is not occurring; thereby either confirming that there is no need to manage aging-related degradation or validate the effectiveness of any existing AMP for the period of extended operation. The one-time inspection program for ASME Code Class 1 small-bore piping includes locations that are susceptible to cracking. Guidelines for identifying piping susceptible to potential effects of thermal stratification or turbulent penetration are provided in ~~EPRI Report 1000701, "Interim Thermal Fatigue Management Guideline (MRP-24)," January 2001~~ EPRI Report 112657, “Revised Risk-informed Inservice Inspection Evaluation Procedure”. . Later revisions of the report are acceptable.

5. ***Monitoring and Trending:*** This is a one-time inspection to determine whether cracking in ASME Code Class 1 small-bore piping resulting from stress corrosion or thermal and mechanical loading is an issue. A one-time volumetric inspection is an acceptable method for confirming that cracking of ASME Code Class 1 small-bore piping, as a result of stress corrosion or thermal and mechanical loading, is not occurring in plants that have not experienced cracking due to these aging effects. However, evaluation of the inspection results may indicate the need for additional examinations, i.e., a plant-specific AMP, consistent with ASME Section XI, Subsection IWB. This inspection should be performed at a sufficient number of locations to assure an adequate sample. This number, or sample size, will be based on susceptibility, inspectability, dose considerations, operating experience, and limiting locations of the total population of ASME Code Class 1 small-bore piping locations.

[1]An applicant may rely on a different version of the ASME Code. An edition of the ASME Code that has been endorsed by the NRC under 10 CFR 50.55a is acceptable and is considered consistent with NUREG-1801.

6. ***Acceptance Criteria:*** If flaws or indications exceed the acceptance criteria of ASME Code, Section XI, Paragraph IWB-3400, they will be evaluated in accordance with ASME Code, Section XI, Paragraph IWB-3131, and additional examinations are performed in accordance with ASME Code, Section XI, Paragraph IWB-2430.

…

**References**

10 CFR Part 50, Appendix B, Quality Assurance Criteria for Nuclear Power Plants, Office of the Federal Register, National Archives and Records Administration, 2005.

10 CFR 50.55a, Codes and Standards, Office of the Federal Register, National Archives and Records Administration, 2005.

ASME Section XI, Rules for Inservice Inspection of Nuclear Power Plant Components, ASME Boiler and Pressure Vessel Code, 2001 edition including the 2002 and 2003 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.

EPRI Report 1000701, "Interim Thermal Fatigue Management Guideline (MRP-24)," January 2001 (ADAMS Accession No. ML010810162).

**XI.M36 External Surfaces Monitoring**

XI.M36-1. See General Comment G-5. Scope appears to be restricted to loss of material to steel components. There seems to be no reason not to include other metallic components in the program.

XI.M36-2. The scope section includes much discussion that duplicates discussion in Detection of Aging Effects. In accordance with SRP Appendix A.1 guidance, frequency of inspection should be addressed in Detection of Aging Effects, not in the scope of the program.

XI.M36-3 The extent of materials addressed in the Program Description and Element 1 Scoping, Element 3 Parameters Monitored and Element 4 Detection of Aging Effects has been expanded to include elastomers.

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## XI.M36 EXTERNAL SURFACES MONITORING

### Program Description

The External Surfaces Monitoring program is based on system inspections and walkdowns. This program consists of periodic visual inspections of ~~steel~~ metallic and elastomer components such as piping, piping components, ducting, elastomer HVAC components, and other components within the scope of license renewal and subject to AMR in order to manage aging effects. The program manages aging effects through visual inspection of external surfaces for evidence of material loss and change in material properties. When appropriate for the component and material, manipulation may be used to augment visual inspection to confirm the absence of elastomer hardening and loss of strength. Loss of material due to boric acid corrosion is managed by the Boric Acid Corrosion Program.

### Evaluation and Technical Basis

1. ***Scope of Program:*** This program visually inspects the external surface of in-scope components and monitors external surfaces of ~~steel~~ metallic components in systems within the scope of license renewal and subject to AMR for loss of material and leakage. This program also visually inspects the external surface of in-scope components and monitors the external surfaces of elastomer components in systems within the scope of license renewal and subject to AMR for hardening and loss of strength and for loss of material due to wear ~~Visual inspections are expected to identify loss of material due to general corrosion in accessible steel components. Loss of material due to pitting and crevice corrosion may not be detectable through these same visual inspections,; however, general corrosion is expected to be present and detectable such that, should pitting and crevice corrosion exist, general corrosion will is expected to manifest itself as visible rust or byproducts (e.g., discoloration or coating degradation). Loss of material due to corrosion is expected to be detectable prior to any loss of intended function. Therefore, this program is acceptable for use in inspecting for loss of material for general, pitting and crevice corrosion.~~

   ~~Surfaces that are inaccessible or not readily visible during plant operations are inspected during refueling outages. Surfaces that are inaccessible or not readily visible during both plant operations and refueling outages are inspected at such intervals that would provide reasonable assurance that the effects of aging will be managed such that applicable components will perform their intended function during the period of extended operation.~~

~~Surfaces that are insulated may be inspected when the external surface is exposed (i.e., maintenance) at such intervals that would provide reasonable assurance that the effects of aging will be managed such that applicable components will perform their intended function during the period of extended operation.~~

The program may also be credited with managing loss of material from internal surfaces of metals and with loss of material hardening and loss of strength from the internal surfaces of elastomers,, for situations in which material and environment combinations are the same for internal and external surfaces such that external surface condition is representative of internal surface condition. When credited, the program should describe the component internal environment and the credited similar external component environment inspected.

2. ***Preventive Actions:*** The External Surfaces Monitoring Program is a visual monitoring program that does not include preventive actions.

3. ***Parameters Monitored/Inspected*:** The External Surfaces Monitoring Program utilizes periodic plant system inspections ~~and walkdowns~~ to monitor for material degradation and leakage. This program inspects components such as piping, piping components, ducting, elastomer HVAC components, and other components. Coatings deterioration is an indicator of possible underlying degradation.

Examples of inspection parameters for metalics include the following~~:~~.

- corrosion and material wastage (loss of material)~~;~~
- leakage from or onto external surfaces~~;~~
- worn, flaking, or oxide-coated surfaces~~;~~
- corrosion stains on thermal insulation~~;~~
- protective coating degradation (cracking and flaking)

Examples of inspection parameters for elastomers include the following

- surface cracking, crazing,scuffing,and dimensional change (“ballooning” and “necking”)
- discoloration
- exposure of internal reinforcement for reinforced elastomers
- hardening as evidenced by a loss of suppleness during manipulation where the component and material are appropriate to manipulation .

4. ***Detection of Aging Effects:*** Degradation of steel surfaces cannot occur without the degradation of the paint or coating. Confirmation of the integrity of the paint or coating is an effective method for managing the effects of corrosion on the steel surface. For stainless steel surfaces a clean shiny surface is expected. The appearance of rust spots may indicate the loss of material on the stainless steel surface. For elastomers, a uniform surface texture and uniform color with no unanticipated dimentional change is expected. Any abnormal surface condition may be an indication of a aging effect for metals and for elastomers. A visual inspection is conducted for component surfaces at least once per refueling cycle. This frequency accommodates inspections of components that may be in locations that are normally only accessible during outages. System walkdowns are normally performed on a frequency ~~that exceeds~~of at least once per fuel cycle. Surfaces that are inaccessible or not readily visible during plant operations and refueling outages are inspected at such intervals that would ensure the components’

intended function is maintained. Surfaces that are insulated may be inspected when the external surface is exposed (i.e., maintenance) at such intervals that would ensure the components' intended function is maintained. The intervals of inspections may be adjusted as necessary based on plant-specific inspection results and industry operating experience.

Visual inspection will identify indirect indicators of elastomer hardening and loss of strength to include the presence of surface cracking, crazing, discoloration and, for elastomers with internal reinforcement, the exposure of reinforcing fibers, mesh or underlying metal. Visual inspection will identify direct indicators of loss of material due to elastomer wear to include dimensional change, scuffing and, for elastomers with internal reinforcement, the exposure of reinforcing fibers, mesh or underlying metal. Manipulation can be used to augment visual inspection to confirm the absence of hardening and loss of strength for elastomers, where appropriate, to the component and material (e.g. HVAC flexible connectors.) Hardening and loss of strength and loss of material due to wear for elastomers is expected to be detectable prior to any loss of intended function.

This program is credited with managing the following aging effects.

- loss of material for external surfaces~~;~~
- loss of material for internal surfaces exposed to the same environment as the external surface
- hardening and loss of strength for elastomers

**XI.M38 Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components**

XI.M38-1. AMP discusses periodic inspections and opportunistic inspections during maintenance and surveillance activities. Discussion in program element 4 should not specify periodic inspections since opportunistic inspections during corrective maintenance will not be periodic inspections.

XI.M38-2. See General Comment G-5. NUREG-1801 AMP is limited to steel components; AMP is used for other materials with corresponding aging effects and inspection techniques.

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## XI.M38 INSPECTION OF INTERNAL SURFACES IN MISCELLANEOUS PIPING AND DUCTING COMPONENTS

### Program Description

The program consists of inspections of the internal surfaces of ~~steel~~ metallic piping, piping components, ducting, and other components that are not covered by other aging management programs. These internal inspections are performed during the periodic system and component surveillances or during the performance of maintenance activities when the surfaces are made accessible for visual inspection. The program includes visual inspections to assure that existing environmental conditions are not causing material degradation that could result in a loss of component intended functions. If visual inspection of internal surfaces is not possible, then the applicant needs to provide a plant-specific program.

### Evaluation and Technical Basis

1. ***Scope of Program:*** The program visual inspections include internal surfaces of ~~steel~~ metallic piping, piping elements, ducting, and components in an internal environment (such as indoor uncontrolled air, condensation, and steam) that are not included in other aging management programs for loss of material. Inspections are performed when the internal surfaces are accessible during the performance of periodic surveillances~~,~~ or during maintenance activities ~~or during scheduled outages~~. This program includes indication of borated water leakage on internal surfaces.

…

3. ***Parameters Monitored/Inspected:*** Visual inspections of internal surfaces of plant components are performed during maintenance or surveillance activities. Parameters monitored or inspected include visible evidence of corrosion to indicate possible loss of materials.

4. ***Detection of Aging Effects:*** ~~Periodic~~ Visual inspections provide for detection of aging effects prior to the loss of component function. For painted or coated surfaces, degradation of ~~steel~~ metallic surfaces cannot occur without the degradation of the paint or coating. Confirmation of the integrity of the paint or coating is an effective method for managing the effects of corrosion on the ~~steel~~ metallic surface. The applicant should identify and justify the inspection technique used for detecting the aging effects of concern. Locations should be chosen to include conditions likely to exhibit these aging effects. Inspection intervals are established such that they provide timely detection of degradation.

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**<u>XI.M39 Lubricating Oil Analysis Program</u>**

XI.M39-1. Exceptions to inspections are taken routinely. Testing for flash point of lubricating oil is only needed for lubricating oil that could become contaminated by fuel or gasoline. Lubricating oil in many applications is not subject to this contamination (such as steam driven turbines or motor driven pumps) and testing for flash point is not needed. Revise NUREG-1801 to delete reference to flash point testing since flash point testing does not provide information related to the effects of aging.

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## XI.M39 LUBRICATING OIL ANALYSIS PROGRAM

### Program Description

The purpose of the Lubricating Oil Analysis Program is to ensure the oil environment in the mechanical systems is maintained to the required quality. The Lubricating Oil Analysis Program maintains oil systems contaminants (primarily water and particulates) within acceptable limits, thereby preserving an environment that is not conducive to loss of material, cracking or reduction of heat transfer. Lubricating oil testing activities include sampling and analysis of lubricating oil for detrimental contaminants. The presence of water or particulates may also be indicative of inleakage and corrosion product buildup.

### Evaluation and Technical Basis

…

**3.** ***Parameters Monitored/Inspected:*** For components with periodic oil changes in accordance with manufacturer's recommendations, a particle count and check for water are performed to detect evidence of abnormal wear rates, contamination by moisture, or excessive corrosion. For components that do not have regular oil changes, viscosity~~,~~ <u>and</u> neutralization number~~, and flash point~~ are also determined to verify the oil is suitable for continued use. In addition, analytical ferrography and elemental analysis are performed to identify wear particles.

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## XI.S1 ASME Section XI, Subsection IWE

XI.S1-1. See General Comment G-2. Details based on a specific ASME code may not be accurate for other editions.

XI.S1-2 Details of the IWE AMP are based on the 1992 edition instead of the 2001 edition of the ASME code. The following elements require update to the 2001 edition of the ASME Code.

Element 3 states, "Table IWE-2500-1 specifies seven categories for examination." A table in this element also shows the seven categories; EA, EB, EC, ED, EF, EG, and EP. The 2001 ed. w/ '02 & '03 add. only includes two examination categories; EA and EC.

Element 4 refers to VT-1 and VT-3 visual inspections. These terms are not used in the 2001 ed. w/ '02 & '03 add. of Subsection IWE. General Visual and Detailed Visual are used instead.

Element 5 states, "…flaws, areas of degradation, or repairs remain essentially unchanged for three consecutive inspection periods, these areas no longer require augmented examination…" The 2001 ed. w/ '02 & '03 add. only requires one unchanged inspection.

Elements 6, 7, and 8 all refer to Table IWE-3410-1 for acceptance standards. That table does not exist in the 2001 ed. w/ '02 & '03 add. Instead, IWE-3410 references IWE-3500 for these standards.

Note: Conforming change required for II.A3-7 and II.B4-7 and Table 1 item 3.5.1.16 to identify moisture barriers only

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## XI.S1 ASME SECTION XI, SUBSECTION IWE

### Program Description

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, Subsection IWE for steel containments (Class MC) and steel liners for concrete containments (Class CC). The full scope of IWE includes steel containment shells and their integral attachments; steel liners for concrete containments and their integral attachments; containment hatches and airlocks; seals, gaskets and moisture barriers; and pressure-retaining bolting. This evaluation covers the 2001 edition[31] including the 2002 and 2003 Addenda, as approved in 10 CFR 50.55a. ASME Code Section XI, Subsection IWE and the additional requirements specified in 10 CFR 50.55a(b)(2) constitute an existing mandated program applicable to managing aging of steel containments, steel liners of concrete containments, and other containment components for license renewal.

[3]An applicant may rely on a different version of the ASME Code~~, but should justify such use. An applicant may wish to refer to the SOC for an update of 10 CFR § 50.55a to justify use of a more recent edition of the Code~~. An edition of the ASME Code that has been endorsed by the NRC under 10 CFR 50.55a is acceptable and is considered consistent with NUREG-1801.

The primary ISI method specified in IWE is visual examination (general visual, VT-3, VT-1). Limited volumetric examination (ultrasonic thickness measurement) ~~and surface examination (e.g., liquid penetrant)~~ may also be necessary in some instances. ~~Bolt preload is checked by either a torque or tension test.~~ IWE specifies acceptance criteria, corrective actions, and expansion of the inspection scope when degradation exceeding the acceptance criteria is found.

The evaluation of 10 CFR 50.55a and Subsection IWE as an aging management program (AMP) for license renewal is provided below.

**Evaluation and Technical Basis**

1. ***Scope of Program**:* Subsection IWE-1000 specifies the components of steel containments and steel liners of concrete containments within its scope. The components within the scope of Subsection IWE are Class MC pressure-retaining components (steel containments) and their integral attachments; metallic shell and penetration liners of Class CC containments and their integral attachments~~; containment seals and gaskets;~~ moisture barriers, containment pressure-retaining bolting; and metal containment surface areas, including welds and base metal. The concrete portions of containments are inspected in accordance with Subsection IWL.

   Subsection IWE exempts the following from examination:

   (1) Components that are outside the boundaries of the containment as defined in the plant-specific design specification;
   (2) Embedded or inaccessible portions of containment components that met the requirements of the original construction code of record;
   (3) Components that become embedded or inaccessible as a result of vessel repair or replacement, provided IWE-1232 and IWE-5220 are met; and
   (4) Piping, pumps, and valves that are part of the containment system or that penetrate or are attached to the containment vessel (governed by IWB or IWC).

   10 CFR 50.55a(b)(2)(ix) specifies additional requirements for inaccessible areas. It states that the licensee is to evaluate the acceptability of inaccessible areas when conditions exist in accessible areas that could indicate the presence of or result in degradation to such inaccessible areas. Examination requirements for containment supports are not within the scope of Subsection IWE.

   ….

3. ***Parameters Monitored or Inspected:*** Table IWE-2500-1 specifies ~~seven~~ two categories for examination. The categories, parts examined, and examination methods are presented in the following table. ~~The first six examination categories (E-A through E-G) constitute the ISI requirements of IWE. Examination category E-P references 10 CFR Part 50, Appendix J leak rate testing. Appendix J leak rate testing is evaluated as a separate AMP for license renewal in XI.S4.~~

| CATEGORY | PARTS EXAMINED | EXAMINATION METHOD[a] |
|---|---|---|
| E-A | E1.11 Containment surface areas<br><br>E1.11 Pressure boundary bolted connections<br><br>E1.12 Pressure boundary wetted surfaces – submerged areas<br><br>E1.20 Pressure boundary BWR Vent system<br><br>E1. 30 Moisture Barriers | General visual~~, visual VT-3~~<br><br>Visual, VT-3<br><br>Visual, VT-3<br><br>Visual, VT-3<br><br>General Visual |
| ~~E-Bb~~ | ~~Pressure retaining welds~~ | ~~Visual VT-1~~ |
| E-C | E4.11 Containment surfaces requiring augmented examination<br><br>E4.12 Surface Area Grid | Visual, VT-1~~, volumetric~~<br><br>Volumetric (Ultrasonic Thickness) |
| ~~E-D~~ | ~~Seals, gaskets, and moisture barriers~~ | ~~Visual VT-3~~ |
| ~~E-Fb~~ | ~~Pressure retaining dissimilar metal welds~~ | ~~Surface~~ |
| ~~E-G~~ | ~~Pressure retaining bolting~~ | ~~Visual VT-1, bolt torque or tension test~~ |
| ~~E-P~~ | ~~All pressure-retaining components (pressure retaining boundary, penetration bellows, airlocks, seals, and gaskets)~~ | ~~10 CFR Part 50, Appendix J (containment leak rate testing)~~ |

**Table Notes:**

Weld examinations are included in Categories E-A as accessible surface area inspections.

Bolted connections examined per Item E1.11 require a VT-3 exam once per interval and each time the connection is disassembled during a scheduled E1.11 exam. Additionally, a VT-1 exam is performed if degradation or flaws are identified during the VT-3 exam. These additional examinations are required by 10 CFR 50.55a(b)(2)(ix)(G) and 10 CFR 50.55a(b)(2)(ix)(H).

Items E1.12 and E1.20 require VT-3 visual examination in lieu of General Visual examination as required by 10 CFR 50.55a(b)(2)(ix)(G).

Item E4.11 requires VT-1 visual examination in lieu of Detailed Visual examination as required by 10 CFR 50.55a(b)(2)(ix)(G).

Table IWE-2500-1 references the applicable section in IWE-3500 that identifies the aging effects that are evaluated. The parameters monitored or inspected depend on the particular examination category. For Examination Category ~~E-A~~E-C, as an example, ~~metallic surfaces (without coatings) are~~ noncoated areas shall be examined for evidence of cracking, discoloration, wear, pitting, excessive corrosion, arc strikes, gouges, surface discontinuities, dents, and other signs of surface irregularities. ~~For Examination Category E-D, seals, gaskets, and moisture barriers are examined for wear, damage, erosion, tear, surface cracks, or other defects that may violate the leak-tight integrity.~~

4. ***Detection of Aging Effects:*** The frequency and scope of examination specified in 10 CFR 50.55a and Subsection IWE ensure that aging effects would be detected before they would compromise the design-basis requirements. As indicated in IWE-2400, inservice examinations and pressure tests are performed in accordance with one of two inspection programs, A or B, on a specified schedule. Under Inspection Program A, there are four inspection intervals (at 3, 10, 23, and 40 years) for which 100% of the required examinations must be completed. Within each interval, there are various inspection periods for which a certain percentage of the examinations are to be performed to reach 100% at the end of that interval. ~~In addition, a general visual examination is performed once each inspection period.~~ After 40 years of operation, any future examinations will be performed in accordance with Inspection Program B. Under Inspection Program B, starting with the time the plant is placed into service, there is an initial inspection interval of 10 years and successive inspection intervals of 10 years each, during which 100% of the required examinations are to be completed. An expedited examination of containment is required by 10 CFR 50.55a in which an inservice (baseline) examination specified for the first period of the first inspection interval for containment is to be performed by September 9, 2001. Thereafter, subsequent examinations are performed every 10 years from the baseline examination. Regarding the extent of examination, all accessible surfaces receive a visual examination ~~such as General Visual, VT-1, or VT-3~~ (see table in item 3 above). IWE-1240 requires augmented examinations (Examination Category E-C) of containment surface areas subject to degradation. A VT-1 visual examination is performed for areas accessible from both sides, and volumetric (ultrasonic thickness measurement) examination is performed for areas accessible from only one side.

5. ***Monitoring and Trending:*** With the exception of inaccessible areas, all surfaces are monitored by virtue of the examination requirements on a scheduled basis. When component examination results require evaluation of flaws, evaluation of areas of degradation, or repairs, and the component is found to be acceptable for continued service, the areas containing such flaws, degradation, or repairs shall be reexamined during the next inspection period, in accordance with Examination Category E-C. When these reexaminations reveal that the flaws, areas of degradation, or repairs remain essentially unchanged for three consecutive inspection periods, these areas no longer require augmented examination in accordance with Examination Category E-C.

   IWE-~~2430~~ 2420 specifies that:
   (a) The sequence of component examinations established during the first inspection interval shall be repeated during each successive inspection interval, to the extent practical.
   (b) When examination results require evaluation of flaws or areas of degradation in accordance with IWE-3000, and the component is acceptable for continued service, the areas containing such flaws or areas of degradation shall be reexamined during the next inspection period listed in the schedule of the inspection program of IWE-2411 or IWE-2412. in accordance with Table IWE-2500- 1, Examination Category E-C.
   (c) When the reexaminations required by IWE-2420(b) reveal that the flaws or areas of degradation remain essentially unchanged for the next inspection period, these areas no longer require augmented examination in accordance with Table I WE-2500- 1.

   ~~(a~~b~~) examinations performed during any one inspection that reveal flaws or areas of degradation exceeding the acceptance standards are to be extended to include an additional number of examinations within the same category approximately equal to the initial number of examinations and (b~~c~~) when additional flaws or areas of degradation that exceed the acceptance standards are revealed, all of the remaining examinations within the same category are to be performed to the extent specified in Table IWE-2500-1 for the inspection interval. Reexamination required by IWE-2420(b) reveal that the flaws or areas of degradation remain essentially unchanged for the next inspection period, these areas no longer require augmented examination. Alternatives to these examinations are provided in 10 CFR 50.55a(b)(2)(ix)(D).~~

6. **Acceptance Criteria**: IWE-3000 provides acceptance standards for components of steel containments and liners of concrete containments. ~~Table IWE-3410-1 presents criteria to evaluate the acceptability of the containment components for service following the preservice examination and each inservice examination. This table~~ IWE-3500 specifies the acceptance standard for each examination category. Most of the acceptance standards rely on visual examinations. Areas that are suspect require an engineering evaluation or require correction by repair or replacement or by corrective measures in accordance with IWE-3122. For some examinations, such as augmented examinations, numerical values are specified for the acceptance standards. For the containment steel shell or liner, material loss exceeding 10% of the nominal containment wall thickness, or material loss that is projected to exceed 10% of the nominal containment wall thickness before the next examination, are documented. Such areas are to be accepted by engineering evaluation or corrected by repair or replacement in accordance with IWE-3122.

7. **Corrective Actions**: Subsection IWE states that components whose examination results indicate flaws or areas of degradation that do not meet the acceptance standards listed in ~~Table-3410-1~~ IWE-3500 are acceptable if an engineering evaluation indicates

that the flaw or area of degradation is nonstructural in nature or has no effect on the structural integrity of the containment. Except as permitted by 10 CFR 50.55a(b)(ix)(D), components that do not meet the acceptance standards are subject to additional examination requirements, and the components are repaired or replaced to the extent necessary to meet the acceptance standards of IWE-3000. For repair of components within the scope of Subsection IWE, IWE-3124 states that repairs and reexaminations are to comply with IWA-4000. IWA-4000 provides repair specifications for pressure retaining components including metal containments and metallic liners of concrete containments. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the corrective actions.

8. **Confirmation Process**: When areas of degradation are identified, an evaluation is performed to determine whether repair or replacement is necessary. If the evaluation determines that repair or replacement is necessary, Subsection IWE specifies confirmation that appropriate corrective actions have been completed and are effective. Subsection IWE states that repairs and reexaminations are to comply with the requirements of IWA-4000. Reexaminations are conducted in accordance with the requirements of IWA-2200, and the recorded results are to demonstrate that the repair meets the acceptance standards set forth in ~~Table IWE-3410-1~~ IWE-3500. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the confirmation process.

**<u>XI.S2 ASME Section XI, Subsection IWL</u>**

XI.S2-1. See General Comment G-2.

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## XI.S2 ASME SECTION XI, SUBSECTION IWL

### Program Description

10 CFR 50.55a imposes the examination requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section XI, Subsection IWL for reinforced and prestressed concrete containments (Class CC). The scope of IWL includes reinforced concrete and unbonded post-tensioning systems. This evaluation covers both the 1992 edition with the 2001 edition[1] including the 2002 and 2003 Addenda, as approved in 10 CFR 50.55a. ASME Code Section XI, Subsection IWL and the additional requirements specified in 10 CFR 50.55a(b)(2) constitute an existing mandated program applicable to managing aging of containment reinforced concrete and unbonded post-tensioning systems for license renewal.

The primary inspection method specified in IWL is visual examination (VT-3C, VT-1, VT-1C). For prestressed containments, tendon wires are tested for yield strength, ultimate tensile strength, and elongation. Tendon corrosion protection medium is analyzed for alkalinity, water content, and soluble ion concentrations. Prestressing forces are measured in selected sample tendons. IWL specifies acceptance criteria, corrective actions, and expansion of the inspection scope when degradation exceeding the acceptance criteria is found.

The evaluation of 10 CFR 50.55a and Subsection IWL as an aging management program (AMP) for license renewal is provided below.

### Evaluation and Technical Basis

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1. An applicant may rely on a different version of the ASME Code~~, but should justify such use. An applicant may wish to refer to the SOC for an update of 10 CFR § 50.55a to justify use of a more recent edition of the Code~~. <u>An edition of the ASME Code that has been endorsed by the NRC under 10 CFR 50.55a is acceptable and is considered consistent with NUREG-1801.</u>

**<u>XI.S3 ASME Section XI, Subsection IWF</u>**

XI.S3-1. See General Comment G-2.

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## XI.S3 ASME SECTION XI, SUBSECTION IWF

### Program Description

10 CFR 50.55a imposes the inservice inspection (ISI) requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, for Class 1, 2, 3, and MC piping and components and their associated supports. Inservice inspection of supports for ASME piping and components is addressed in Section XI, Subsection IWF. This evaluation covers the 2001 edition[1] including the 2002 and 2003 Addenda, as approved in 10 CFR 50.55a. ASME Code Section XI, Subsection IWF constitutes an existing mandated program applicable to managing aging of ASME Class 1, 2, 3, and MC supports for license renewal.

The IWF scope of inspection for supports is based on sampling of the total support population. The sample size varies depending on the ASME Class. The largest sample size is specified for the most critical supports (ASME Class 1). The sample size decreases for the less critical supports (ASME Class 2 and 3). Discovery of support deficiencies during regularly scheduled inspections triggers an increase of the inspection scope, in order to ensure that the full extent of deficiencies is identified. The primary inspection method employed is visual examination. Degradation that potentially compromises support function or load capacity is identified for evaluation. IWF specifies acceptance criteria and corrective actions. Supports requiring corrective actions are re-examined during the next inspection period.

The evaluation of Subsection IWF as an aging management program (AMP) for license renewal is provided below.

### Evaluation and Technical Basis

…

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1. An applicant may rely on a different version of the ASME Code~~, but should justify such use. An applicant may wish to refer to the SOC for an update of 10 CFR § 50.55a to justify use of a more recent edition of the Code~~. <u>An edition of the ASME Code that has been endorsed by the NRC under 10 CFR 50.55a is acceptable and is considered consistent with NUREG-1801.</u>

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## XI.S6 Structures Monitoring Program

XI.S6-1. See General Comment G-4. Reference to specific revision of an industry standard should be revised to allow later revisions or revisions to which a plant is committed. Specifically, this addresses reference to ACI standard 349.3 in Elements 3, 4 and 6.

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## XI.S6 STRUCTURES MONITORING PROGRAM

### Program Description

Implementation of structures monitoring under 10 CFR 50.65 (the Maintenance Rule) is addressed in Nuclear Regulatory Commission (NRC) Regulatory Guide (RG) 1.160, Rev. 2, and NUMARC 93-01, Rev. 2. These two documents provide guidance for development of licensee-specific programs to monitor the condition of structures and structural components within the scope of the Maintenance Rule, such that there is no loss of structure or structural component intended function.

Because structures monitoring programs are licensee-specific, the Evaluation and Technical Basis for this aging management program (AMP) is based on the implementation guidance provided in Regulatory Guide 1.160, Rev. 2, and NUMARC 93-01, Rev. 2. Existing licensee-specific programs developed for the implementation of structures monitoring under 10 CFR 50.65 are acceptable for license renewal provided these programs satisfy the 10 attributes described below.

If protective coatings are relied upon to manage the effects of aging for any structures included in the scope of this AMP, the structures monitoring program is to address protective coating monitoring and maintenance.

### Evaluation and Technical Basis

1. ***Scope of Program:*** The applicant specifies the structure/aging effect combinations that are managed by its structures monitoring program.

2. ***Preventive Action:*** No preventive actions are specified.

3. ***Parameters Monitored or Inspected:*** For each structure/aging effect combination, the specific parameters monitored or inspected are selected to ensure that aging degradation leading to loss of intended functions will be detected and the extent of degradation can be determined. Parameters monitored or inspected are to be commensurate with industry codes, standards and guidelines, and are to also consider industry and plant-specific operating experience. Although not required, ACI 349.3R-96 <u>(or revision to which the plant is committed)</u> and ANSI/ASCE 11-90 provide an acceptable basis for selection of parameters to be monitored or inspected for concrete and steel structural elements and for steel liners, joints, coatings, and waterproofing membranes (if applicable). If necessary for managing settlement and erosion of porous concrete subfoundations, the continued functionality of a site dewatering system is to be monitored. The plant-specific structures monitoring program is to contain sufficient detail on parameters monitored or inspected to conclude that this program attribute is satisfied.

4. ***Detection of Aging Effects:*** For each structure/aging effect combination, the inspection methods, inspection schedule, and inspector qualifications are selected to ensure that aging degradation will be detected and quantified before there is loss of intended functions. Inspection methods, inspection schedule, and inspector qualifications are to be commensurate with industry codes, standards and guidelines, and are to also consider industry and plant-specific operating experience. Although not required, ACI 349.3R-96 (or revision to which the plant is committed) and ANSI/ASCE 11-90 provide an acceptable basis for addressing detection of aging effects. The plant-specific structures monitoring program is to contain sufficient detail on detection to conclude that this program attribute is satisfied.

5. ***Monitoring and Trending:*** Regulatory Position 1.5, "Monitoring of Structures," in RG 1.160, Rev. 2, provides an acceptable basis for meeting the attribute. A structure is monitored in accordance with 10 CFR 50.65 (a)(2) provided there is no significant degradation of the structure. A structure is monitored in accordance with 10 CFR 50.65 (a)(1) if the extent of degradation is such that the structure may not meet its design basis or, if allowed to continue uncorrected until the next normally scheduled assessment, may not meet its design basis.

6. ***Acceptance Criteria:*** For each structure/aging effect combination, the acceptance criteria are selected to ensure that the need for corrective actions will be identified before loss of intended functions. Acceptance criteria are to be commensurate with industry codes, standards and guidelines, and are to also consider industry and plant-specific operating experience. Although not required, ACI 349.3R-96 (or revision to which the plant is committed) provides an acceptable basis for developing acceptance criteria for concrete structural elements, steel liners, joints, coatings, and waterproofing membranes. The plant-specific structures monitoring program is to contain sufficient detail on acceptance criteria to conclude that this program attribute is satisfied.

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**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, 2005.

ACI Standard 349.3R-96, Evaluation of Existing Nuclear Safety-Related Concrete Structures, American Concrete Institute.

ANSI/ASCE 11-90, Guideline for Structural Condition Assessment of Existing Buildings, American Society of Civil Engineers.

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

NUMARC 93-01, Rev. 2, Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants (Line-In/Line-Out Version), Nuclear Energy Institute, April 1996.

**<u>XI.S7 RG 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants</u>**

XI.S7-1. See General Comment G-4. Reference to specific revision of an industry standard should be revised to allow later revisions or revisions to which a plant is committed. Specifically, reference to ACI standard 349.3 in Element 6.

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## XI.S7 RG 1.127, INSPECTION OF WATER-CONTROL STRUCTURES ASSOCIATED WITH NUCLEAR POWER PLANTS

### Program Description

Nuclear Regulatory Commission (NRC) Regulatory Guide (RG) 1.127, Revision 1, "Inspection of Water-Control Structures Associated with Nuclear Power Plants," describes an acceptable basis for developing an inservice inspection and surveillance program for dams, slopes, canals, and other water-control structures associated with emergency cooling water systems or flood protection of nuclear power plants. The RG 1.127 program addresses age-related deterioration, degradation due to extreme environmental conditions, and the effects of natural phenomena that may affect water-control structures. The RG 1.127 program recognizes the importance of periodic monitoring and maintenance of water-control structures so that the consequences of age-related deterioration and degradation can be prevented or mitigated in a timely manner.

RG 1.127 provides detailed guidance for the licensee's inspection program for water-control structures, including guidance on engineering data compilation, inspection activities, technical evaluation, inspection frequency, and the content of inspection reports. Water-control structures covered by the RG 1.127 program include concrete structures; embankment structures; spillway structures and outlet works; reservoirs; cooling water channels and canals, and intake and discharge structures; and safety and performance instrumentation. RG 1.127 delineates current NRC practice in evaluating inservice inspection programs for water-control structures. The attributes of an acceptable aging management program (AMP) for license renewal are described below.

For plants not committed to RG 1.127, Revision 1, aging management of water-control structures may be included in the Structures Monitoring Program (XI.S6). Even if plant is committed to RG 1.127, Revision 1, aging management of certain structures and components may be included in the Structures Monitoring Program (XI.S6). However, details pertaining to water-control structures are to incorporate the attributes described herein.

### Evaluation and Technical Basis

1. ***Scope of Program***: RG 1.127 applies to water-control structures associated with emergency cooling water systems or flood protection of nuclear power plants. The water-control structures included in the RG 1.127 program are concrete structures; embankment structures; spillway structures and outlet works; reservoirs; cooling water channels and canals, and intake and discharge structures; and safety and performance instrumentation.

2. ***Preventive Action:*** No preventive actions are specified; RG 1.127 is a monitoring program.

3. ***Parameters Monitored or Inspected:*** RG 1.127 identifies the parameters to be monitored and inspected for water-control structures. The parameters vary depending on the particular

structure. Parameters to be monitored and inspected for concrete structures include cracking, movements (e.g., settlement, heaving, deflection), conditions at junctions with abutments and embankments, erosion, cavitation, seepage, and leakage. Parameters to be monitored and inspected for earthen embankment structures include settlement, depressions, sink holes, slope stability (e.g., irregularities in alignment and variances from originally constructed slopes), seepage, proper functioning of drainage systems, and degradation of slope protection features. Further details of parameters to be monitored and inspected for these and other water-control structures are specified in Section C.2 of RG 1.127.

4. ***Detection of Aging Effects:*** Visual inspections are primarily used to detect degradation of water-control structures. In some cases, instruments have been installed to measure the behavior of water-control structures. RG 1.127 indicates that the available records and readings of installed instruments are to be reviewed to detect any unusual performance or distress that may be indicative of degradation. RG 1.127 describes periodic inspections, to be performed at least once every five years. Similar intervals of five years are specified in ACI 349.3R for inspection of structures continually exposed to fluids or retaining fluids. Such intervals have been shown to be adequate to detect degradation of water-control structures before they have a significant effect on plant safety. RG 1.127 also describes special inspections immediately following the occurrence of significant natural phenomena, such as large floods, earthquakes, hurricanes, tornadoes, and intense local rainfalls.

5. ***Monitoring and Trending:*** Water-control structures are monitored by periodic inspection as described in RG 1.127. In addition to monitoring the aging effects identified in Attribute (3) above, inspections also monitor the adequacy and quality of maintenance and operating procedures. RG 1.127 does not discuss trending.

6. ***Acceptance Criteria:*** Acceptance criteria to evaluate the need for corrective actions are not specified in RG 1.127. However, the “Evaluation Criteria” provided in Chapter 5 of ACI 349.3R-96 provides acceptance criteria (including quantitative criteria) for determining the adequacy of observed aging effects and specifies criteria for further evaluation. Although not required, plant-specific acceptance criteria based on Chapter 5 of ACI 349.3R-96 (or revision to which the plant is committed) are acceptable. Acceptance criteria for earthen structures such as dams, canals, and embankments are to be consistent with programs falling within the regulatory jurisdiction of the Federal Energy Regulatory Commission (FERC) or the U.S. Army Corps of Engineers.

7. ***Corrective Actions:*** RG 1.127 recommends that the licensee's inservice inspection and surveillance program include periodic inspections of water-control structures to identify deviations in structural conditions due to age-related deterioration and degradation from the original design basis. When findings indicate that significant changes have occurred, the conditions are to be evaluated. This includes a technical assessment of the causes of distress or abnormal conditions, an evaluation of the behavior or movement of the structure, and recommendations for remedial or mitigating measures. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the corrective actions.

8. ***Confirmation Process:*** As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the confirmation process.

9. ***Administrative Controls:*** As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the administrative controls.

10. ***Operating Experience:*** Degradation of water-control structures has been detected, through RG 1.127 programs, at a number of nuclear power plants, and in some cases, it has required remedial action. No loss of intended functions has resulted from these occurrences. Therefore, it can be concluded that the inspections implemented in accordance with the guidance in RG 1.127 have been successful in detecting significant degradation before loss of intended function occurs.

NOTE: For dam inspection and maintenance, programs under the regulatory jurisdiction of FERC or the U.S. Army Corps of Engineers, continued through the period of extended operation, will be adequate for the purpose of aging management. For programs not falling under the regulatory jurisdiction of FERC or the U.S. Army Corps of Engineers, the staff will evaluate the effectiveness of the aging management program based on compatibility to the common practices of the FERC and Corps programs.

**References**

ACI Standard 349.3R-96, Evaluation of Existing Nuclear Safety-Related Concrete Structures, American Concrete Institute.

NRC Regulatory Guide 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants, Revision 1, U.S. Nuclear Regulatory Commission, March 1978.

## XI.S8 Protective Coating Monitoring and Maintenance Program

XI.S8-1. See General Comment G-4. Reference to specific revision of an industry standard should be revised to allow later revisions or revisions to which a plant is committed.

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## XI.S8 PROTECTIVE COATING MONITORING AND MAINTENANCE PROGRAM S

### Program Description

Proper maintenance of protective coatings inside containment (defined as Service Level I in Nuclear Regulatory Commission [NRC] Regulatory Guide [RG] 1.54, Rev. 1) is essential to ensure operability of post-accident safety systems that rely on water recycled through the containment sump/drain system. Degradation of coatings can lead to clogging of strainers, which reduces flow through the sump/drain system. This has been addressed in NRC Generic Letter (GL) 98-04.

Maintenance of Service Level I coatings applied to carbon steel surfaces inside containment (e.g., steel liner, steel containment shell, penetrations, hatches) also serves to prevent or minimize loss of material due to corrosion. Regulatory Position C4 in RG 1.54, Rev. 1, describes an acceptable technical basis for a Service Level I coatings monitoring and maintenance program that can be credited for managing the effects of corrosion for carbon steel elements inside containment. The attributes of an acceptable program are described below.

A comparable program for monitoring and maintaining protective coatings inside containment, developed in accordance with RG 1.54, Rev. 0 or the American National Standards Institute (ANSI) standards (since withdrawn) referenced in RG 1.54, Rev. 0, and coatings maintenance programs described in licensee responses to GL 98-04, is also acceptable as an aging management program (AMP) for license renewal.

### Evaluation and Technical Basis

1. ***Scope of Program:*** The minimum scope of the program is Service Level I coatings, defined in RG 1.54, Rev 1, as follows: "Service Level I coatings are used in areas inside the reactor containment where the coating failure could adversely affect the operation of post-accident fluid systems and thereby impair safe shutdown."

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**XI.E1 Electrcial Cables and Connections Not Subject to 10CFR50.49 Environmental Qualification Requirements**

XI.E1-1. See General Comment G-4. Reference to specific revision of an industry standard should be revised to update to a later revision. GALL Electrical AMPs show a pending version of the IEEE Std. P1205-2000, "IEEE Guide for Assessing, Monitoring and Mitigating Aging Effects on Class 1E Equipment Used in Nuclear Power Generating Stations". This standard is no longer pending.

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## XI.E1 ELECTRICAL CABLES AND CONNECTIONS NOT SUBJECT TO 10 CFR 50.49 ENVIRONMENTAL QUALIFICATION REQUIREMENTS

### Program Description

In most areas within a nuclear power plant, the actual ambient environments (e.g., temperature, radiation, or moisture) are less severe than the plant design environment. However, in a limited number of localized areas, the actual environments may be more severe than the plant design environment for those areas. Conductor insulation materials used in cables and connections may degrade more rapidly than expected in these adverse localized environments. An adverse localized environment is a condition in a limited plant area that is significantly more severe than the specified service environment for the cable. An adverse variation in environment is significant if it could appreciably increase the rate of aging of a component or have an immediate adverse effect on operability.

The purpose of the aging management program described herein is to provide reasonable assurance that the intended functions of electrical cables and connections that are not subject to the environmental qualification requirements of 10 CFR 50.49 and are exposed to adverse localized environments caused by heat, radiation, or moisture will be maintained consistent with the current licensing basis through the period of extended operation. This program considers the technical information and guidance provided in NUREG/CR-5643, IEEE Std. ~~P1205,~~ 1205, SAND96-0344, and EPRI TR-109619.
…

### References

EPRI TR-109619, *Guideline for the Management of Adverse Localized Equipment Environments,* Electric Power Research Institute, Palo Alto, CA, June 1999.

IEEE Std. ~~P1205,~~ 1205-2000, *IEEE Guide for Assessing, Monitoring and Mitigating Aging Effects on Class 1E Equipment Used in Nuclear Power Generating Stations.*

**<u>XI.E2 Electrcial Cables and Connections Not Subject to 10CFR50.49 Environmental Qualification Requirements Used in Instrument Circuits</u>**

XI.E2-1. See General Comment G-4. Reference to specific revision of an industry standard should be revised to update to a later revision. GALL Electrical AMPs show a pending version of the IEEE Std. P1205-2000, "IEEE Guide for Assessing, Monitoring and Mitigating Aging Effects on Class 1E Equipment Used in Nuclear Power Generating Stations". This standard is no longer pending.

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**XI. E2 ELECTRICAL CABLES AND CONNECTIONS NOT SUBJECT TO 10 CFR 50.49 ENVIRONMENTAL QUALIFICATION REQUIREMENTS USED IN INSTRUMENTATION CIRCUITS**

**Program Description**

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The purpose of the aging management program described herein is to provide reasonable assurance that the intended functions of electrical cables that are not subject to the environmental qualification requirements of 10 CFR 50.49 and are used in instrumentation circuits with sensitive, high voltage, low-level signals exposed to adverse localized environments caused by heat, radiation or moisture will be maintained consistent with the current licensing basis through the period of extended operation. This program considers the technical information and guidance provided in NUREG/CR-5643, IEEE Std. ~~P1205,~~ <u>1205</u>, SAND96-0344, and EPRI TR-109619.

**References**

EPRI TR-109619, *Guideline for the Management of Adverse Localized Equipment Environments,* Electric Power Research Institute, Palo Alto, CA, June 1999.

IEEE Std. ~~P1205,~~ <u>1205</u>-2000, *IEEE Guide for Assessing, Monitoring and Mitigating Aging Effects on Class 1E Equipment Used in Nuclear Power Generating Stations.*

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**<u>XI.E3 Inaccessible Medium Voltage Cables Not Not Subject to 10CFR50.49 Environmental Qualification Requirements</u>**

XI.E3-1. See General Comment G-4. Reference to specific revision of an industry standard should be revised to update to a later revision. GALL Electrical AMPs show a pending version of the IEEE Std. P1205-2000, "IEEE Guide for Assessing, Monitoring and Mitigating Aging Effects on Class 1E Equipment Used in Nuclear Power Generating Stations". This standard is no longer pending.

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## XI. E3 INACCESSIBLE MEDIUM-VOLTAGE CABLES NOT SUBJECT TO 10 CFR 50.49 ENVIRONMENTAL QUALIFICATION REQUIREMENTS

**Program Description**

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The purpose of the aging management program described herein is to provide reasonable assurance that the intended functions of inaccessible medium-voltage cables that are not subject to the environmental qualification requirements of 10 CFR 50.49 and are exposed to adverse localized environments caused by moisture while energized will be maintained consistent with the current licensing basis through the period of extended operation. An adverse localized environment is a condition in a limited plant area that is significantly more severe than the specified service environment for the cable. An adverse variation in environment is significant if it could appreciably increase the rate of aging of a component or have an immediate adverse effect on operability. This program considers the technical information and guidance provided in NUREG/CR-5643, IEEE Std. ~~P1205,~~ <u>1205</u> SAND96-0344, and EPRI TR-109619.

…

**References**

EPRI TR-103834-P1-2, *Effects of Moisture on the Life of Power Plant Cables,* Electric Power Research Institute, Palo Alto, CA, August 1994.

EPRI TR-109619, *Guideline for the Management of Adverse Localized Equipment Environments,* Electric Power Research Institute, Palo Alto, CA, June 1999.

IEEE Std. ~~P1205,~~ <u>1205</u>-2000, *IEEE Guide for Assessing, Monitoring and Mitigating Aging Effects on Class 1E Equipment Used in Nuclear Power Generating Stations.*

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## XI.E4 Metal Enclosed Bus

XI.E4-1. See General Comment G-4. Reference to specific revision of an industry standard should be revised to update to a later revision. GALL Electrical AMPs show a pending version of the IEEE Std. P1205-2000, "IEEE Guide for Assessing, Monitoring and Mitigating Aging Effects on Class 1E Equipment Used in Nuclear Power Generating Stations". This standard is no longer pending.

XI.E4-2 Element 3 Add the following sentence to Parameters Monitored/Inspected: "Accessible gaskets and sealants will be inspected for degradation which could permit water to enter the bus." AMR line-item LP-10 in Chapter VI and Table 6 ID #10 for electrical systems to be revised to show AMP Chapter XI.E4, "Metal Enclosed Bus" versus Chapter XI.S6, "Structures Monitoring Program".

XI.E4-3 The GALL AMP references EPRI TR-109619, "Guideline for the Management of Adverse Localized Equipment Environments". However, EPRI TR-109619 does not pertain to bus duct and the GALL AMP does not use the concept of adverse localized environments to drive its inspections. Delete EPRI TR-109619 from the reference section.

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## XI. E4 METAL ENCLOSED BUS

3. **Parameters Monitored/Inspected:** A sample of accessible bolted connections will be checked for loose connection. Alternatively, bolted connections covered with heat shrink tape, sleeving, insulating boots, etc., may be visually inspected for insulation material surface anomalies. This program provides for the inspection of the internal portion of the MEBs for cracks, corrosion, foreign debris, excessive dust buildup, and evidence of water intrusion. Accessible gaskets and sealants will be inspected for degradation which could permit water to enter the bus. The bus insulation will be inspected for signs of embrittlement, cracking, melting, swelling, or discoloration, which may indicate overheating or aging degradation. The internal bus supports will be inspected for structural integrity and signs of cracks.

### References

IEEE Std. ~~P1205,~~ 1205-2000, *IEEE Guide for Assessing, Monitoring and Mitigating Aging Effects on Class 1E Equipment Used in Nuclear Power Generating Stations*.

SAND 96-0344, *Aging Management Guideline for Commercial Nuclear Power Plants – Electrical Cable and Terminations*, prepared by Sandia National Laboratories for the U.S. Department of Energy, September 1996.

~~EPRI TR-109619, *Guideline for the Management of Adverse Localized Equipment Environments,* Electric Power Research Institute, Palo Alto, CA, June 1999.~~

## XI.E6 Electrcial Cable Connections Not Subject to 10CFR50.49 Environmental Qualification Requirements

XI.E6-1. See General Comment G-4. Reference to specific revision of an industry standard should be revised to update to a later revision. GALL Electrical AMPs show a pending version of the IEEE Std. P1205-2000, “IEEE Guide for Assessing, Monitoring and Mitigating Aging Effects on Class 1E Equipment Used in Nuclear Power Generating Stations”. This standard is no longer pending.

XI.E6-2 Markup for rollup of LR-ISG-2007-02 into AMP XI.E6. Note: markup also includes allowance for visual inspection of cable connections per NEI Letter dated October 18, 2007, “Subject: Proposed License Renewal Interim Staff Guidance LR-ISG-2007-02: Changes to Generic Aging Lesson Learned (GALL) Aging Management Program (AMP) XI.E6, Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements.”

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## XI. E6 ELECTRICAL CABLE CONNECTIONS SUBJECT TO 10 CFR 50.49 ENVIRONMENTAL QUALIFICATION REQUIREMENTS

### Program Description

Cable connections are used to connect cable conductors to other cables or electrical devices. Connections associated with cables within the scope of license renewal are part of this program. The most common types of connections used in nuclear power plants are splices (butt or bolted), crimp-type ring lugs, connectors, and terminal blocks. Most connections involve insulating material and metallic parts. This aging management program (AMP) focuses on the metallic parts of the ~~for~~ electrical cable connections. ~~(metallic parts)~~This program provides a one-time inspection, on a sampling basis, to confirm the absence of age-related degradation of cable connections due to: ~~account for the following aging stressors:~~ thermal cycling, ohmic heating, electrical transients, vibration, chemical contamination, corrosion, and oxidation.

Generic Aging Lessons Learned (GALL) XI.E1, “Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements,” manages the aging of insulating material but not the metallic parts of the electrical connections. GALL XI.E1 is based on only a visual inspection of accessible cables and connections. Visual inspection may ~~is~~ not be sufficient to detect the aging effects from thermal cycling, ohmic heating, electrical transients, vibration, chemical contamination, corrosion, and oxidation on the metallic parts of cable connections.

Electrical cable connections ~~Circuits~~ exposed to appreciable ohmic or ambient heating during operation may experience loosening related to repeated cycling of connected loads or of the ambient temperature environment. Different materials used in various cable system components can produce situations where stresses ~~existing~~ between these components change with repeated thermal cycling. For example, under loaded conditions, ~~appreciable~~ ohmic heating may raise the temperature of a compression termination and cable conductor well above the ambient temperature, thereby causing thermal expansion of both components. ~~Different~~ ~~t~~Thermal expansion coefficients of different materials may alter mechanical stresses between the components so that the termination may be impacted ~~tighten on the conductor~~. When the ~~load or~~ current is reduced, the affected components cool and contract. Repeated cycling in this fashion can cause ~~produce~~ loosening of the termination ~~, under ambient conditions,~~ and may lead to high electrical resistance joints or eventual separation of~~to~~ compression-type terminations. Threaded connectors~~, splices, and terminal blocks~~ may loosen if subjected to significant thermally induced stress and cycling.

Cable connections within the scope of license renewal should be tested or inspected at least once prior to the period of extended operation to provide an indication of the integrity of the cable connections. The specific type of test or inspection to be performed will be determined based on the type of connection and will ~~prior to the initial test, and is~~ ~~to~~ be a proven method ~~test~~ for detecting loose connections, such as thermography, contact resistance testing, visual inspection or other appropriate test~~ing~~ or inspection justified in the application.

This program, as described, can be thought of as a sampling program. The following factors shall be ~~are~~considered for sampling: voltage level ~~application~~ (~~high,~~medium and low voltage), circuit loading (high loading), and location (high temperature, high humidity, vibration, etc.). The technical basis for the sample selections should be~~is~~ documented. If an unacceptable condition or situation is identified in the selected sample, the corrective action program will be used to evaluate the condition and determine appropriate corrective action ~~a determination is made as to whether the same condition or situation is applicable to other connections not tested~~.

SAND 96-0344, "Aging Management Guidelines for Electrical Cable and Terminations," indicated loose terminations were identified by several plants. The major concern is that the failures of a deteriorated cable system (cables, connections including fuse holders, and penetrations) could prevent it from performing its intended function ~~might be induced during accident conditions~~. This program is not applicable to cable connections in harsh environments since they are already addressed by the requirements of ~~Since the connections are not subject to the environmental qualification requirements of~~ 10 CFR 50.49. Even though cable connections may not be exposed to harsh environments, loosening or high resistance of connection is a concern due to aging mechanisms discussed above~~, an aging management program is required to manage the aging effects. This program will ensure that electrical cable connections will perform their intended function for the period of extended operation~~.

**Evaluation and Technical Basis**

1. ***Scope of Program:*** Cable connections ~~associated with cables~~ within the scope of license renewal, which are external connections terminating at active or passive devices are in the scope ~~are part~~ of this program. Wiring connections internal to an active assembly are considered a part of the active assembly and therefore are not within the scope of this program. This program does not include high-voltage (>35 kV) switchyard connections. The cable connections covered under the EQ program are not included in the scope of this program, ~~regardless of their association with active or passive components~~.

2. ***Preventive Actions*****:** No actions are taken as part of this program to prevent or mitigate aging degradation.

3. ***Parameters Monitored/Inspected*****:** This program will focus on the metallic parts of the connection. The one-time inspection verifies that ~~monitoring includes~~ loosening of bolted connections or high resistance of cable connections due to thermal cycling, ohmic heating, electrical transients, vibration, chemical contamination, corrosion, and oxidation is not an aging effect that requires a periodic aging management program. A representative sample of electrical cable connections is tested or inspected. The following factors are to be considered for sampling: voltage level ~~application~~(~~high,~~ medium and low voltage), circuit

loading (high load), and location (high temperature, high humidity, vibration, etc.). The technical basis for the sample selected is to be documented.

4. ***Detection of Aging Effects*****:** A representative sample of electrical connections within the scope of license renewal will be tested or inspected at least once prior to the period of extended operation to confirm that there are no aging effects requiring management during the period of extended operation. ~~every 10 years.~~ Inspection and testing methods may include thermography, contact resistance testing, visual or other appropriate ~~testing~~ methods without removing the connection insulation such as heat shrink tape, sleeving, insulating boots, etc. The one-time inspection provides additional confirmation to support industry operating experience that shows electrical connections have not experienced a high degree of failures, and that existing installation and maintenance practices are effective. ~~This is an adequate period to preclude failures of the electrical connections since experience has shown that aging degradation is a slow process. A 10-year testing interval will provide two data points during a 20-year period, which can be used to characterize the degradation rate. The first tests for license renewal are to be completed before the period of extended operation~~.

5. ***Monitoring and Trending*****:** Trending actions are not included as part of this program because it is a one-time inspection program. ~~the ability to trend test results is dependent on the specific type of test chosen. However, test results that are trendable provide additional information on the rate of degradation.~~

6. ***Acceptance Criteria*****:** The acceptance criteria for each test or inspection are to be defined for~~by~~ the specific type of test or inspection performed and the specific type of cable connections tested or inspected.

7. ***Corrective Actions*****:** If test or inspection acceptance criteria are not met, the corrective action program will be used to perform an evaluation that will consider the extent of the condition, the indications of aging effect, and potential changes to the one-time inspection program. Corrective actions may include, but are not limited to sample expansion, increased inspection frequency, and replacement or repair of the affected cable connection components.. ~~An engineering evaluation is performed when the test acceptance criteria are not met in order to ensure that the intended functions of the cable connections can be maintained consistent with the current licensing basis. Such an evaluation is to consider the significance of the test results, the operability of the component, the reportability of the event, the extent of the concern, the potential root causes for not meeting the test acceptance criteria, the corrective action necessary, and the likelihood of recurrence. When an unacceptable condition or situation is identified, a determination is made on whether the same condition or situation is applicable to other in-scope cable connections not tested.~~ As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the corrective actions.

8. ***Confirmation Process*****:** As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the confirmation process.

9. ***Administrative Controls*****:** As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the administrative controls.

10. ***Operating Experience***: Electrical cable connections exposed to appreciable ohmic or ambient heating during operation may experience loosening caused by repeated cycling of connected loads or of the ambient temperature environment. There have been a limited number of age related failures of cable connections reported. This one-time inspection confirms the absence of aging degradation of metallic cable connections. ~~Operating experience has shown that loosening of connections and corrosion of connections are aging mechanisms that, if left unmanaged, could lead to a loss of electrical continuity and potential arcing or fire~~

**References**


EPRI TR-109619, Guideline for the Management of Adverse Localized Equipment Environments, Electric Power Research Institute, Palo Alto, CA, June 1999.

IEEE Std. ~~P1205~~1205-2000, *IEEE Guide for Assessing, Monitoring and Mitigating Aging Effects on Class 1E Equipment Used in Nuclear Power Generating Stations*.

NUREG/CR-5643, *Insights Gained From Aging Research,* U. S. Nuclear Regulatory Commission, March 1992.

SAND96-0344, *Aging Management Guideline for Commercial Nuclear Power Plants – Electrical Cable and Terminations*, prepared by Sandia National Laboratories for the U.S. Department of Energy, September 1996.

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

Staff's Response to the NEI White Paper on Generic Aging Lessons Learned (GALL) Report Aging Management Program (AMP) XI.E6, "Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements," dated March 16, 2007 (ADAMS Accession Number ML070400349).